September 20, 1988

Docket No. 50-305

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Mr. D. C. Hintz
Vice President - Nuclear Power
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Hintz:

The Commission has issued the enclosed Amendment No. 80 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated October 26, 1987 and supplemented June 16, 1988.

The amendment involves administrative changes that clarify existing specifications and increase the consistency within the 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/

Joseph G. Giitter, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 80 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

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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. 80
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 October 26, 1987 as clarified by letter dated June 16, 1988 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) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 80, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth E. Perkins

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 20, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 80

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 3.6-4	TS 3.6-4
TS 3.10-7	TS 3.10-7
TS 3.10-17	TS 3.10-17
TS 3.10-18	TS 3.10-18
TS 3.10-20	TS 3.10-20
TS 4.5-2	TS 4.5-2
Table TS 4.1-1 (page 2 of 5)	Table TS 4.1-1 (page 2 of 5)
Table TS 4.2-2	Table TS 4.2-2

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 Specification 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. This 10% shutdown margin prevents 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 2100 ppm[1] or more if the Containment System is to be disabled with the reactor pressure vessel open.

2. Not more than one inoperable full length rod shall be allowed at any time.

3. If reactor operation is continued with one inoperable full length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is made operable earlier. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

h. Rod Drop Time

At operating temperature and full flow, the drop time of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If drop time is > 1.8 seconds, the rod shall be declared inoperable.

i. Rod Position Deviation Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power or > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change greater than 10 percent of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

If the tilt ratio is greater than 1.09, and it is not due to a misaligned rod, the reactor shall be brought to a no-load condition; i.e., reactor power less than or equal to 5%, for investigation by flux mapping. Although the reactor may be maintained critical for flux mapping, the generator must be disconnected from the grid since the cause of the tilt condition is not known, or it cannot be readily corrected.

ROD MISALIGNMENT LIMITATIONS

During normal power operation it is desirable to maintain the rods in alignment with their respective banks to provide consistency with the assumption of the safety analyses, to maintain symmetric neutron flux and power distribution profiles, to provide assurance that peaking factors are within acceptable limits and to assure adequate shutdown margin.

Analyses have been performed which indicate that the above objectives will be met if the rods are aligned within the limits of Specification 3.10.e. A relaxation in those limits for power levels below 85% is allowable because of the increased margin in peaking factors and available shutdown margin obtained while operating at lower power levels. This increased flexibility is desirable to account for the non-linearity inherent in the rod position indication system and for the effects of temperature and power as seen on the rod position indication system.

Rod position measurement is performed through the effects of the rod drive shaft metal on the output voltage of a series of vertically stacked coils located above the head of the reactor pressure vessel. The rod position can be determined by the analog individual rod position indicators, the plant process computer which receives a voltage input from the conditioning module, or through the conditioning module output voltage via a correlation of rod position vs. voltage.

INOPERABLE ROD POSITION INDICATOR CHANNELS

The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

INOPERABLE ROD LIMITATIONS

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30 day period is provided for the re-analysis of all accidents sensitive to the changed initial condition.

ROD DROP TIME

The required drop time to dashpot entry is consistent with safety analysis.

DNB PARAMETERS

The DNB related accident analysis assumed as initial conditions that the T inlet was 4°F above nominal design or T_{avg} was 4°F above nominal design. The Reactor Coolant System pressure was assumed to be 30 psi below nominal design.

REFERENCES

(1) FSAR Section 4.3

(2) FSAR Section 4.4

(3) FSAR Section 14

(4) "Rod Misalignment Analysis," July 27, 1981, submitted to NRC with proposed Technical Specification Amendment 46 by letter from E.R. Mathews (WPSC) to D.G. Eisenhut (NRC) dated August 7, 1981.

2. Containment Vessel Internal Spray System

A. System tests shall be performed once every operating cycle or once every 18 months, whichever occurs first. The test shall be performed with the isolation valves in the supply lines at the containment blocked closed.

B. The spray nozzles shall be checked for proper functioning at least every five years using either air with telltales or smoke tests to determine that all nozzles are clear.

C. The test will be considered satisfactory if control board indications or visual observations indicate all components have operated satisfactorily.

3. Containment Fan-Coil Units

Each fan-coil unit shall be tested once every operating cycle or once every 18 months, whichever occurs first, to verify proper operation of the motor-operated service water outlet valves and the fan coil emergency discharge and associated backdraft dampers (RBV034-001 through RBV034-004).

b. Component Tests

1. Pumps

A. The safety injection pumps, residual heat removal pumps, and containment spray pumps shall be started and operated on recirculation flow quarterly during power operation and within one week after the plant is returned to power operation, if the test was not performed during plant shutdown.

B. Acceptable levels of performance shall be that the pumps start, reach their required developed heat at miniflow, and operate for at least fifteen minutes on the miniflow line.

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND
TEST OF INSTRUMENT CHANNELS
(Page 2 of 5)

Channel Description	Check	Calibrate	Test	Remarks
9. Analog Rod Position	S (1,2)	R**	R	1) With step counters 2) Following rod motion in excess of 24 steps when computer is out of service
10. Rod Position Bank Counters	S (1,2)	N.A.	R	1) With analog rod position 2) Following rod motion in excess of 24 steps when computer is out of service
11. Steam Generator Level	S	R**	M	
12. Steam Generator Flow Mismatch	S	R**	M	
13. Charging Flow	S	R	N.A.	
14. Residual Heat Removal Pump Flow	S (when in operation)	R	N.A.	
15. Boric Acid Tank Level	D	R	M	
16. Refueling Water Storage Tank Level	W	A	N.A.	
17. Volume Control Tank Level	S	R	N.A.	
18A. Containment Pressure (SIS signal)	S	R**	M(1)	1) Isolation Valve signal
18B. Containment Pressure (Steamline Isol)	S	R**	M	Narrow range containment pressure (-3.0, +3.0 psig excluded)
18C. Containment Pressure (Cont. Spray Act)	S	R**	M	
18D. Annulus Pressure (Vacuum Breaker)	N.A.	R**	R	

TABLE 4.2-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G. (2)	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G. (2)	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., (2) plug or repair defective tubes and inspect 2S tubes in the other S. G. (2) Prompt notification of the Commission. (1)	The other S.G.'s are C-1	None	N/A	N/A
			Some S.G.'s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S. G. and plug or repair defective tubes. Prompt notification of the Commission.(1)(2)	N/A	N/A

S= 6%/n Where n is the number of steam generators inspected during an inspection.

Notes: 1. Refer to Specification 4.2(b)(5)(c)

2. As allowed by TS 4.2.b.2.d, the second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 80 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 October 26, 1987, as supplemented and clarified June 16, 1988, the Wisconsin Public Service Corporation (WPSC), the licensee, submitted a proposed amendment to Facility Operating License No. DPR-43, for the Kewaunee Nuclear Power Plant. The proposed amendment involves editorial changes that clarify existing Technical Specifications (TS) and increase the consistency within technical specifications. The June submittal provided clarifying information related to the surveillances conducted on the Internal Containment Spray system (ICS) to verify operability.

2.0 DISCUSSION

In its application, the licensee stated that, "The technical specification amendment [was submitted as part of a] continuing effort to improve the Technical Specifications (TS) for the Kewaunee Nuclear Power Plant (KNPP)". Eight pages of TS would be affected by the proposed changes. The proposed changes are discussed in the following paragraphs.

Page TS 3.6-4, which describes the bases for TS 3.6 (KNPP Containment System), would be modified in two places. The last sentence of the second paragraph would be changed to clarify that the containment vessel is being described and the last paragraph on the page would be changed to reflect the refueling boron concentration required by TS 3.8.a.5. These are purely administrative changes intended to clarify existing TS and to establish consistency within TS.

On Page TS 3.10-7, the phrase "...after [greater than] 30 in. of control rod motion..." would be changed to "... after [greater than] 24 steps of control rod motion..." The purpose of this change is to establish, in TS, a consistent unit for referencing control rod motion. This more restrictive change would require operators to log individual rod position in the event that the rod deviation monitor is inoperable after more than 24 steps (or 15 inches) instead of after more than 30 inches.

Pages TS 3.10-17 and TS 3.10-18 would be changed to clarify the basis for TS 3.10-c, "Quadrant Power Tilt Limits." The changes described in the licensee's submittal are either editorial (e.g., replacing "utilizing" with "using") or add additional phrases for clarification and consistency.

On Page TS 3.10-20, the phrase "rod ± 7.5 inches" would be changed to "rod ± 12 steps." The purpose of this change, as stated previously for page TS 3.10-7, is to establish a consistent unit for referencing control rod motion. Because 7.5 inches is equivalent to 12 steps (one control rod step is equal to .625 inches), the change is purely editorial.

Page TS 4.5-2 would be changed to clarify surveillance requirements for the Internal Containment Spray (ICS) system. The phrase, "Operation of the system is initiated by tripping the normal actuation instrumentation" would be deleted. Actuation of the ICS system via the "normal actuation instrumentation" is an ambiguous expression that could be interpreted as initiation of the ICS system by injecting false pressure signals at the test signal injection points of the containment pressure transmitters. Since automatic start of the ICS requires (at least) three 1-out-of-2 containment pressure Hi-Hi signals in coincidence, a minimum of eight starts of the ICS system would be required to test each containment pressure channel. Eight automatic starts of the ICS system could lead to excessive wear on the pumps, since the pumps are isolated and run against their shutoff head during testing to prevent injection into containment. On June 16, 1988, the licensee submitted supplemental information that explains how the existing surveillance procedures for the ICS system are adequate for demonstrating operability of the ICS - including the ICS system initiation logic. The proposed change would not result in a change of present surveillance requirements (i.e., the ICS system functional test is currently initiated with the manual push buttons).

Presently items 9 and 10 of Table TS 4.1-1 require rod position to be logged when the plant computer is out of service following rod motion in excess of 6 inches. The proposed change would replace "six inches" with "24 steps." This change would be consistent with changes on pages TS 3.10-7 and TS 3.10-20. This proposed change is less conservative since 6 inches of rod motion is roughly equivalent to about 10 steps.

The last change would add a note to the bottom of Table TS 4.2-2 clarifying that second and third steam generator tube sample inspections during each inservice inspection may be less than the full length of the tube by concentrating the inspection on those areas of the tube sheet array and on those portions of the tube where imperfections were previously found. This change would not remove any existing inspection requirements and is purely administrative in nature.

3.0 EVALUATION

The changes proposed in Section 2 can be categorized as follows:

3.1 Changes only to the TS bases

Page TS 3.6-4
Page TS 3.10-17
Page TS 3.10-18

The NRC approves of the above changes that clarify and hence, improve the existing TS bases.

3.2 Administrative changes to TS

Page TS 3.10-20
Page TS 4.2-2
Page TS 4.5-2

The staff approves the straight-forward, administrative changes proposed to TS Page 3.10-20 and 4.2-2. Change to TS page 4.5-2 clarifies TS 4.5.a.2.A, making it consistent with existing surveillance procedures. Surveillance testing of the ICS pumps using the manual push buttons instead of the "normal actuation instrumentation" is acceptable because (1) combined procedures for testing the ICS system are adequate for demonstrating operability of the ICS, and (2) initiation of the test using the normal actuation instrumentation could lead to rapid degradation of the ICS pumps. Therefore, the staff approves the proposed administrative change to page TS 4.5-2.

3.3 Changes more restrictive

Page TS 3.10-7

This change, which will help improve TS consistency, is acceptable.

3.4 Changes less conservative

Page Table TS 4.1-1 (page 2 of 5)

Table TS 4.1-1 provides minimum frequencies (or requirements) for checks, calibrations and tests of instrument channels. For items 9 and 10 of this table (Analog Rod Position and Rod Position Bank Counters, respectively) it is stated that a channel check will be performed following rod motion of six inches whenever the plant computer is out of service. The six inches of rod motion would be determined from the digital control rod group demand position indication (or "step counters") which counts pulses generated in the rod drive control system--not from the analog rod position indication which uses a linear differential transformer to measure the position of each individual rod cluster.

Changing six inches (about 10 steps) to 24 steps is non-conservative because, in the event of a plant computer outage, it allows greater rod movement before a channel check is required. The plant computer provides the rod deviation alarm that alerts reactor operators to a potential rod misalignment and requires the operators to compare the digital and analog readings. Without the plant computer the operators would not receive this alarm, but would be required to compare the analog and digital readings based on the number of steps of rod movement. Of potential concern is whether there is an increased possibility of a significant rod misalignment condition, in the absence of a rod deviation monitor, before the reactor operators are required to compare the digital and analog readings.

Because of the "stepping" nature of control rod motion, it is highly unlikely that a demand for 24 steps of rod motion could result in greater than 24 steps of rod deviation. Furthermore, the reactor operators have instrumentation in the control room that accurately indicates when the rods have traveled 24 steps. This is particularly true for the digital rod step counters. Unlike the rod step counters, the analog rod position indicators are subject to inaccuracies due to non-linearity and the effects of temperature and power. Because of these potential inaccuracies a 12 step allowance is applied to the analog rod position indicators. It is inappropriate to apply this 12 step allowance to the digital rod position indication. Thus, because of the "stepping" nature of control rod motion and the relatively high accuracy of the digital rod position indicators, it is highly unlikely that 24 steps of indicated control rod motion could result in greater than 24 steps of rod deviation.

Additionally, in support of TS amendment 41, WPSC provided analyses of the impact of various single control rod cluster misalignments of 36 steps on peaking factors, reactivity worths, and thermal margins for cycle 7. The results for the analyses indicated that adequate conservatism exists in the bounding transient analyses to absorb the penalties associated with a 36 step rod misalignment. Thus, even if the 12 step allowance for analog rod position indication is coupled with a highly unlikely 24 step misalignment, there would be no increase in the risk to public health and safety.

In their submittal, WPSC stated that the proposed change is consistent with standard Westinghouse Technical Specifications. Standard Westinghouse Technical Specifications require an increased surveillance frequency for the rod position indicator (every 4 hours instead of every 12 hours) whenever the rod deviation monitor is inoperable--which for Kewaunee would be the case during a plant computer outage. In the event that a rod position indicator is inoperable, Standard Technical Specifications require that rod position be verified after 24 steps in one direction since the last determination of rod position. This aspect of Standard Technical Specifications is consistent with the proposed change.

In consideration of the factors discussed previously, the staff considers this change acceptable.

3.5 Summary

The staff has reviewed the previously described changes and concludes that these changes are administrative in nature and have no effect on safety, and are, therefore, acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the 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 the amendments.

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; 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: J. Giitter

Dated: September 20, 1988