

January 21, 1987

Docket No. 50-305

Mr. D. C. Hintz
Manager, 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. 71 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment is in response to your application dated August 1, 1986 and as supplemented October 1, 1986.

The amendment corrects errors, replaces obsolete references with current references and reinserts an inadvertently deleted requirement in the Technical Specifications. Section 2.C.(2) of the Operating License is being updated to agree with NRC License Amendment No. 47 issued on November 29, 1982 regarding deletion of the Appendix B Technical Specifications. This action completes our TAC No. 62085.

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's next regular biweekly notice publication in the Federal Register.

Sincerely,

/s/

Morton B. Fairtile, Project Manager
Project Directorate #1
Division of PWR Licensing-A

8702060249 870121
PDR ADDCK 05000305 PDR
P

Enclosures:

1. Amendment No. 71 to License No. DPR-43
2. Safety Evaluation
3. Section 2.C.(2) of the OL

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE
Office: LA/PAD#1:DPLA
Surname: *PShuttleworth
Date: 12/23/86

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1/21/87

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Mr. D. C. Hintz
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

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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.71
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 August 1, 1986 and as supplemented October 1, 1986, 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:

8702060262 870121
PDR ADOCK 05000305
P PDR

(2) Technical Specifications

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

Morton B. Fairtile

Morton B. Fairtile, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 21, 1987

- C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: Part 20, Section 30.34 of Part 30 Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensees are authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts (thermal).

(2) Technical Specifications

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

Amdt.
No. 47
11-29-82

- (3) The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1, 3.1.2 and 3.1.4 through 3.1.28, of the Fire Protection Safety Evaluation Report. These modifications shall be completed by the dates specified in Table 3.1. Dates for resolution of items are specified in Table 3.2. In the event that these dates for completion cannot be met, the licensee shall submit a report explaining the circumstances and propose a revised schedule.

Amdt.
No. 39
4-21-82

ATTACHMENT TO LICENSE AMENDMENT NO. 71

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 the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE

v
viii
2.1-2
2.3-3
2.3-6
3.2-2
3.3-6
3.4-1a
3.4-3
3.5-6
3.5-7
3.10-1
Table TS 3.5-1 (P.2 of 2)
Table TS 3.5-2 (P.1 of 3)
Table TS 3.5-2 (P.2 of 3)
Table TS 3.5-2 (P.3 of 3)
Table TS 3.5-3 (P.1 of 2)
Table TS 3.5-6
Table TS 4.1-2 (P.1 of 2)
6-1
6-4
6-5
6-10
6-20
6-23
7/8-9
Table TS 8-4 (P.3 of 3)

INSERT

v
viii
2.1-2
2.3-3
2.3-6
3.2-2
3.3-6
3.4-1a
3.4-3
3.5-6
3.5-7
3.10-1
Table TS 3.5-1 (P.2 of 2)
Table TS 3.5-2 (P.1 of 3)
Table TS 3.5-2 (P.2 of 3)
Table TS 3.5-2 (P.3 of 3)
Table TS 3.5-3 (P.1 of 2)
Table TS 3.5-6
Table TS 4.1-2
6-1
6-4
6-5
6-10
6-20
6-23
7/8-9
Table TS 8-4 (P.3 of 3)

<u>Section</u>	<u>Title</u>	<u>Page TS</u>
6.7	Safety Limit Violation	6-12
6.8	Procedures	6-12
6.9	Reporting Requirements	6-13
6.9.1	Routine Reports	6-13
6.9.1.a	Start-up Report	6-13
6.9.1.b	Annual Reporting Requirements	6-14
6.9.1.c	Monthly Operating Report	6-15
6.9.2	Deleted	
6.9.3	Unique Reporting Requirements	6-16
6.9.3.a	Annual Radiological Environmental Monitoring Reports	6-16
6.9.3.b	Semiannual Radioactive Effluent Release Report	6-17
6.9.3.c	Special Reports	6-19
6.9.3.d	Deleted	
6.10	Record Retention	6-20
6.11	Radiation Protection Program	6-21
6.12	System Integrity	6-21
6.13	High Radiation Area	6-22
6.14	Postaccident Sampling and Monitoring	6-22
6.15	Secondary Water Chemistry	6-23
6.16	Radiological Effluents	6-23
6.17	Process Control Program (PCP)	6-23
6.18	Offsite Dose Calculation Manual (ODCM)	6-24
6.19	Major Changes to Radioactive Liquid, Gaseous and Solid Waste Treatment	6-25
7/8.0	Radiological Effluent Technical Specifications and Surveillance Requirements	7/8-1
7/8.1	Radioactive Liquid Effluent Monitoring Instrumentation	7/8-2
7/8.2	Radioactive Gaseous Effluent Monitoring Instrumentation	7/8-3
7/8.3	Liquid Effluents	7/8-4
7/8.3.1	Concentration	7/8-4
7/8.3.2	Dose	7/8-5
7/8.3.2	Liquid Radwaste Treatment	7/8-6
7/8.4	Gaseous Effluents	7/8-7
7/8.4.1	Dose Rate	7/8-7
7/8.4.2	Dose - Noble Gases	7/8-8
7/8.4.3	Dose - Iodine-131, Iodine-133 and Radionuclides In Particulate Form	7/8-9
7/8.4.4	Gaseous Radwaste Treatment System	7/8-10
7/8.5	Solid Radioactive Waste	7/8-11
7/8.6	Total Dose	7/8-12
7/8.7	Radiological Environmental Monitoring	7/8-14
7/8.7.1	Monitoring Program	7/8-14
7/8.7.2	Land Use Census	7/8-16
7/8.7.3	Interlaboratory Comparison Program	7/8-18
7/8.8	Basis	7/8-19

LIST OF FIGURES

<u>Figure - TS</u>	<u>Titles</u>
2.1-1	Safety Limits Reactor Core, Thermal and Hydraulic
3.1-1	Reactor Coolant System Heat-up Limitations
3.1-2	Reactor Coolant System Cool-down Limitations
3.1-3	Deleted
3.1-4	Deleted
3.10-1	Required Shut-down Reactivity vs. Reactor Boron Concentration
3.10-2	Hot Channel Factor Normalized Operating Envelope
3.10-3	Control Rod Insertion Limits as a Function of Power
3.10-4	Permissible Operating Band on Indicated Flux Difference as a Function of Burn-up (Typical)
3.10-5	Target Band on Indicated Flux Difference as a Function of Operating Power Level (Typical)
3.10-6	V(Z) as a function of Core Height
3.10-7	Deleted
6.2-1	Functional Organization - Power Supply and Engineering Department - Wisconsin Public Service Corporation
6.2-2	Kewaunee Nuclear Power Plant Organization Chart Unit No. 1 - Wisconsin Public Service Corporation

heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.(1)

The curves of Figure TS 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNBR ratio is equal to 1.3 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches 1.3 and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is assured is below these lines.

The curves are based on the following nuclear hot channel factors:

$$F_{\Delta H}^N = 1.55 \quad F_Q^N = 2.51$$

and include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F_{\Delta H}^N = 1.55 [1 + 0.2 (1 - P)] \text{ where } P \text{ is the fraction of rated power}$$

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in Specification 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure TS 3.10-3 insure that the DNBR is always greater at partial power than at full power.

4. Reactor Coolant Flow
 - A. Low reactor coolant flow per loop $\geq 90\%$ of normal indicated flow as measured by elbow taps.
 - B. Reactor coolant pump motor breaker open
 - 1) Low frequency set point ≥ 57.5 Hz
 - 2) Low voltage set point $\geq 75\%$ of normal voltage
5. Steam Generators

Low-low steam generator water level $\geq 5\%$ 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 $\geq 75\%$ of normal voltage
 - Turbine Trip
 - Manual Trip
 - Safety Injection Trip (Refer to table TS 3.5-1 for trip settings)

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. (5)

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 set points 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 set points will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10 percent of power.

Table TS 3.5-1 lists the various parameters and their set points 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 Sec. 3.5 of these specifications for details of SIS signals.

References

- | | |
|--------------------------|-------------------------|
| (1) FSAR Section 14.1.1 | (2) USAR, Page 14.1-5 |
| (3) FSAR Section 14.3.1 | (4) FSAR Section 14.1.8 |
| (5) FSAR Section 14.1.10 | (6) WCAP-8092 |

- c. Any one of the following conditions of inoperability may exist during the time intervals specified. The reactor shall be placed in the hot shutdown condition if operability is not restored within the time specified, and it shall be placed in the cold shutdown condition if operability is not restored within an additional 48 hours.
1. ONE of the operable charging pumps may be removed from service provided two pumps are again operable within 24 hours.
 2. ONE boric acid transfer pump may be out of service provided both pumps are operable within 24 hours.
 3. ONE channel of heat tracing may be out of service provided it is restored to operable status within 48 hours.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with any one of the two boric acid transfer pumps. An alternate method of boration will be use of the charging pumps directly from the Refueling Water Storage Tank. A third method will be to use the safety injection pumps. There are two sources of borated water available for injection through 3 different paths.

- (1) The boric acid transfer pumps can deliver the boric acid tank contents to the suction of the charging pumps.
- (2) The charging pumps can take suction directly from the Refueling Water Storage Tank containing a concentration of 1950 ppm boron solution. Reference is made to Specification 3.3.b.1.A.

d. Component Cooling System

1. The reactor shall not be made or maintained critical unless the following conditions are satisfied, except for low power physics tests and except as provided by Specification 3.3.d.2.
 - A. TWO component cooling water trains are operable with each train consisting of:
 1. ONE component cooling water pump
 2. ONE component cooling water heat exchanger
 3. An operable flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions.
2. During power operation or recovery from an inadvertent trip, ONE component cooling water train may be inoperable for a period of 72 hours. If operability is not restored within 72 hours, then within 1 hour action shall be initiated to:
 - Achieve Hot Standby within the next 6 hours.
 - Achieve Hot Shutdown within the following 6 hours.
 - Achieve Cold Shutdown within an additional 36 hours.

e. Service Water System

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by Specification 3.3.e.2.
 - A. TWO service water trains are operable with each train consisting of:

- b. If, when the reactor is above 350°F, any of the conditions of Specification 3.4.a cannot be met within 48 hours, and except for the conditions of 3.4.c, the reactor shall be shut down and cooled below 350°F using normal operating procedures.
- c. When the reactor is above 350°F, one auxiliary feedwater pump may be inoperable provided the pump is restored to operable status within 72 hours, or the reactor shall be shutdown and cooled below 350°F using normal operating procedures.
- d. Reactor Power shall not exceed 50% of rated power unless two of the three turbine overspeed protection systems are operable. If two or more of the turbine overspeed protection systems are inoperable, then maintain power less than 50% of rated power. When only two systems are operable, an individual system may be blocked for no longer than 4 hours to allow for testing.

The secondary coolant activity is based on a postulated release of the contents of one steam generator to the atmosphere. This could happen, for example, as a result of a steam break accident combined with failure of a steam line isolation valve. The limiting dose for this case results from iodine-131 because of its low MPC, and because its long half-life relative to the other iodine isotopes results in its greater concentration in the liquid. The accident is assumed to occur at zero load when the steam generators contain maximum water. With allowance for plate-out retention in water droplets, one-tenth of the contained iodine is assumed released from the plant. The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot X/Q \cdot DCF$$

where: C = secondary coolant activity, 1.0 uCi/cc

V = water volume in one steam generator,
3510 ft³ = 99 m³

B(t) = breathing rate, 3.47 x 10⁻⁴ m³/sec

X/Q = 2.9 x 10⁻⁴ sec/m³

DCF = 1.48 x 10⁶ rem/Ci iodine-131 inhaled

The resultant dose is less than 1.5 rem.

Turbine Overspeed Protection

Turbine overspeed protection is provided to limit the possibility of turbine missiles. Overspeed protection is provided by three independent systems based on diverse operating principles. The three systems are the electro-hydraulic (E-H) system, the mechanical trip system, and the Redundant Overspeed Trip System (ROST). The E-H and mechanical systems are single channel and operate on a one-out-of-one to trip logic; the ROST system is a three channel system, requiring two out of three channels to trip.

References:

FSAR Section 10

FSAR Section 14.1

Each relay in the undervoltage protection channels will fail safe and is alarmed to alert the operator to the failure.

A blackout signal which occurs during the sequence loading following a safety injection signal will result in a reinitiation of the sequence loading logic at time step 0 as long as the Safety Injection signal has not been re-set. The Kewaunee Emergency Procedures warn the operators that a Blackout Signal occurring after reset of Safety Injection will not actuate the sequence loading and instructs to re-initiate Safety Injection if needed.

Instrument Operating Conditions

During plant operations, the complete protective instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection Systems, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three

circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in another channel.

The operability of the instrumentation noted in Table TS 3.5-6 assures that sufficient information is available on these selected plant parameters to aid the operator in identification of an accident and assessment of plant conditions during and following an accident. In the event the instrumentation noted in Table TS 3.5-6 is not operable, the operator is given instruction on compensatory actions.

References:

- (1) FSAR Section 7.5
- (2) FSAR Section 14.3
- (3) FSAR Section 14.2.5
- (4) Deleted

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objective

To ensure 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the hot shutdown margin shall be at least that shown in Figure TS 3.10-1. Shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron, or part length rod position.

b. Power Distribution Limits

1. At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

A. $\frac{N}{Q} F(Z)$ Limits:

(i) Westinghouse Electric Corporation Fuel

$$\frac{N}{Q} F(Z) \times 1.03 \times 1.05 \leq (2.14)/P \times K(Z) \text{ for } P > .5$$

$$\frac{N}{Q} F(Z) \times 1.03 \times 1.05 \leq (4.28) \times K(Z) \text{ for } P \leq .5$$

(ii) Exxon Nuclear Company Fuel

$$\frac{N}{Q} F(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z) \text{ for } P > .5$$

$$\frac{N}{Q} F(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z) \text{ for } P \leq .5$$

TABLE TS 3.5-1 (Page 2 of 2)

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

<u>NO.</u>	<u>FUNCTIONAL UNIT</u>	<u>CHANNEL</u>	<u>SETTING LIMIT</u>
9	Safeguards Bus Undervoltage (4)	Loss of Power	85.0% + 2% nominal bus voltage ≤ 2.5 second time delay
10	Safeguards Bus Second Level (5) Undervoltage	Degraded Grid Voltage	92.5% + 2% of nominal bus voltage ≤ 5 minutes time delay

- (1) Initiates containment isolation, feedwater line isolation shield building ventilation, auxiliary building special vent, and starting of all containment fans. In addition, the signal overrides any bypass on the accumulator valves.
- (2) Confirm main steam isolation valves closure within 5 seconds when tested.
d/p = differential pressure
- (3) The setting limits for max radiation levels are derived from the technical specification 7.4.1, Table E of the ODCM, and section 6.5 of the USAR.
- (4) This undervoltage protection channel ensures ESF equipment will perform as assumed in the FSAR.
- (5) This undervoltage protection channel protects ESF equipment from long term low voltage operation.

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP
(Page 1 of 3)

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	Manual	2	1	1	-		Maintain hot shutdown
2	Nuclear Flux Power Range ⁽³⁾						
	low setting	4(2)	2	3	1	P-10	Maintain hot shutdown
	high setting	4(2)	2	3	1		
	Positive Rate	4(2)	2	3	1		
	Negative Rate	4(2)	2	3	1		
3	Nuclear Flux Intermediate Range	2	1	1	-	P-10	Maintain hot shutdown - (1)
4	Nuclear Flux Source Range	2	1	1	-	P-6	Maintain hot shutdown - (1)
5	Overtemperature ΔT	4(2)	2	3	1		Maintain hot shutdown
6	Overpower ΔT	4(2)	2	3	1		Maintain hot shutdown
7	Low Pressurizer Pressure	4(2)	2	3	1	P-7	Maintain hot shutdown

TABLE TS 3.5-2
INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP
(Page 2 of 3)

NO.	FUNCTIONAL UNIT	1	2	3	4	5	6
		NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
8	High Pressurizer Pressure	3	2	2	-		Maintain hot shutdown
9	Pressurizer High Water Level	3	2	2	-	P-7	Maintain hot shutdown
10	Low Flow In one Loop	3/loop	2/loop (any loop)	2	-	P-8	Maintain hot shutdown
	Low Flow Both Loops	3/loop	2/loop (any loop)	2	-	P-7	Maintain Hot shutdown
11	Deleted						
12	Lo-Lo Steam Generator Water Level	3/loop	2/loop	2/loop	-		Maintain hot shutdown
13	Undervoltage 4-KV Bus	2/bus	1/bus (both buses)	1/bus	-	P-7	Maintain hot shutdown
14	Underfrequency 4-KV Bus(4)	2/bus	1/bus (both buses)	1/bus	-		Maintain hot shutdown
15	Deleted						
16	Steam Flow/Feedwater Flow Mismatch	2	1	1	-		Maintain hot shutdown

Table 3.5-2 (Page 2 of 3)

Amendment No. 51, 71

TABLE TS 3.5-2 (Page 3 of 3)

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

NOTES

- P-6: 1 of 2 Intermediate Range Nuclear Instrument Channels greater than 10^{-10} amps.
- P-7: 3 of 4 Power Range Nuclear Instrument channels less than 10% power AND 2 of 2 Turbine Impulse Pressure channels less than 10% power.
- P-8: 3 of 4 Power Range Nuclear Instrument Channels less than 10% power.
- P-10: 2 of 4 Power Range Nuclear Instrument Channels greater than 10% power.
- Note 1: When a block condition exists, maintain normal operation.
- Note 2: when one channel is out of service, a bypass may be used to allow testing other channels; a channel shall not be bypassed longer than 4 hours.
- Note 3: One additional channel may be taken out of service for zero power physics testing.
- Note 4: Underfrequency on the 4 kV Buses trips the Reactor Coolant Pump breakers, which in turn trips the reactor when power is above P-7.

TABLE TS 3.5-3 (Page 1 of 2)

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	SAFETY INJECTION						
	a. Manual	2	1	1	-		Hot Shutdown***
	b. High Containment Pressure	3	2	2	-		Hot Shutdown***
	c. Low Steam Pressure/Line	3	2	2	-	Primary pressure < 2000 psig	Hot Shutdown***
	d. Pressurizer Low Pressure	3	2	2	-	Primary pres- sure < 2000 psig	Hot Shutdown***
2	SELECTED BORIC ACID STORAGE TANK LEVEL	2 sets of 2	1 of 2 in each set	2 per set	1/set		One channel may be inoperable for 72 hours otherwise maintain cold shutdown
3	CONTAINMENT SPRAY						
	a. Manual	2	2	2	**		Hot Shutdown***
	b. Hi-Hi Containment Pressure (Containment Spray)	3 sets of 2	1 of 2 in each set	1 per set	1/set		Hot Shutdown***

(Deleted)

TABLE TS 3.5-6

INSTRUMENTATION OPERATING CONDITIONS FOR INDICATION

NO.	FUNCTIONAL UNIT	¹	²
		REQUIRED TOTAL NO. OF CHANNELS*	MINIMUM CHANNELS OPERABLE**
1	Auxiliary Feedwater Flow to Steam Generators (Narrow Range Level Indication already required operable by Tech Spec Table TS 3.5-2 Item 12)	1/steam gen	1/steam gen
2	Reactor Coolant System Subcooling Margin	2	1
3	Pressurizer Power Operated Relief Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve	1/valve
4	Pressurizer Power Operated Relief Block Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve	1/valve
5	Pressurizer Safety Valve Position (One Channel Temperature, and one Acoustic Sensor per valve)	2/valve	1/valve
6	Containment Water Level (wide range)	2	1
7	Containment Hydrogen Monitor	2	1
8	Containment Pressure Monitor (wide range)	2	1

*With the number of Operable monitoring instrumentation channels less than the Required Total Number of Channels shown, either restore the inoperable channels to Operable status within fourteen days, or be in at least Hot Shutdown within the next 12 hours.

**With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirements, either restore the minimum number of channels to Operable status within 72 hours or be in at least Hot Shutdown within the next 12 hours.

TABLE TS 4.1-2
MINIMUM FREQUENCIES FOR SAMPLING TESTS

<u>Sampling Tests</u>	<u>Test</u>	<u>Frequency</u>	<u>Maximum Time Between Tests (Days)</u>
1. Reactor Coolant Samples	Gross Beta-Gamma activity (excluding tritium)	5/week	3
	Tritium activity	Monthly	37
	*Chemistry (Cl, F, O ₂)	3/week	4
2. Reactor Coolant Boron ⁽¹⁾	*Boron concentration	2/week	5
3. Refueling Water Storage Tank Water Sample ⁽²⁾	Boron concentration	Monthly***	37
4. Boric Acid Tanks	Boron concentration	Weekly	8
5. Accumulator	Boron concentration	Monthly	37
6. Spent Fuel Pool	Boron concentration	Monthly**	37
7. Secondary Coolant	Gross Beta or Gamma activity	Weekly	8
	Iodine concentration	Weekly when gross Beta or Gamma activity <u>>1.0 uCi/cc</u>	8

Notes

- * See Spec 4.1.D
- ** Sample will be taken monthly when fuel is in the pool.
- *** And after adjusting tank contents.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager has overall on-site responsibility for plant operation. In the absence of the Plant Manager, the succession to this responsibility shall be in the following order:

- a. Assistant Manager-Plant Operations
- b. Assistant Manager-Plant Maintenance
- c. Superintendent-Plant Operations
- d. Assistant Manager-Plant Technical and Services
- e. Shift Supervisor

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for plant management and technical support shall be as shown on Figure TS 6.2-1.

FACILITY STAFF

6.2.2 The plant organization shall be as shown on Figure TS 6.2-2 and:

- a. Each on-duty shift complement shall consist of at least:
 - (1) One Shift Supervisor (SRO)
 - (2) Two licensed Reactor Operators
 - (3) One Auxiliary Operator
 - (4) One Equipment Operator
 - (5) One Radiation Technologist
- b. While above cold shutdown, the on-duty shift complement shall consist of the personnel required by 6.2.2a. above and an additional SRO.
- c. In the event that one of the shift members becomes incapacitated due to illness or injury or the Radiation Technologist has to accompany an injured person to the hospital, reactor operations may continue with the reduced complement until a replacement arrives. In all but severe weather conditions, a replacement is required within two hours.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

- a. Review of operating, maintenance and other procedures including emergency operating procedures which affect nuclear safety as determined by the plant manager. Changes to those procedures are made in accordance with the provisions of TS 6.8.1.
- b. Review of all proposed tests and experiments that affect nuclear safety.
- c. Review of all proposed changes to the Technical Specifications.
- d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.
- e. Review of all proposed changes to the Security Plan and Emergency Plan and their respective implementing procedures.
- f. Review all reports covering the investigation of all violations of the Technical Specifications and the recommendations to prevent recurrence.
- g. Review plant operations to detect potential safety hazards.
- h. Performance of special reviews and investigations and prepare reports thereon as requested by the Chairman of the Nuclear Safety Review and Audit Committee.

- 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 NUCLEAR ENGINEERING STAFF (CNES)

FUNCTION

- 6.5.2.1 The CNES shall function to provide engineering,

- f. Reports covering significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.
- g. Reports covering all Reportable Events.
- h. Reports covering any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.
- i. Reports and meeting minutes of the PORC.

AUDITS

- 6.5.3.8 Audits of plant activities shall be performed under the cognizance of the NSRAC; these audits shall include:
- a. Conformance of plant operation to the provisions contained within the Technical Specifications and applicable license conditions at least annually.
 - b. Performance, training, and qualifications of the entire plant staff at least annually.
 - c. Results of all actions taken to correct deficiencies occurring in plant equipment, structures, systems, or method of operation that affect nuclear safety at least semiannually.
 - d. Performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once every two years.
 - e. Deleted.
 - f. The Plant Fire Protection Program, implementing procedures and the independent fire protection and loss prevention program at least once every 24 months.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

- a. Records and logs of plant operation, including power levels and periods of operation at each power level.
- b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment pertaining to nuclear safety.
- c. Reports of all Reportable Events.
- d. Records of periodic checks, inspections, and calibrations required by these Technical Specifications.
- e. Records of nuclear safety related tests or experiments.
- f. Records of radioactive shipments.
- g. Records of changes to operating procedures.
- h. Records of sealed source leak tests and results.
- i. Records of annual physical inventory of all source material of record.
- j. Records of Quality Assurance activities required by the Operational Quality Assurance Program (OQAP) except where it is determined that the records should be maintained for a longer period of time.

6.10.2 The following records shall be retained for the duration of the Plant Operating License.

- a. Records of a complete set of as-built drawings for the plant as originally licensed and all print changes showing modifications made to the plant.
- b. Records of new and spent fuel inventory, fuel transfers, and assembly burnup histories.
- c. Records of plant radiation and contamination surveys.

4. Pre-planned procedures and back-up instrumentation to be used if one or more monitoring instruments become inoperable.
5. Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

6.15 SECONDARY WATER CHEMISTRY

The licensee shall implement a secondary water chemistry monitoring program. The intent of this program will be to control corrosion thereby inhibiting steam generator tube degradation. The secondary water chemistry program shall act as a guide for the chemistry group in their routine as well as non-routine activities.

6.16 RADIOLOGICAL EFFLUENTS

Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. Process Control Program implementation.
- b. Offsite Dose Calculation Manual implementation.
- c. Quality Assurance Program for effluent and environmental monitoring.

6.17 PROCESS CONTROL PROGRAM (PCP)

6.17.1 The PCP shall be approved by the Commission prior to implementation.

6.17.2 Licensee initiated changes to the PCP:

- a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:
 1. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information;
 2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing cri-

DOSE - IODINE-131, IODINE-133 AND RADIONUCLIDES IN PARTICULATE FORM

SPECIFICATIONS

7.4.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,
- b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

- a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3.c, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENTS

8.4.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM once per 31 days.

TABLE 8.4 (CONTINUED)

- b The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.3.b.
- c The ratio of the sample flow rate to the sampled stream flow rate shall be known (based on sampler and ventilation system flow measuring devices or periodic flow estimates) for the time period covered by each dose or dose rate calculation made in accordance with Specifications 7.4.1, 7.4.2 and 7.4.3.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 71 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

INTRODUCTION

By letter dated August 1, 1986, Wisconsin Public Service Corporation, et al., (the licensees) submitted a proposed amendment to the Kewaunee Nuclear Power Plant Technical Specifications. The proposed amendment revises the Technical Specifications to correct typographical errors, deletes obsolete references and replaces them with current ones.

EVALUATIONS

The following changes proposed by the licensee are primarily of an administrative nature, and do not adversely affect the safety of the plant or the general public. The staff finds these changes to be acceptable.

Description of Change; Operating License, Section 2.C.2

The proposed change would delete the reference to Appendix B. Appendix B was replaced with guidelines from the State of Wisconsin by Approved Amendment 47. Therefore, this change will reconcile this section with Approved Amendment 47.

Evaluation

Since Appendix B had been deleted by Amendment 47, this change is merely administrative in nature and has no effect on safety.

Description of Proposed Change, Technical Specification Page TS v

The proposed change would delete the reference to TS 6.9.3.d from the table of contents. TS 6.9.3.d was deleted by approved Amendment 64. Therefore, this change will reconcile the section with Amendment 64.

Evaluation

The change is administrative in nature and has no effect on safety.

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Description of Proposed Change, Technical Specification Page TS viii

The proposed change would delete the references to TS 6-26 and TS 6-27 from this page. Figures TS-6.2-1 and 6.2-2 were moved to the figure section of the Technical Specifications by Approved Amendment 48.

Evaluation

The proposed change deletes an old reference and reconciles the list of figures with Amendment 48. This change is administrative in nature and has no effect on safety.

Description of Proposed Change, Technical Specification Page 2.1-2

There are two proposed changes on this page. The first, corrects a typographical error by inserting an equal sign (=) between FQ^N and 2.51. The second, replaces the reference to Figure TS-3.10-2 with a reference to Figure TS-3.10.3. Figure TS-3.10-2 was renumbered by Approved Amendment 9. Therefore, this change will reconcile this section with Approved Amendment 9.

Evaluation

The first change was verified as a typographical error and the second change reconciled the section with Amendment 9 to the Technical Specifications. These changes are administrative in nature and have no effect on safety.

Description of Proposed Change, TS 2.3.a.6.A

This interlock (known as P-7) describes the reactor trips which become effective above 10% of rated power, which also includes the turbine trip-reactor trip. The wording has been revised to reflect this.

Evaluation

A turbine trip initiates a reactor trip. On decreasing power, the turbine trip is automatically blocked by P-7 and on power, reinstated automatically by P-7. This change is purely administrative in nature in that it only clarifies the Specification and has no effect on safety.

Description of Proposed Change, Technical Specification Page TS 2.3-6
Reference 2

The proposed change would change Reference 2 to reflect the Updated Safety Analysis Report. This is accomplished by changing Reference 2 from "FSAR, page 14-3" to "USAR, page 14.1-5." In the original FSAR the material referenced by Reference 2 was found on page 14-3. However, in the updated USAR the information is on page 14.1-5.

Evaluation

This change is administrative in nature and has no effect on safety.

Description of Proposed Change, Technical Specification Page TS-3.2-2

The proposed change would remove the reference to Specification 3.3.a and replace it with a reference to Specification 3.3.b.1.A. TS 3.3.a was renumbered TS 3.3.b.1.A by Approved Amendment 63.

Evaluation

This change will reconcile this section with Amendment 63, is administrative in nature and has no effect on safety.

Description of Proposed Change, Technical Specification 3.3.d.2

The proposed change would correct a typographical error by removing the word "on" and inserting the word "or".

Evaluation

This change was verified as a typographical error. The change is administrative in nature and has no effect on safety.

Description of Proposed Change, Technical Specification 3.4.d, Page TS 3.4-1a

Item 3.4.d has been added to this page, requiring two of three turbine overspeed protection systems to be operable when power is greater than 50% of rated power. This item was formerly located on Table 3.5-2, Item 11.

Evaluation

This item was removed from the Reactor Protection and ESF Instrumentation Systems section that addressed instrumentation necessary to ensure reactor safety. Since the primary purpose of the Turbine Overspeed Protection System is the protection of the turbine, this item was placed in the Power Conversion portion of the Technical Specifications, a more appropriate section. Since the substantive requirements of the specification have not changed, but have been merely relocated to a more appropriate section of the Technical Specifications, this change is merely administrative in nature and has no effect on safety.

Description of Proposed Change, Page TS 3.4-3

A new section has been added to the basis discussing turbine overspeed protection. This is the same wording formerly located on page TS 3.5-6, which has been deleted.

Evaluation

Since the substance of the basis remains the same, but has been merely relocated to a different section, this change has no effect on safety.

Description of Proposed Change, Page TS 3.5-6

The discussion regarding Turbine Overspeed Protection has been deleted from this page and added to page TS-3.4-3.

Evaluation

Since the substance of this basis remains the same, but has been merely relocated to a different section, this change has no effect on safety.

Description of Proposed Change, Technical Specification Page TS 3.5-7

The proposed change removes references to Table TS 3.5-5 and replaces them with references to Table 3.5-6. Table TS 3.5-5 (page 2 of 2) was renumbered Table TS 3.5-6 by Approved Amendment 59. Therefore, this change will reconcile this section with Approved Amendment 59.

Reference 4 on this page has been deleted because it refers to the turbine overspeed protection. This section of the Basis has been moved to Section 3.4 of the Technical Specifications.

Evaluation

Since the changes are administrative in nature, they have no effect on plant safety.

Description of Proposed Change, Technical Specification 3.10.b.1.A.ii, Page TS 3.10-1

The proposed change would correct a typographical error by changing a " \geq " sign in the first equation to a ">" sign. This change will clarify that the second equation in TS 3.10.b.1.A.ii should be used when $P = .5$.

Evaluation

This change is consistent with the equations contained in Technical Specification 3.10.b.1.A.i and the Westinghouse Standard Specifications. Since the change is primarily administrative in nature and is for clarification purposes, this change has no effect on safety.

Description of Proposed Change, Table TS 3.5-1 (Page 2 of 2) Note 3

The proposed change deletes the reference in the note to TS 3.9.b and replaces it with current references. TS 3.9.b was replaced with TS 7.4.1 and the Offsite Dose Calculation Manual (ODCM) by Approved Amendment 64.

Evaulation

This change is administrative in nature and has no effect on safety.

Description of Proposed Change, Table TS 3.5-2

Item 2:

The double asterisks have been replaced with a (3), the single asterisks have been replaced with a (2), and the "Permissible Bypass Condition" for the low setting has been restated as "P-10." P-10 is defined on page 3 of the table.

Item 3:

"P-10" has been added in column 5 under permissible bypass conditions, in lieu of the previous wording.

Item 4:

"P-6" has been added in column 5 under permissible bypass conditions in lieu of the previous wording. "P-6" is defined on page 3 of the table.

Items 5 and 6

The single asterisks have been replaced with a (2).

Item 7

The single asterisk has been replaced by a (2), and "P-7" has been added as a permissible bypass condition, in accordance with original plant design and TS 2.3.a.6.A.

Item 9

"P-7" has been added as a permissible bypass condition in accordance with original plant design and TS 2.3.a.6.A.

Item 10

The parenthetical phrase "(\leq 50% full power)" has been deleted and "P-8" has been added to column 5, in accordance with original plant design and TS 2.3.a.6.B, for the low flow in one loop trip.

The parenthetical phrase "(10-50% full power)" has been deleted and "P-7" added in column 5, in accordance with original plant design and TS 2.3.a.6.A.

Item 11

The requirement for turbine overspeed protection has been deleted from this table and moved to page TS 3.4-1a.

Item 13

"P-7" has been added in column 5 in accordance with original plant design and TS 2.3.a.6.A.

Item 14

Reference (4) has been added to clarify that this is not a direct reactor trip, but a trip at the reactor coolant pump breakers.

Item 15

Item 15 has been deleted. It was titled "Control Rod Misalignment Monitor," and had two sub-items:

- a. Rod position deviation, and
- b. Quadrant power tilt monitor

The requirements of this item are identical to TS 3.10.i and 3.10.j, and therefore redundant.

Notes:

The notes for Table 3.5-2 have been collected and grouped on this single page. The definitions of P-6, P-7, P-8, and P-10 have been added. Note 4, a clarification of the underfrequency trip of the Reactor Coolant Pump Breakers, has been added. The note in regard to turbine overspeed protection has been deleted from this page and incorporated into the verbiage on page TS 3.4-1a.

Evaluation

The changes to this table are primarily for clarification and reflect original plant design and current plant practice. They have been reviewed for their accuracy and adequacy separately and collectively for their effect on the intent of this Technical Specification Table. These changes are at a minimum consistent with and are generally more conservative than the existing Technical Specifications. They have no effect on safety.

Description of Proposed Change, Table TS 3.5-3

Item 1.C

The proposed change would add the phrase "Primary pressure < 2000 psig" to column 5, PERMISSIBLE BYPASS CONDITIONS. This phrase was inadvertently omitted when the Technical Specifications were first issued. The phrase clarifies the last paragraph on page TS 3.5-2 by defining under which conditions the safety injection signals can be blocked. The low steam pressure safety injection signal is blocked below a primary pressure of 2000 psig to prevent a safety injection when the plant is in a normal cool down to cold shutdown. This is a standard design feature for all Westinghouse plants and was part of the original licensing basis for the Kewaunee Plant. The inclusion of this statement would not remove or add any new requirements to the Technical Specifications. However, it will clarify the Specifications and reflect the operating parameters of the plant.

Item 1.d

The proposed change would add the phrase "Hot Shutdown***" to column 6, OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET. This phrase was inadvertently deleted when Approved Amendment 63 was transmitted to WPSC.

The phrase requires the Kewaunee Plant be placed in hot shutdown if fewer than two pressurizer pressure channels are operable. Presently, the literal interpretation of the Technical Specification would not require the Kewaunee Plant be placed in hot shutdown if two or more pressurizer pressure channels became inoperable. Presently, the Kewaunee Plant is being operated under the intent of the original specification.

Item 2

The proposed changes to columns 1, 2, 3, and 4 do not change the requirements concerning the selected boric acid storage tank (BAST) level. However, the changes would cause the Technical Specification to reflect the physical configuration of the plant more accurately.

The change to column 6 will clarify the operator actions if the conditions in columns 3 or 4 cannot be met.

Presently, column 6 appears to allow one channel of BAST level to be inoperable for 72 hours before entering the 72 hour limiting condition of operation (LCO) specified in TS 3.3.b.2.A. The proposed change would clarify the more conservative interpretation which would require that the 72-hour LCO specified in TS 3.3.b.2.A be entered immediately. This is the more conservative interpretation intended by Approved Amendment 63.

Evaluation

The changes to this table are primarily for clarification and to reflect the original plant design and current plant practice. They have been reviewed for their accuracy and adequacy separately and collectively for their effect on the intent of this Technical Specification Table. These changes are, at a minimum, consistent with and are generally more conservative than the existing Technical Specifications. They have no effect on safety.

Description of Proposed Change, Table TS 3.5-6

The proposed change would correct the misspelling of the word "Minimum" in the last footnote on this page.

Evaluation

This change is administrative in nature and has no effect on safety.

Description of Proposed Change, Table TS 4.1-2

The proposed change would restore two inadvertently deleted footnote references from Sampling Tests 2 and 3 in Table TS 4.1-2. The NRC previously approved these footnotes in our Amendment No. 63 issued July 5, 1985.

Evaluation

This change corrects two errors and has no effect on safety.

Description of Proposed Change, Technical Specifications 6.1.1.a and 6.1.1.b, Page TS 6-1

The proposed change would change the order of succession for overall on-site responsibility in the absence of the Plant Manager. The change is being initiated to reflect personnel changes at the Kewaunee Plant.

Evaluation

This change is administrative in nature and had no effect on safety.

Description of Proposed Change, Technical Specification 6.5.1.6.h, Page TS 6-4

The proposed change would correct the misspelling of the word "Committee."

Evaluation

This change is administrative and has no effect on safety.

Description of Proposed Change, Technical Specification 6.5.1.6.h, Page TS 6-5

The proposed change would delete the phrase "changes to the Vice-President-Power Production" from the top of Page TS 6-5. This TS was revised in our Amendment No. 63 issued July 5, 1985. The NRC should have removed this obsolete phrase from the page at that time.

Evaluation

This change corrects an error and has no effect on safety.

Description of Proposed Change, Technical Specification 6.5.3.8.e, Page TS 6-10

The proposed change would delete this obsolete Technical Specification.

Generic letter 82-17 requires all Licensees to meet the requirements of 10 CFR 50.54(t) regardless of existing Technical Specifications. 10 CFR 50.54(t) requires that emergency preparedness programs be reviewed at least every 12 months. However, TS 6.5.3.8.e requires an audit of the emergency plan every 24 months.

Generic letter 82-23 requires all Licensees to meet the requirements of 10 CFR 73.40(d) regardless of existing Technical Specifications. 10 CFR 73.40(d) requires all licensees to review their security plan at least every 12 months. TS 6.5.3.8.e requires an audit of the plan every 24 months.

Since all Licensees are required to abide by the rules as stated in Title 10 of the Code of Federal Regulations, the existing Technical Specification, TS 6.5.3.8.e, is obsolete. Therefore, it can be deleted without increasing the risk to the public's health and safety.

Evaluation

Since this change will reduce the possibility for confusion between the Technical Specification and CFR requirements and increase the review frequency of the emergency preparedness and security programs by deleting an obsolete Technical Specification the change has no effect on safety.

Description of Proposed Change, Technical Specification 6.10.1.i, Page TS 6-20

The proposed change would correct a typographical error by removing the word "of" and inserting the word "on."

Evaluation

This change was verified as a typographical error. The change is administrative in nature and has no effect on safety.

Description of Proposed Change, Technical Specifications 6.17.1 and 6.17.2, Page TS 6.23

The proposed change would correct two typographical errors by removing the Technical Specification numbers 6.16.1 and 6.16.2 and inserting Technical Specification numbers 6.17.1 and 6.17.2. These specifications were incorrectly numbered by Approved Amendment 64.

Evaluation

This change is administrative in nature and has no effect on safety.

Description of Proposed Change, Technical Specification 7.4.3, Page TS 7/8-9

The proposed change would correct a typographical error by removing the word "that" and inserting the word "than."

Evaluation

This change was verified as a typographical error. The change is administrative in nature and has no effect on safety.

Description of Proposed Change, Table TS 8.4 Item C.c

The proposed change would correct a typographical error by removing a reference to Technical Specification 7.41, which does not exist, and inserting a reference to Technical Specification 7.4.1.

Evaluation

This change was verified as a typographical error. The change is administrative in nature and has no effect on safety.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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.

ACKNOWLEDGEMENT

Principal Contributors:

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Dated: January 21, 1987

JAN 21 1987

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LHarmon
EJordan
BGrimes
JPartlow
EButcher, TSCB
TBarnhart (4)
WJones
FOB, DPLA
ACRS (10)
OPA
LFMB (TAC No. 62085)