December 8, 1994

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Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: AMENDMENT NOS. 157 AND 161 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 (TACS M85689 AND M85690)

Dear Mr. Link:

The Commission has issued the enclosed Amendment Nos. 157 and 161 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments revise the Technical Specifications in response to your application dated December 10, 1992, as supplemented on March 8, 1994.

These amendments revised Technical Specifications (TS) Section 15.3.5, "Instrumentation System," and Section 15.4.1, "Operational Safety Review." Specifically, extensive additions and modifications were made to various tables which specify requirements for the instrumentation and safety circuits necessary to ensure reactor safety and provide for the automatic initiation of the engineered safety features.

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

Sincerely,

Original signed by Allen G. Hansen

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

Enclosures:
1. Amendment No. 157 to DPR-24
2. Amendment No. 161 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

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*See previous concurrence

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

December 8, 1994

Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

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Allen G. Hansen, Project Manager
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and 50-301

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1. Amendment No. 157 to DPR-24
2. Amendment No. 161 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

Mr. Robert E. Link, Vice President
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated December 10, 1992, as supplemented on March 8, 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 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 157 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: December 8, 1994

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated December 10, 1992, as supplemented on March 8, 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 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 161 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: December 8, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 157 AND 161

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

DOCKET NOS. 50-266 AND 50-301

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 vertical lines indicating the area of change.

REMOVE	INSERT
TS 15.1-5	TS 15.1-5
TS 15.1-6	TS 15.1-6
TS 15.3.5-1	TS 15.3.5-1
Table 15.3.5-2 (2 pages)	Table 15.3.5-2 (3 pages)
Table 15.3.5-3 (2 pages)	Table 15.3.5-3 (2 pages)
Table 15.3.5-4 (1 page)	Table 15.3.5-4 (1 page)
Table 15.3.5-5 (2 pages)	Table 15.3.5-5 (4 pages)
Table 15.4.1-1 (5 pages)	Table 15.4.1-1 (6 pages)
Table 15.4.1-2 (4 pages)	Table 15.4.1-2 (4 pages)

l. Reactor Critical

The reactor is said to be critical when the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

m. Low Power Operation

The reactor is in the low-power operating condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates less than or equal to 2% of rated power.

n. Fire Suppression Water System

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source; pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard post indicating valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

o. <u>Dose Equivalent I-131</u>

Dose Equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

p. <u>$\bar{E}$ - Average Disintegration Energy</u>

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

15.3.5 INSTRUMENTATION SYSTEM

Operational Safety Instrumentation

Applicability: Applies to plant instrumentation systems.

Objectives: To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

A. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 15.3.5-1.

B. For on-line testing or in the event of a sub-system instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 15.3.5-2 through 15.3.5-4.

C. In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, operation shall be limited according to the requirement shown in Tables 15.3.5-2 through 15.3.5-4, Operator Action when minimum operable channels unavailable.

D. The post-accident monitoring instrumentation channels in Table 15.3.5-5 shall be operable. In the event the number of channels in a particular sub-system falls below the minimum number of operable channels given in Column 2, operation and subsequent operator action shall be in accordance with Column 3. This specification is not applicable in the cold or refueling shutdown conditions.

Basis: Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features(1).

TABLE 15.3.5-2
INSTRUMENT CONDITIONS FOR REACTOR TRIP

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
1.	Manual	2	1	1		Be in hot shutdown in 8 hours
2.	Nuclear Power Range					
a.	low setting	4	2	$3^{\#,**}$	2 of 4 power range channels greater than 10% full power (low setting only)	Be in hot shutdown in 8 hours
b.	high setting	4	2	$3^{\#,**}$		Be in hot shutdown in 8 hours
3.	Nuclear Flux Intermediate Range	2	1	1	2 of 4 power range channels greater than 10% full power	Be in hot shutdown in 8 hours*
4.	Nuclear Flux Source Range	2	1	1	1 of 2 intermediate range channels greater than 10^{-10} amps.	Be in hot shutdown in 8 hours*
5.	Overtemperature Delta T	4	2	3^{**}		Be in hot shutdown in 8 hours
6.	Overpower Delta T	4	2	3^{**}		Be in hot shutdown in 8 hours
7.	Low Pressurizer Pressure	4	2	3^{**}		Be in hot shutdown in 8 hours
8.	Hi Pressurizer Pressure	3	2	2^{**}		Be in hot shutdown in 8 hours

TABLE 15.3.5-2(continued)

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
9.	Hi Pressurizer Water Level	3	2	2^{**}		Be in hot shutdown in 8 hours
10.	Low Reactor Coolant System Flow					
a.	Low Flow in One Loop (>50% full power)	3/loop	2/loop (any loop)	$2/loop^{**}$		Be in hot shutdown in 8 hours
b.	Low Flow in Both Loops (10-50% full power)	3/loop	2/loop (both loops)	$2/loop^{**}$		Be in hot shutdown in 8 hours
11.	Turbine Trips					
a.	Turbine Autostop Oil Pressure	3	2	2^{**}		Be <50% of rated power within 4 hours
b.	Turbine Stop Valve Position	2	2	2^{**}		Be <50% of rated power within 4 hours
12.	Steam Flow-Feedwater Flow Mismatch	2/loop	1/loop	1/loop		Be in hot shutdown in 8 hours
13.	Lo Lo Steam Generator Water Level (input to reactor trip)	3/loop	2/loop	$2/loop^{**}$		Be in hot shutdown in 8 hours
14.	4KV Bus (A01 and A02)					
a.	Undervoltage (input to reactor trip)	2/each bus	1/each bus	1/each bus		Be in hot shutdown in 8 hours
b.	Underfrequency	2/each bus	1/each bus	1/each bus		Be in hot shutdown in 8 hours

TABLE 15.3.5-2(continued)

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
15.	Safety Injection	See Table 15.3.5-3	1	See Table 15.3.5-3		Be in hot shutdown in 8 hours***
16.	RCP Breaker Open Position					
a.	(>50% full power)	2	1	2		Be in hot shutdown in 8 hours
b.	(10 - 50% full power)	2	2	2		Be in hot shutdown in 8 hours
17.	Reactor Trip Breakers	2	1	2	****	Be in hot shutdown in 8 hours

One additional channel may be taken out of service for low power physics testing.

* When block condition exists, maintain normal operation.

** If a channel is determined to be inoperable, resulting in one less than the total number of channels being operable, power operation may continue if the following conditions are met:

1. The minimum number of operable channels is still satisfied.
2. The affected channel is placed in trip within 1 hour.

*** If minimum conditions are not met within 24 hours after reaching hot shutdown, the unit shall be in cold shutdown within 48 hours of the event causing the unit shutdown.

**** When at power, one channel may be bypassed for up to 8 hours provided that the other channel is operable. When the plant is shutdown and rod withdrawal is possible, restore the inoperable channel to operable status within 48 hours or open the Reactor Trip Breakers within 1 hour.

TABLE 15.3.5-3
ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
1.	SAFETY INJECTION					
a.	Manual	2	1	1		Be in hot shutdown in 8 hours*
b.	Hi Containment Pressure	3	2	2**		Be in hot shutdown in 8 hours*
c.	Steam Generator Low Steam Pressure/Loop	3	2	2**	Primary pressure <1800 psig	Be in hot shutdown in 8 hours*
d.	Low Pressurizer Pressure	3	2	2**	Primary pressure <1800 psig	Be in hot shutdown in 8 hours*
2.	CONTAINMENT SPRAY					
a.	Manual****	2	2	2		Be in hot shutdown in 8 hours*
b.	Hi-Hi Containment Pressure (Containment Spray)	2 sets of 3	2 of 3 in each set	2 per set**		Be in hot shutdown in 8 hours*
3.	AUXILIARY FEEDWATER					
a.	Start Motor-Driven Pumps					
i.	Low Low Steam Gen. Water Level	3/SG	2/either SG	2/SG**		Be in hot shutdown in 8 hours*
ii.	SI signal	SI Initiating Conditions as in Item 1				Be in hot shutdown in 8 hours*

TABLE 15.3.5-3 (continued)
ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
b.	Start Turbine-Driven Pump					
i.	Undervoltage on 4KV Buses (A01 & A02)	2/each bus	1/each bus	1/each bus		Be in hot shutdown in 8 hours*
ii.	Low Low Steam Gen. Water Level	3/SG	2/each SG	2/SG**		Be in hot shutdown in 8 hours*
4.	SAFETY-RELATED ELECTRICAL LOADS					
a.	4.16KV Buses (A05, A06)					
i.	Degraded Voltage	3/bus	2/bus	2/bus**		***
ii.	Loss of Voltage	2/bus	1/bus	1/bus		***
b.	480V Buses (B03, B04)					
i.	Loss of Voltage	3/bus	2/bus	2/bus**		Be in hot shutdown in 8 hours*

* If minimum conditions are not met within 24 hours after reaching hot shutdown, the unit shall be in cold shutdown within 48 hours of the event causing the unit shutdown.

** If a channel is determined to be inoperable, resulting in one less than the total number of channels being operable, power operation may continue if the following conditions are met:
1. The minimum number of operable channels is still satisfied.
2. The affected channel is placed in trip within 1 hour.

*** Declare the associated emergency diesel generator inoperable for the affected bus. The applicable Limiting Condition for Operation (LCO) shall be entered. Separate LCOs may be entered for the Degraded Voltage and Loss of Voltage functions.

**** Both switches must be activated simultaneously.

TABLE 15.3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTIONS IF CONDITIONS OF COLUMN 3 CANNOT BE MET
1.	CONTAINMENT ISOLATION					
a.	Safety Injection	See Items 1b, c, and d of Table 15.3.5-3				Be in hot shutdown in 8 hours*
b.	Manual	2	1	1		Be in hot shutdown in 8 hours
2.	STEAM LINE ISOLATION					
a.	Hi Hi Steam Flow with	2/loop	1/either loop	1/loop		Be in hot shutdown in 8 hours*
	Safety Injection	See Items 1b, c, and d of Table 15.3.5-3				Be in hot shutdown in 8 hours*
b.	Hi Steam Flow and	2/loop	1/either loop	1/loop		Be in hot shutdown in 8 hours*
	Low Tavg with	4	2	3**		Be in hot shutdown in 8 hours*
	Safety Injection	See Items 1b, c, and d of Table 15.3.5-3				Be in hot shutdown in 8 hours*
c.	Hi Containment Pressure	3	2	2**		Be in hot shutdown in 8 hours*
d.	Manual	1/loop	1/loop	1/loop		Be in hot shutdown in 8 hours
3.	FEEDWATER ISOLATION					
a.	Hi Steam Generator Water Level	3/SG	2/SG	2/SG**		Be in hot shutdown in 8 hours*
b.	Safety Injection	See Item 1 of Table 15.3.5-3				Be in hot shutdown in 8 hours*

* If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in cold shutdown conditions.

** If a channel is determined to be inoperable, resulting in one less than the total number of channels being operable, power operation may continue if the following conditions are met:

1. The minimum number of operable channels is still satisfied.
2. The affected channel is placed in trip within 1 hour.

TABLE 15.3.5-5
INSTRUMENT OPERATING CONDITIONS FOR POST ACCIDENT MONITORING INSTRUMENTATION

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM OPERABLE CHANNELS	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
1.	PORV Position Indicator	1/Valve	1/Valve	If the operability of the PORV position indicator cannot be restored within 48 hours, shut the associated PORV Block Valve.
2.	PORV Block Valve Position Indicator	1/Valve	1/Valve	If the operability of the PORV Block Valve Position Indicator cannot be restored within 48 hours, shut and verify the Block Valve shut by direct observation or declare the Block Valve inoperable.
3.	Safety Valve Position Indicator	2/Valve	1/Valve	If the operability of at least one of the Safety Valve Position Indicators cannot be restored within seven days, be in at least hot shutdown within the next 12 hours.
4.	Reactor Coolant System Subcooling	2	1	If operability of at least one subcooling monitor cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
5.	AFW Pump Discharge Flowrate	3	#	If the minimum number of AFW Pump Discharge Flowrate channels required to provide indication of AFW flow to both steam generators cannot be restored to an operable status within 48 hours, be in hot shutdown within the next twelve hours.

\# The minimum number of operable channels for AFW Pump Discharge Flowrate is the number of AFW Pump Discharge Flowrate channels, in conjunction with the number of operable AFW to Steam Generator Flowrate channels, required to provide indication of AFW flow to both steam generators.

NOTE: The channel requirements in this table refer only to that portion of the instrument channel required for post accident monitoring. The applicable channels are listed in FSAR Table 7.7-2.

TABLE 15.3.5-5 (continued)

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM OPERABLE CHANNELS	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
6.	AFW to Steam Generator Flowrate	2	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
7.	Containment High Range Radiation	3	2	If the operability cannot be restored within seven days after failure, prepare a special report to be submitted within thirty days in accordance with 15.6.9.2.D.
8.	Containment Sump Level (Sump A)	2	1	Operation may continue up to thirty days. If operability cannot be restored, be in hot shutdown within the next twelve hours.
9.	Containment Sump Level (Sump B)	2	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
10.	Containment Hydrogen Concentration	2*	1	If operability cannot be restored within 72 hours, be in hot shutdown within the next six hours.
11.	Reactor Vessel Wide Range Level	2	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
12.	Reactor Vessel Narrow Range Level	2	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.

* With only one hydrogen monitor operable, restore an inoperable monitor with an independent power supply to an OPERABLE status within 30 days or be in hot shutdown within 6 hours.

NOTE: The channel requirements in this table refer only to that portion of the instrument channel required for post accident monitoring. The applicable channels are listed in FSAR Table 7.7-2.

TABLE 15.3.5-5 (continued)

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM OPERABLE CHANNELS	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
13.	In-Core Thermocouples	39 installed per core	2/core quadrant	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
14.	Main Steam Line Radiation	1/steam line	1/steam line	If operability cannot be restored within seven days, prepare a special report to be submitted within thirty days in accordance with 15.6.9.2.E.
15.	Refueling Water Storage Tank Level	2	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
16.	RCS Wide Range Pressure	3	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
17.	RCS Narrow Range Pressure	4	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
18.	RCS Wide Range Hot Leg Temperature	2/loop	1/loop	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
19.	RCS Wide Range Cold Leg Temperature	2/loop	1/loop	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
20.	Pressurizer Level	4	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.

NOTE: The channel requirements in this table refer only to that portion of the instrument channel required for post accident monitoring. The applicable channels are listed in FSAR Table 7.7-2.

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM OPERABLE CHANNELS	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
21.	Containment Wide Range Pressure	2	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
22.	Containment Intermediate Range Pressure	3	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
23.	Containment Low Range Pressure	3	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
24.	Condensate Storage Tank Level	2/tank	1/tank	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
25.	Steam Generator Wide Range Level	2/SG	1/SG	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
26.	Steam Generator Narrow Range Level	3/SG	1/SG	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
27.	Steam Generator Pressure	3/SG	1/SG	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
28.	Containment Isolation Position Indication	1	1	If the operability of the shut position indication of a Valve containment isolation valve cannot be restored within seven days, close the valve or be in hot shutdown within the next twelve hours.

NOTE: The channel requirements in this table refer only to that portion of the instrument channel required for post accident monitoring. The applicable channels are listed in FSAR Table 7.7-2.

TABLE 15.4.1-1
MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTS OF INSTRUMENT CHANNELS

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
1.	Nuclear Power Range	-	R	-	ALL
	-Heat Balance	S(1)	D(1,19)	-	ALL
	-Signal to delta T; bistable actions(rod stops, trips)	-	-	Q(1,2)	ALL
	-Compare results of the incore detector measurements to NIS axial flux difference	M(4,5,20)	-	-	PWR
2.	Nuclear Intermediate Range	-	R	-	ALL
	-when not blocked	S(1)	-	-	ALL
	-logarithmic level;bistable action (rod stop, trips)	-	-	P	ALL
3.	Nuclear Source Range	-	R	-	ALL
	-when not blocked	S	-	-	ALL
	-Bistable action (alarm and trips)	-	-	P	ALL
4.	Reactor Coolant Temperature	S	R	-	PWR,HOT S/D,COLD S/D
	-Overtemperature delta T	-	-	Q(1,2)	ALL
	-Overpower delta T	-	-	Q(1,2)	ALL
5.	Reactor Coolant Flow	S(1)	R	-	ALL
	-Analog and single loop loss-of-flow logic testing	-	-	Q(1,2)	ALL
	-Logic channel testing for reactor trip on loss of reactor coolant flow in both loops	-	-	R	ALL
6.	Pressurizer Water Level	S(1)	R	Q(1,2)	ALL
7.	Pressurizer Pressure	S(1)	R	Q(1,2)	ALL
8.	Steam Generator Level	S(1)	R	Q(1,17)	ALL

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
9.	Steam Generator Flow Mismatch	S(22)	R	Q(1)	ALL
10.	Steam Generator Pressure	S(16)	R	Q(1)	ALL
11.	4KV Bus Undervoltage (A01 & A02)				
	-AFW pump actuation	-	R	M(1)	ALL
	-Reactor Protection actuation	-	R	M(1,2)	ALL
12.	4KV Bus Underfrequency (A01 & A02)				
	-to Reactor Coolant Pump trip	-	R	-	ALL
13.	Safeguards Bus Voltage				
	-Loss of 4KV	S	R	M(1)	ALL
	-Degraded 4KV	S	R	M(1)	ALL
	-Loss of 480V	S	R	M(1)	ALL
14.	120 Vac Instrument Buses	W(6)	-	-	ALL
15.	Reactor Trip Signal From Turbine				
	-Turbine Autostop	-	-	M(1)	ALL
	-Turbine Stop Valve	-	-	M(1)	ALL
16.	Reactor Trip Signal From SI	-	-	M(1)	ALL
17.	Feedwater Isolation on SI				
	-MFP Trip on Safety Injection	-	-	R	ALL
	-MFRV Shutting on Safety Injection	-	-	R	ALL
18.	Accumulator Level and Pressure	S	R	-	ALL
19.	Analog Rod Position	S(8,22)	R	-	ALL
	-with step counters	S(22)	-	-	ALL
	-Monitoring by On-Line Computer	(18)	-	-	PWR,HOT S/D

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
20.	Auxiliary Feedwater Flowrate	(13)	R	-	ALL
21.	Boric Acid Control System	-	R	-	ALL
22.	Boric Acid Tank Level	D	R	-	ALL
23.	Charging Flow	-	R	-	ALL
24.	Condensate Storage Tank Level	S(1)	R	-	ALL
25.	Containment High Range Radiation	S(1)	R(14)	M(1)	ALL
26.	Containment Hydrogen Monitor -Gas Calibration -Electronic Calibration	D - -	- Q(15) R	- - -	ALL ALL ALL
27.	Containment Pressure	S	R	Q(1,3,9)	ALL
28.	Containment Water Level	M	R	-	ALL
29.	Emergency Plan Radiation Survey Instruments	Q	R	Q	ALL
30.	Environmental Monitors	M	-	-	ALL
31.	In-Core Thermocouples	M	R(14)	-	ALL
32.	Overpressure Mitigating System	S(12)	R	(10)	ALL
33.	PORV Block Valve Position Indicator	Q	R	-	ALL
34.	PORV Operability	-	R	Q(11)	ALL
35.	PORV Position Indicator	S(21)	R	R	ALL

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
36.	Radiation Monitoring System	D(7)	R(7)	M(7)	ALL
37.	Reactor Vessel Fluid Level System	M	R	-	ALL
38.	Refueling Water Storage Tank Level	-	R	-	ALL
39.	Residual Heat Removal Pump Flow	-	R	-	ALL
40.	Safety Valve Position Indicator	M	R	-	ALL
41.	Subcooling Margin Monitor	M	R	-	ALL
42.	Turbine Overspeed Trips				
	-Independent Overspeed Protection System	-	R	M(1)	ALL
	-Overspeed Block trip	-	R	M(1)	ALL
43.	Volume Control Tank Level	-	R	-	ALL
44.	Reactor Protection System and Emergency Safety Feature Actuation System Logic	-	-	M(1,23)	ALL
45.	Reactor Trip System Interlocks				
	-Intermediate Range Neutron Flux, P-6	-	R(24)	R	ALL
	-Power Range Neutron Flux, P-8	-	R(24)	R	ALL
	-Power Range Neutron Flux, P-9	-	R(24)	R	ALL
	-Power Range Neutron Flux, P-10	-	R(24)	R	ALL
	-1st Stage Turbine Impulse Pressure	-	R(24)	R	ALL

NOTATION USED IN TABLE 15.4.1-1

S- Each shift
D- Daily
W- Weekly
Q- Quarterly
M- Monthly
P- Prior to reactor criticality if not performed during the previous week.
R- Each refueling interval (but not to exceed 18 months)
PWR- Power and Low Power Operation, as defined in Specifications 15.1.h. and 15.1.n.
HOT S/D- Hot Shutdown, as defined in Specification 15.1.g.1.
COLD S/D- Cold Shutdown, as defined in Specification 15.1.g.2.
REF S/D- Refueling Shutdown, as defined in Specification 15.1.g.3.
ALL- All conditions of operation, as defined in Specifications 15.1.g, h and n.

NOTES USED IN TABLE 15.4.1-1

(1) Not required during periods of refueling shutdown, but must be performed prior to reactor criticality if it has not been performed during the previous surveillance period.

(2) Tests of the low power trip bistable setpoints which cannot be done during power operations shall be conducted prior to reactor criticality if not done in the previous surveillance interval.

(3) Perform test of the isolation valve signal.

(4) Perform by means of the moveable incore detector system.

(5) Recalibrate if the absolute difference is ≥ 3 percent.

(6) Verification of proper breaker alignment and that the 120 Vac instrument buses are energized.

(7) Radioactive Effluent Monitoring Instrumentation Surveillance Requirements are specified in Section 15.7.4.

(8) Verify that the associated rod insertion limit is not being violated at least once per 4 hours whenever the rod insertion limit alarm for a control bank is inoperable.

(9) Test of Narrow Range Pressure, 3.0 psig, -3.0 psig excluded.

(10) When used for the Overpressure Mitigating System, each PORV shall be demonstrated operable by:
   a. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

(11) Performance of a channel functional test is required, excluding valve operation.

(12) Shiftly check is required when the reactor coolant system is not open to the atmosphere and the reactor coolant system temperature is less than the minimum temperature for the in-service pressure test as specified in TS Figure 15.3.1-1.

(13) An AFW flow path to each steam generator shall be demonstrated operable, following each cold shutdown of greater than 30 days, prior to entering power operation by verifying AFW flow to each steam generator.

(14) Calibration is to be a verification of response to a source.

(15) Sample gas for calibration at 2% and 6%.

(16) A check of one pressure channel per steam generator is required whenever the steam generator could be pressurized.

(17) Includes test of logic for reactor trip on low-low level, automatic actuation logic for auxiliary feedwater pumps, and test of logic for feedwater isolation on high steam generator level.

(18) Rod positions must be logged at least once per hour, after a load change >10% or after >30 inches of control rod motion if the on-line computer is inoperable.

(19) The daily heat balance is a gain adjustment performed to match Nuclear Instrumentation System indicated power level with reactor thermal output.

(20) To confirm that hot channel factor limits are being satisfied, the requirements of TS 15.3.10.B must be met.

(21) Check required only when the overpressure mitigation system is in operation.

(22) Not required during period of cold and refueling shutdowns, but must be performed prior to reactor criticality if it has not been performed during the previous surveillance period.

(23) Each train tested at least every 62 days on a staggered basis.

(24) Neutron detectors excluded from calibration.

TABLE 15.4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Test	Frequency
1.	Reactor Coolant Samples	Gross Beta-gamma activity (excluding tritium)	5/week[7]
		Tritium activity	Monthly
		Radiochemical $\bar{E}$ Determination	Semiannually [2][10]
		Isotopic Analysis for Dose Equivalent I-131 Concentration	Every two weeks[1]
		Isotopic Analysis for Iodine including I-131, I-133, and I-135	a.) Once per 4 hours whenever the specific activity exceeds 1.0µCi/gram Dose Equivalent I-131 or 100/$\bar{E}$ µCi/gram.[6] b.) One sample between 2 and 6 hours following a thermal power change exceeding 15% of rated power in a one-hour period.
		Chloride Concentration	5/week[8]
		Diss. Oxygen Conc.	5/week[6]
		Fluoride Conc.	Weekly
2.	Reactor Coolant Boron	Boron Concentration	Twice/week
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly[6]
4.	Boric Acid Tanks	Boron Concentration	Twice/week
5.	Spray Additive Tank	NaOH Concentration	Monthly
6.	Accumulator	Boron Concentration	Monthly
7.	Spent Fuel Pit	a) Boron Concentration b) Water Level Verification	Monthly Weekly

TABLE 15.4.1-2 (Continued)

	Test	Frequency
8. Secondary Coolant	Gross Beta-gamma Activity or gamma isotopic analysis	Weekly (6)
	Iodine concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.2 μCi/cc (6)
9. Control Rods	a) Rod drop times of all full length rods (3)	Each refueling or after maintenance that could affect proper functioning (4)
	b) Rodworth measurement	Following each refueling shutdown prior to commencing power operation
10. Control Rod	Partial movement of all rods	Every 2 weeks (18)
11. Pressurizer Safety Valves	Set point	Every five years (11)
12. Main steam Safety Valves	Set Point	Every five years (11)
13. Containment Isolation Trip	Functioning	Each refueling shutdown
14. Refueling System Interlocks	Functioning	Each refueling shutdown
15. Service Water System	Functioning	Each refueling shutdown
16. Primary System Leakage	Evaluate	Monthly (6)
17. Diesel Fuel Supply	Fuel inventory	Daily
18. Turbine Stop and Governor Valves	Functioning	Annually (6)
19. Low Pressure Turbine Rotor Inspection (5)	Visual and magnetic particle or liquid penetrant	Every five years
20. Boric Acid System	Storage Tank Temperature	Daily
21. Boric Acid System	Visual observation of piping temperatures (all $\geq$ 145°F)	Daily

TABLE 15.4.1-2 (Continued)

		Test	Frequency
22.	Boric Acid Piping Heat Tracing	Electrical circuit operability	Monthly
23.	PORV Block Valves	a. Complete Valve Cycle b. Open position check	Quarterly [13] Every 72 hours [14]
24.	Integrity of Post Accident Recovery Systems Outside Containment	Evaluate	Each refueling cycle
25.	Containment Purge Supply and Exhaust Isolation Valves	Verify valves are locked closed	Monthly [9]
26.	Reactor Trip Breakers	a. Verify independent operability of automatic shunt and undervoltage trip functions.	Monthly [9]
		b. Verify independent operability of manual trip to shunt and undervoltage trip functions.	Each refueling shutdown
27.	Reactor Trip Bypass Breakers	a. Verify operability of the undervoltage trip function.	Prior to breaker use
		b. Verify operability of the shunt trip functions.	Each refueling shutdown
		c. Verify operability of the manual trip to undervoltage trip functions.	Each refueling shutdown
28.	120 VAC Vital Instr. Bus Power	Verify Energized [12]	Shiftly
29.	Power Operated Relief Valves (PORVs), PORV Solenoid Air Control Valves, and Air System Check	Operate [16]	Each shutdown [15]
30.	Atmospheric Steam Dumps	Complete valve cycle	Quarterly
31.	Crossover Steam Dump System	Verify operability of each steam dump valve.	Quarterly

TABLE 15.4.1-2 (Continued)

32.	Pressurizer Heaters	Verify that 100KW of heaters are available.	Quarterly
33.	CVCS Charging Pumps	Verify operability of pumps.(17)	Quarterly
34.	Potential Dilution in Progress Alarm	Verify operability of alarm.	Prior to placing plant in cold shutdown.

(1) Required only during periods of power operation.
(2) $\bar{E}$ determination will be started when the gross activity analysis of a filtered sample indicates $\geq 10\mu Ci/cc$ and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than $10\mu Ci/cc$.
(3) Drop test shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot condition, but cold drop tests need not be timed.
(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.
(5) As accessible without disassembly of rotor.
(6) Not required during periods of refueling shutdown.
(7) At least once per week during periods of refueling shutdown.
(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown.
(9) Not required during periods of cold or refueling shutdown, but must be performed prior to exceeding 200°F if it has not been performed during the previous surveillance period.
(10) Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last subcritical for 48 hours or longer.
(11) An approximately equal number of valves shall be tested each refueling outage such that all valves will be tested within a five year period. If any valve fails its tests, an additional number of valves equal to the number originally tested shall be tested. If any of the additional tested valves fail, all remaining valves shall be tested.
(12) The specified buses shall be determined energized in the required manner at least once per shift by verifying correct static transfer switch alignment and indicated voltage on the buses.
(13) Not required if the block valve is shut to isolate a PORV that is inoperable for reasons other than excessive seat leakage.
(14) Only applicable when the overpressure mitigation system is in service.
(15) Required to be performed only if conditions will be established, as defined in Specification 15.3.15, where the PORVs are used for low temperature overpressure protection. The test must be performed prior to establishing these conditions.
(16) Test valve operation in accordance with the inservice test requirements of the ASME Boiler and Pressure Vessel Code, Section XI.
(17) Operability of charging pumps is verified by ensuring that the pumps develop the required flowrate, as specified by the In-Service Test program.
(18) Not required to be performed if the reactor is subcritical.

Unit 1 - Amendment No. ~~106~~, ~~129~~, ~~148~~, ~~155~~, 157
Unit 2 - Amendment No. ~~109~~, ~~133~~, ~~152~~, ~~159~~, 161

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO LICENSEE RESPONSE TO GENERIC LETTER 89-19

AND PROPOSED TECHNICAL SPECIFICATION UPGRADES

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

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 BACKGROUND

By letter dated December 10, 1992, as supplemented on March 8, 1994, Wisconsin Electric Power Company (WEPCo), the licensee for the Point Beach Nuclear Plant (PBNP), submitted Technical Specification (TS) Change Request (CR) 154 entitled "Modifications to Technical Specifications Sections 15.3.5 and 15.4.1." This submittal was the first of three change requests to be submitted in partial fulfillment of the commitment made by WEPCo in their letter dated December 3, 1991, to upgrade their TSs.

As part of this submittal, TS changes are proposed to complete WEPCo's response to Generic Letter (GL) 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implications of Control Systems in LWR [Light Water Reactor] Nuclear Power Plants' Pursuant to 10 CFR 50.54(f)." This generic letter recommended that licensees provide automatic steam generator (SG) overfill protection together with appropriate plant procedures and TSs.

The staff evaluation of this change request is presented in two separate sections. In the first section, the licensee's response to GL 89-19 is addressed, including the associated TS upgrades. In the second section, all remaining TS upgrades are addressed.

2.0 SG OVERFILL PROTECTION (GL 89-19)

2.1 Background

By GL 89-19, the NRC recommended that a system be provided to mitigate main feedwater overfill events for all boiling and pressurized LWRs that currently do not have such protection. This action was part of the technical resolution of Unresolved Safety Issue (USI) A-47, "Safety Implications of Control Systems in LWR Nuclear Power Plants." Furthermore, it was requested that all LWR plants have TSs that address the operability of the overfill protection systems that are provided in response to the GL.

WEPCo responded to GL 89-19 by letters dated March 20, 1990, July 31, 1990, and December 10, 1992. In its March 20, 1990, submittal, WEPCo described the SG overfill protection system for PBNP, Units 1 and 2. In its July 31, 1990, submittal, WEPCo applied for a TS change which would add a monthly test of the logic for high SG water level (part of the SG overfill protection system, which isolates main feedwater) at PBNP. The December 10, 1992, letter, proposed an additional LCO (Limiting Condition for Operation) for the high SG water level instrument operating conditions, and also served as a resubmittal, for completeness, of the July 31, 1990, letter. The December 1992 letter, proposes a total of 34 TS changes intended to improve the quality of the Point Beach TSs and to bring them closer to the newer standard TSs.

In a letter dated January 8, 1993, WEPCo requested that Technical Specification Change Request (TSCR) 138, dated March 30, 1990, and Technical Specification Change Request (TSCR) 140, dated July 31, 1990, be withdrawn from the docket. The requested changes have been included in TSCR 154.

2.2 Evaluation

Overfill protection for each LWR consists of protection channels that initiate the termination of main feedwater flow to the reactor vessel for a BWR or to the SGs for a PWR, on sensing a high water level condition. The overfill protection mitigates the consequences of main feedwater control system failures as an event which could lead to overfill conditions, as well as limiting the operating water level to within the bounds of the assumptions used in the safety analysis.

GL 89-19 requests that license amendments be proposed to provide TSs for overfill protection, including requirements for LCOs, setpoints, and surveillance requirements, which are commensurate with the safety actions required by the existing plant TSs.

Per GL 89-19, an acceptable overfill protection system design is one which (a) is separate from the feedwater control system, so that it is not powered from the same source, (b) is not located in the same cabinet as the feedwater control system, and (c) the cables are not routed so that a fire is likely to affect both the feedwater control system and the overfill protection system simultaneously. However, common-mode failures that could disable overfill protection and the feedwater control system, but would still cause a feedwater pump trip, are considered acceptable failure modes.

Enclosure 2 to GL 89-19 identifies three groups of Westinghouse-designed plants to be considered. PBNP is consistent with Group I, with the following exception. PBNP uses a two-out-of-three hi-hi SG water level initiating logic, which is safety grade, and uses one out of the three channels for both level control and overfill protection. The system isolates main feedwater (MFW) by closing the MFW control valves. This arrangement is the same as that described for the MFW isolation of the Group I Westinghouse plants, except that the MFW pump is not tripped at PBNP. Instead, a recirculation line allows limited flow back to the main condensers for pump protection.

The SG level instrument loop components used for SG overfill protection are safety-grade and addressed in the TSs, as they are also used for the low-low steam generator level reactor trip actuation. The MFW control valves are closed on a 2/3 hi-hi steam generator water level signal by venting the valve air actuators through two parallel solenoid valves. One of these solenoid valves also isolates air to the actuators. Either of these solenoid valves is able to vent the actuator pressure within sufficient time to protect from an overfill incident. Both of these solenoid valves are environmentally qualified.

The overfill protection system at Point Beach Nuclear Plant uses three water level channels per SG, each channel being powered from a different instrument bus. Each of the instrument buses is connected to its own battery and DC bus through an inverter. Each of the channels is located in a different instrument cabinet. One of the three channels also supplies the MFW control system. The 2/3 logic uses energized relays in series which are fed from different instrument buses. The loss of power will de-energize the relay and close the MFW control valves.

The overfill protection system solenoid valves have Battery "D05" as their power supply, and are fed through a series of DC distribution panels. If the power supply is lost, the solenoid valves will operate to prevent overfilling the steam generators. DC power is required to keep the solenoid valves in their proper position for normal operation. Air is required to keep the MFW control valve open. Loss of DC power will de-energize the solenoid valves, isolating air to the MFW control valve operator and releasing the air in the operator, closing the valve. The MFW control system for the "A" SG is powered from the 120 volt AC, "Y01," "red" instrument bus, which is powered from the battery or the "D05" battery charger. The MFW control system for the "B" SG is powered from the yellow channel inverter. The yellow channel inverter uses the "D106" battery or the "D106" battery charger as its source of power. Failure of the inverters or failure of the associated DC buses will cause the MFW control valves to close, preventing overfill of the SGs.

The logic for the overfill protection system is functionally tested monthly, SG level transmitters are calibrated each refueling, and feedwater isolation valves are tested each refueling. TSs for calibration and functional testing of the overfill protection system, as well as an LCO regarding instrument operating conditions for high SG level, are proposed as part of TSCR 154.

The WEPCo response does not specifically meet one criterion of GL 89-19 because some fire locations could disable both the automatic overfill protection system and the normal MFW control system. WEPCo indicated that the loss of SG level indication for fires in these areas was identified in the Point Beach Appendix R review. As a result, alternate wide-range steam generator level indication is available outside the control room, independent of the fire zones of concern for the "B" steam generator on each unit. There is no such alternate wide range level indication for the "A" steam generator available at either unit. This would necessitate operator action to ensure a feedwater isolation in the event of a fire-related loss of steam generator level instrumentation.

TSCR 154 states that "the feedwater system will be automatically isolated by tripping the associated main feed pumps and shutting the associated main feedwater regulating valves." The required operator actions will be identical to the actions already existing for the remaining items in Table 15.3.5-4 (INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS). The operator actions require the plant to be placed in hot shutdown within 8 hours, should the minimum number of operable channels not be satisfied.

The existing PBNP TSs for systems that initiate safety actions define requirements which the NRC has previously reviewed and found to be in conformance with the applicable regulatory requirements for TSs, namely those set forth in 10 CFR 50.36 with regard to limiting conditions for operation, limiting safety system settings, and surveillance requirements. The licensee has proposed TSs for the overfill protection system which are equivalent to the existing TSs, and which ensure operability of the system at appropriate times.

2.3 Conclusions Regarding Response to GL 89-19

The licensee's overfill protection system design is acceptable because it is separate from the feedwater control system, because it is not located in the same cabinets as the feedwater control system, and because emergency procedures specify the operator action to ensure the feedwater isolation for fires which could affect both the feedwater control system and the overfill protection system simultaneously. In addition, the proposed TSs for the overfill protection system are acceptable because they ensure operability of the system and are consistent with existing requirements for systems providing a commensurate level of safety. Therefore, the staff concludes that the licensee's response to GL 89-19 is acceptable.

3.0 ADDITIONAL TECHNICAL SPECIFICATION UPGRADES

3.1 Background

TSCR 154 includes 34 distinctly noted proposed TS changes and several miscellaneous changes. These changes have been grouped into five categories: (1) Additions to upgrade the TSs; (2) Modifications to better align with Standard TSs; (3) Modifications which are consistent with current staff positions; (4) Editorial changes; and (5) Miscellaneous changes (no WEPCo change number). Each category is discussed separately below.

3.2 Additions to Upgrade Technical Specifications

Eighteen of the 34 proposed changes are additions to upgrade the TSs. Each is described below. The corresponding WEPCo change number is provided.

Conditions for Reactor Trip (WEPCo Change No. 3)

Additions are proposed for TS Table 15.3.5-2, "Instrument Conditions for Reactor Trip." Proposed Item 11, Turbine Trips, includes the turbine autostop oil pressure and the turbine stop valve position, and states that the operator has 4 hours to achieve <50% of rated power. Proposed Item 15, Safety Injection, is being added. Proposed Item 16, RCP (reactor coolant pump) Breaker Open Position, is being added.

Additions are also proposed for TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." Proposed Item 15, Reactor Trip Signal From Turbine, and Proposed Item 16, Reactor Trip Signal From SI, are being added.

Reactor Trip Breaker Conditions (WEPCo Change No. 4)

Additions are proposed for TS Table 15.3.5-2, "Instrument Conditions for Reactor Trip." Proposed Item 17, Reactor Trip Breakers, is being added together with a note explaining permissible bypass conditions. The note states that "one channel may be bypassed for up to 8 hours provided the other channel is operable." This 8-hour period is a new requirement which is consistent with current plant practice, though it is less conservative than the current staff position. However, the staff is accepting this 8-hour period on a plant-specific basis because it adds conservatism to the TSs.

Feedwater Isolation (WEPCo Change No. 7)

Additions are proposed for TS Table 15.3.5-4, "Instrument Operating Conditions for Isolation Functions." Proposed Item 3a, Hi Steam Generator Water Level (part of resolution of GL 89-19), and proposed Item 3b, Safety Injection, are being added.

An addition is also proposed for TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." Proposed Item 8, Steam Generator Level, will now include a logic test as part of the resolution of GL 89-19.

Post-Accident Monitoring (PAM) Instrumentation (WEPCo Change No. 11)

Additions are proposed to TS Table 15.3.5-5, "Instrument Operating Conditions for Indications." Proposed Item 15, Refueling Water Storage Tank Level, Item 16, RCS Wide Range Pressure, Item 17, RCS Narrow Range Pressure, Item 18, RCS Wide Range Hot Leg Temperature, Item 19, RCS Wide Range Cold Leg Temperature, Item 20, Pressurizer Level, Item 22, Containment Intermediate Range Pressure, Item 23, Containment Low Range Pressure, Item 24, Condensate Storage Tank Level, Item 25, Steam Generator Wide Range Level, Item 26, Steam Generator Narrow Range Level, Item 27, Steam Generator Pressure, and Item 28, Containment Isolation Valve Position Indication, are being added. Item 28, Containment Isolation Valve Position Indication, follows the format of Item 3 in Table 15.3.5-5. The main difference is if the operability (of the shut position indication) cannot be restored in seven days, for item 28, the valve must be closed, or the plant must be in hot shutdown within twelve hours.

In addition, Item 5, Auxiliary Feedwater Flow Rate, and Item 21, Containment High Range Pressure, are being restructured for clarification.

Nuclear Instrumentation Calibration (WEPCo Change No. 14)

Additions are proposed to TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." Calibration requirements for the power range (Item 1), intermediate range (Item 2), and source range (Item 3) are being added for the nuclear instrumentation.

Feedwater Isolation on SI (WEPCo Change No. 15)

An addition is proposed to TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." Surveillance requirements are being added as Item 17, Feedwater Isolation on SI, to ensure that the feedwater system can be isolated following a safety injection.

Condensate Storage Tank Level (WEPCo Change No. 17)

An addition is proposed to TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." Item 24, Condensate Storage Tank Level, is being added to provide surveillance requirements.

120 Vac Instrument Buses (WEPCo Change No. 18)

An addition is proposed to TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." Item 14, 120 Vac Instrument Buses, is being added to require channel checks.

Analog Rod Position (WEPCo Change No. 19)

An addition is proposed to TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." A note is being added to Item 19, Analog Rod Position, to ensure that rod insertion limits are not being violated.

Overpressure Mitigating System (WEPCo Change No. 25)

An addition is proposed to TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." A note is being added to Item 32, Overpressure Mitigating System, to clarify that requirements are only applicable when the system is required to be in operation.

Auxiliary Feedwater Flowrate (WEPCo Change No. 26)

A modification is proposed to TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels." The requirement for checking the flowrate at startup and shutdown would be changed to checking prior to entering power operation after cold shutdowns greater than 30 days.

Spent Fuel Pit (WEPCo Change No. 27)

An addition is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." Water level verification is being added to Item 7, Spent Fuel Pit.

Control Rods (WEPCo Change No. 28)

An addition is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." A requirement to perform a measurement of control rod worth is being added to Item 9, Control Rods.

Atmospheric Steam Dumps (WEPCo Change No. 29)

An addition is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." A requirement to cycle each atmospheric steam dump is being added as Item 28, Atmospheric Steam Dumps.

Crossover Steam Dump System (WEPCo No. 30)

An addition is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." A requirement to verify the operability of each crossover steam dump valve is being added as Item 29, Crossover Steam Dump System.

Pressurizer Heaters (WEPCo Change No. 31)

An addition is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." Item 30, Pressurizer Heaters, is being added to require verification of availability of adequate pressurizer heaters.

CVCS Charging Pumps (WEPCo Change No. 32)

An addition is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." Item 31, CVCS Charging Pumps, is being added to require verification of the operability of the charging pumps.

Dilution in Progress Alarm (WEPCo Change No. 33)

An addition is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." Item 32, Potential Dilution in Progress Alarm, is being added to require verification of alarm operability prior to placing the plant in cold shutdown. Verification of the alarm will be performed according to plant procedures.

Evaluation and Conclusion Regarding Additions to Upgrade Technical Specifications

All of the additions and modifications described in Section 3.2 above, enhance the PBNP TSs by adding test and evaluation requirements, and by clarifying applicability. They are also conservative and consistent with the TSs already approved by the staff at Point Beach. Therefore, these proposed TS changes are acceptable.

3.3 Modifications to Better Align With Standard Technical Specifications

Two of the 34 changes proposed by the licensee modify the TSs to more closely conform to Standard TSs (NUREG-1431, "Standard Technical Specifications, Westinghouse Plants"). Each is described below.

Minimum Degree of Redundancy (WEPCo Change No. 1)

Changes are proposed to TS Table 15.3.5-2, "Instrument Operation Conditions for Reactor Trip," TS Table 15.3.5-3, "Emergency Cooling," and TS Table 15.3.5-4, "Instrument Operating Conditions for Isolation Functions." The changes remove Column 4, Minimum Degree of Redundancy, from the tables, together with the definition and references to this in TS 15.1.C.1 and TS 15.3.5.C. An end note is being added to all tables to ensure that the minimum redundancy is maintained.

Anticipated Transient Without Scram Mitigating System Actuation Circuitry (AMSAC) (WEPCo Change No. 6)

A change is proposed to TS Table 15.3.5-3, "Emergency Cooling." Item 3.b, Trip of Both Main Feedpumps Starts Motor Driven Pumps, is being deleted as it does not provide a safety function.

Evaluation and Conclusion Regarding Modifications to Better Align With Standard Technical Specifications

The two proposed changes described above, clarify the intent of the TSs and/or delete unnecessary portions of the TSs, and are consistent with current staff positions. In addition, the changes more closely align with the Standard TSs. Therefore, they are acceptable.

3.4 Modifications Which are Consistent With Current Staff Positions

Two of the 34 changes proposed by WEPCo modify the TSs to clarify statements and remove inconsistencies, consistent with current staff positions. Each is described below.

Operator Actions (WEPCo Change No. 2)

Changes are proposed to TS Table 15.3.5-2, "Instrument Conditions for Reactor Trip," TS Table 15.3.5-3, "Emergency Cooling," and TS Table 15.3.5-4, "Instrument Operating Conditions for Isolation Functions." The changes provide clearer direction to the operator.

Control Rod (WEPCo Change No. 34)

A change is proposed to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." Note 13 is being added to Item 10, Control Rod, to state that the requirement is not applicable when the reactor is subcritical.

Evaluation and Conclusion Regarding Modifications Which are Consistent With Current Staff Positions

The two changes described above, clarify the applicability of the requirements and/or establish specific times for completion, enhancing the PBNP TSs. In addition, they are consistent with current staff positions. Therefore, they are acceptable.

3.5 Editorial Changes

Twelve of the 34 changes proposed by the licensee are editorial only, and do not change the meaning of the TSs. Each of these is discussed below.

Emergency Cooling (WEPCo Change No. 5)

The title of TS Table 15.3.5-3, "Emergency Cooling," is being changed.

Instrument Operating Conditions (WEPCo Change No. 8)

The title of TS Table 15.3.5-5, "Instrument Operating Conditions for Indications," is being changed.

Post-Accident Monitoring Instrumentation (WEPCo Change No. 9)

A note is being added to TS Table 15.3.5-5, "Instrument Operating Conditions for Indications," to provide additional guidance on its applicability.

Control Rod Position (WEPCo Change No. 10)

Item 6 of TS Table 15.3.5-5, "Instrument Operating Conditions for Indications," is being moved to Item 19 of TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," so that only instrumentation associated with post accident monitoring is in TS Table 15.3.5-5.

TS Table 15.3.5-5 Total Number of Channels (WEPCo Change No. 12)

Items 3 and 4 of TS Table 15.3.5-5, "Instrument Operating Conditions for Indications," are being updated to show that there are now two channels instead of one.

TS Table 15.4.1-1 Reformatting (WEPCo Change No. 13)

TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," is being reformatted for easier use.

Turbine Trip Functions (WEPCo Change No. 16)

Item 43, Turbine Overspeed Trips, in TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," is being reformatted and expanded for clarification.

Nuclear Instrumentation (WEPCo Change No. 20)

Items 2 and 3 of TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," are being modified to state that the channel check will only be performed when the instrumentation is not blocked.

Nuclear Power Range Instrumentation (WEPCo Change No. 21)

Item 1 of TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," is being revised for clarification of the surveillance requirements.

TS Table 15.4.1-1 Surveillance Interval (WEPCo Change No. 22)

Note P of TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," is being revised for clarification of a surveillance interval.

TS Table 15.4.1-1 Surveillance Requirements (WEPCo Change No. 23)

Note (1) of TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," is being revised for clarification of a surveillance requirement.

TS Table 15.4.1-1 Test Requirements (WEPCo Change No. 24)

Note (2) of TS Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels," is being revised for clarification of a test requirement.

Evaluation and Conclusion Regarding Editorial Changes

The changes described above, provide clarification of the TSs, and do not change the intent. They are strictly editorial. Therefore, they are acceptable.

3.6 Miscellaneous Changes

Several other minor changes were made that were not part of the list of 34 WEPCo numbered changes. In TS Section 15.3.5.D, the word "post" is added for clarification in the reference to the new title of Table 15.3.5-5, "Instrument Operating Conditions for Post-Accident Monitoring Instrumentation." In TS Table 15.3.5-3, "Emergency Cooling," the actual bus numbers are being referenced in Item 3.b and the word "Buses" is being changed to "Loads" in Item 4. Item 2.b of TS Table 15.3.5-4, "Instrument Operating Conditions for Isolation Functions," is being revised for clarification of the operating conditions for a steam line isolation. The titles to Items 8, 9, 10, and 14 of TS Table 15.3.5-5, "Instrument Operating Conditions for Indications," are being revised for clarification. Item 11 of TS Table 15.3.5-5 is being divided into two items resulting in a more conservative minimum number of operable channels. Item 13 of TS Table 15.3.5-5 is being changed to accurately reflect the total number of channels of in-core thermocouples. Footnote #9 of Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests," is being modified by adding the statement, "but must be performed prior to exceeding 200 °F if it has not been performed during the previous surveillance period." This change results in a more restrictive testing frequency for Items 25 and 26 of Table 15.4.1-2. The changes just described provide clarification of the TSs. In addition, they are consistent with current staff positions. Therefore, they are acceptable.

The supplemental letter on March 8, 1994, for the licensee's proposal modifies a proposed change. In the original letter, Item 3.b.i of TS Table 15.3.5-5 is proposed to include a double asterisk (WEPCo Change No. 1) in the Minimum Operable Channels column. The supplemental letter proposes to remove the double asterisk from both that item and the corresponding restrictions associated with the footnote. The change is consistent with the other items in the table and is equivalent to the existing specification. Therefore, the change is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change 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 surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (58 FR 16236). Accordingly, these amendments meet 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 these amendments.

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

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Date: December 8, 1994