DOCKET FILE

50-266

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

July 1, 1997

Mr. Richard R. Grigg
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS FOR TECHNICAL SPECIFICATION CHANGE REQUESTS 188 AND 189 (TAC NOS. M95682, M95683, M95697, AND M95698)

Dear Mr. Grigg:

The Commission has issued the enclosed Amendment Nos. 173 and 177 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your applications dated June 4, 1996 (two applications), as supplemented August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997.

These amendments revise Technical Specification (TS) 15.1, "Definitions;" TS 15.2.1, "Safety Limit, Reactor Core;" TS 15.2.3, "Limiting Safety System Settings, Protective Instrumentation;" TS 15.3.1, "Reactor Coolant System," Section C, "Maximum Coolant Activity," and Section G, "Operational Limitations;" TS 15.3.4, "Steam and Power Conversion System;" TS 15.3.5, "Instrumentation System;" TS 15.4.1, "Operational Safety Review;" TS 15.5.3, "Design Features-Reactor;" and TS 15.6.9, "Plant Reporting Requirements" for both units. These amendments authorize you to incorporate the above changes to the facility into the TS and the Final Safety Analysis Report (FSAR), as described in your applications dated June 4, 1996, as supplemented on August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997, and evaluated in the enclosed Safety Evaluation. The amendments approve operation of the units at either 2250 psia or 2000 psia primary system pressure and the applicable setpoints for either operating condition. We understand that you will decide at which primary system pressure you will operate at the start of each operating cycle or after an outage of sufficient duration to modify system setpoints to ensure consistency with the desired reactor coolant system operating pressure. The decision to operate at the lower pressure was evaluated based on your desire to minimize primary to secondary pressure differential, reduce load on pressurizer heaters, reduce charging pump discharge pressure, and lower stresses on the reactor coolant system. DFO1

In approving the proposed action, we have relied upon the following commitment: incorporation of changes resulting from the information submitted for replacement steam generators that resulted in changes to specifications, limiting conditions for operation, surveillance requirements, associated bases and accident analyses for steam generator tube rupture, main steam line break, reactor coolant pump locked rotor, and control rod ejection into the FSAR by June 30, 1998. The revised FSAR should include the assumptions, input parameters, and methodologies used in the analyses, results, and bases for acceptability of the results.

The staff will address the quality of the submittals associated with these amendments in future correspondence. A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY

Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

Enclosures:
1. Amendment No. 173 to DPR-24
2. Amendment No. 177 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

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See previous concurrence

DOCUMENT NAME: G:\WPDOCS\PTBEACH\PTB95682.AMD *See previous concurrence
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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. 173
License No. DPR-24

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

   A. The applications for amendment (TSCR 188 and 189) by Wisconsin Electric Power Company (the licensee) dated June 4, 1996, as supplemented August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 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. 173, 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 shall be implemented within 45 days from the date of issuance and the Final Safety Analysis Report changes shall be implemented by June 30, 1998. Implementation of this amendment includes incorporation of accident analyses submitted in support of this amendment into the Final Safety Analysis Report in sufficient detail to support future evaluations performed in accordance with 10 CFR 50.59 and as described in the licensee's applications dated June 4, 1996, as supplemented on August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997, and evaluated in the staff's safety evaluation dated July 1, 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: July 1, 1997

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
License No. DPR-27

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

   A. The applications for amendment (TSCR 188 and 189) by Wisconsin Electric Power Company (the licensee) dated June 4, 1996, as supplemented August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 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. 177, 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 shall be implemented within 45 days from the date of issuance and the Final Safety Analysis Report changes shall be implemented by June 30, 1998. Implementation of this amendment includes incorporation of accident analyses submitted in support of this amendment into the Final Safety Analysis Report in sufficient detail to support future evaluations performed in accordance with 10 CFR 50.59 and as described in the licensee's applications dated June 4, 1996, as supplemented on August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997, and evaluated in the staff's safety evaluation dated July 1, 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: July 1, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 173 AND 177

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
15.1-6	15.1-6
15.2.1-1	15.2.1-1
Figure 15.2.1-1	Figure 15.2.1-1
Figure 15.2.1-2	Figure 15.2.1-2
15.2.2-1	15.2.2-1
15.2.3-1	15.2.3-1
15.2.3-2	15.2.3-2
15.2.3-3	15.2.3-3
15.2.3-3a	15.2.3-3a
15.2.3-6	15.2.3-6
15.3.1-9	15.3.1-9
15.3.1-10	15.3.1-10
15.3.1-19	15.3.1-19
Figure 15.3.1-5	Figure 15.3.1-5
15.3.4-2	15.3.4-2
15.3.4-3	15.3.4-3
Table 15.3.5-1 (Page 1 of 2)	Table 15.3.5-1 (Page 1 of 2)
Table 15.4.1-2 (Page 1 of 5)	Table 15.4.1-2 (Page 1 of 5)
(Page 2 of 5)	(Page 2 of 5)
15.5.3-2A	15.5.3-3
15.6.9-3	15.6.9-3

o. Dose Equivalent I-131

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 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988.

p. $\bar{E}$ - Average Disintegration Energy

$\bar{E}$ 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.

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, and coolant temperature during operation.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1 for Units 1 and 2*. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

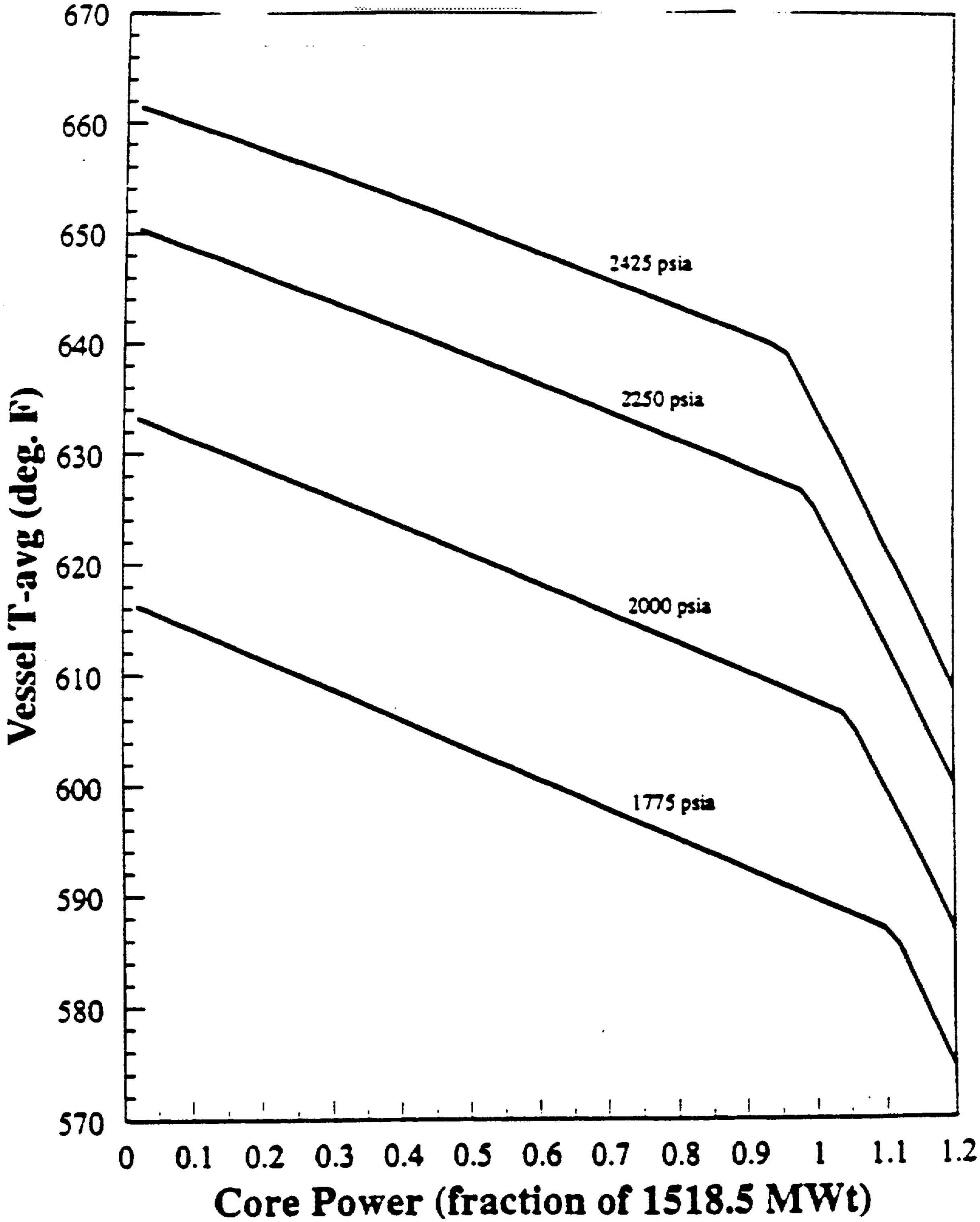
Basis:

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excess cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant temperature and pressure have been related to DNB.

* Figure 15.2.1-1 applies to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, Figure 15.2.1-2 applies to Unit 1.

Figure 15.2.1-1
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2*
REACTOR CORE SAFETY LIMITS



* This figure applies to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, Figure 15.2.1-2 applies to Unit 1.

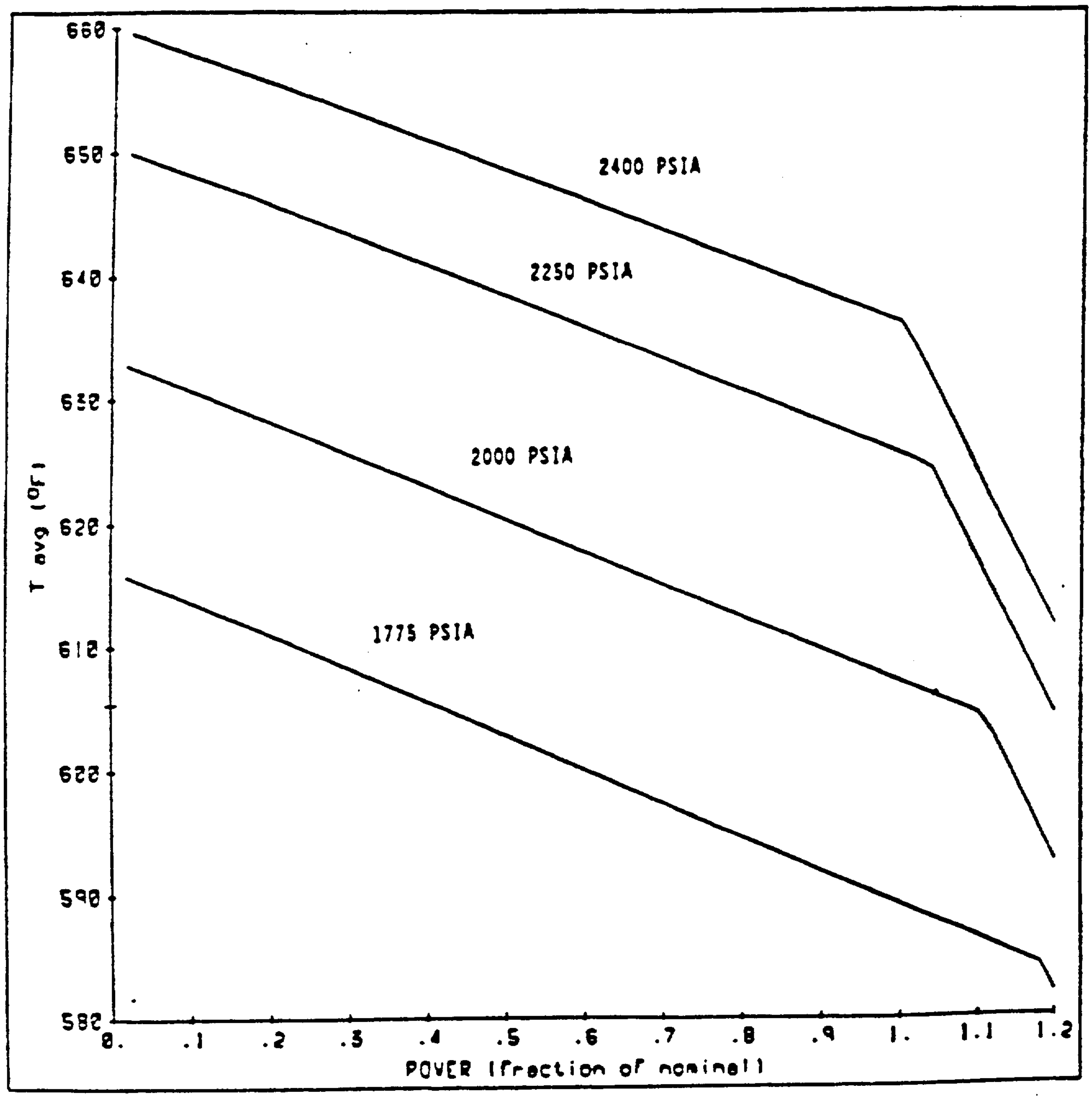
Unit 1 - Amendment No. ~~86~~, ~~120~~, ~~142~~, 173

Unit 2 - Amendment No. ~~146~~, ~~160~~, ~~169~~, 177

Figure 15.2.1-2*

REACTOR CORE SAFETY LIMITS

POINT BEACH UNIT 1



* This figure applies to Unit 1 prior to U1R24. Following U1R24, Figure 15.2.1-1 applies to Unit 1.

Unit 1 - Amendment No. ~~86~~, ~~120~~, ~~142~~, 173

Unit 2 - Amendment No. ~~146~~, ~~160~~, ~~169~~, 177

15.2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System Pressure.

Objective

To maintain the integrity of the Reactor Coolant System.

Specification

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

Basis

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.[2]

The nominal settings of the power-operated relief valves (2335 psig), the reactor high-pressure trip and the safety valves (2485 psig) have been established to assure never reaching the Reactor Coolant System pressure safety limit except for the hypothetical locked rotor and rod ejection accidents which use the faulted condition stress limit acceptance criterion of 3105 psig (3120 psia). The initial hydrostatic test was conducted at 3110 psig to assure the integrity of the Reactor Coolant System.

Reference

(1) FSAR, Section 4
(2) FSAR, Section 4.3

15.2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability:

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

Objective:

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification:

1. Protective instrumentation for reactor trip settings shall be as follows:
   A. Startup protection
      (1) High flux, source range - within span of source range instrumentation.

      (2) High flux, intermediate range - ≤40% of rated power.

      (3) High flux, power range (low setpoint) - ≤25% of rated power.

   B. Core limit protection
      (1) High flux, power range (high setpoint) - ≤108% of rated power

      (2) High pressurizer pressure* - ≤2385 psig for operation at 2250 psia primary system pressure
      ≤2210 psig for operation at 2000 psia primary system pressure.

* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the high pressurizer pressure reactor trip setpoint for Unit 1 is ≤2385 psig.

Unit 1 - Amendment No. ~~(Chg 4)~~, 173
Unit 2 - Amendment No. ~~3~~, 177

(3) Low pressurizer pressure* - ≥1905 psig for operation at 2250 psia primary system pressure

≥1800 psig for operation at 2000 psia primary system pressure

(4) Overtemperature $\Delta T \left(\frac{1}{1+\tau_3 S}\right)$

$$\leq \Delta T_o \left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3(P-P') - f(\Delta I)\right)$$

where (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)

ΔT_o = indicated ΔT at rated power, °F
T = average temperature, °F
T′ ≤ 572.9°F**
P = pressurizer pressure, psig
P′ = 2235 psig (2250 psia operation only)
P′ = 1985 psig (2000 psia operation only)**
K_1 ≤ 1.19 (2250 psia operation only)
K_1 ≤ 1.14 (2000 psia operation only)**
K_2 = 0.025 (2250 psia operation only)
K_2 = 0.022 (2000 psia operation only)**
K_3 = 0.0013 (2250 psia operation only)
K_3 = 0.001 (2000 psia operation only)**
τ_1 = 25 sec
τ_2 = 3 sec
τ_3 = 2 sec for Rosemont or equivalent RTD
= 0 sec for Sostman or equivalent RTD
τ_4 = 2 sec for Rosemont or equivalent RTD
= 0 sec for Sostman or equivalent RTD

and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within -17, +5 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the low pressurizer pressure reactor trip setpoint for Unit 1 is ≥ 1790 psig.

** These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the values are: T′ ≤ 573.9°F, P′ = 2235 psig, K_1 = ≤ 1.30, K_2 = 0.0200, and K_3 = 0.000791.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

(5) Overpower $\Delta T \left(\frac{1}{1+\tau_3 S}\right)$

$$\leq \Delta T_o \left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S+1}\right)\left(\frac{1}{1+\tau_4 S}\right)T - K_6\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right]\right]$$

where (values are applicable to operation at both 2000 psia and 2250 psia)

ΔT_o = indicated ΔT at rated power, °F
T = average temperature, °F
T' ≤ 572.9°F*
K_4 ≤ 1.09 of rated power*
K_5 = 0.0262 for increasing T
= 0.0 for decreasing T
K_6 = 0.00123 for $T \geq T'$
= 0.0 for $T < T'$
τ_5 = 10 sec
τ_3 = 2 sec for Rosemont or equivalent RTD
= 0 sec for Sostman or equivalent RTD
τ_4 = 2 sec for Rosemont or equivalent RTD
= 0 sec for Sostman or equivalent RTD

(6) Undervoltage - ≥75 percent of normal voltage

(7) Indicated reactor coolant flow per loop - ≥90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point ≥55.0 HZ

(b) Low voltage set point ≥ 75 percent of normal voltage.

* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the values for Unit 1 are: $T' \leq 573.9$°F and $K_4 \leq 1.09$ of rated power.

Other reactor trips:

(1) High pressurizer water level - ≤95% of span

(2) Low-low steam generator water level -
≥20% of narrow range instrument span
≥5% of narrow range instrument span (Unit 1)*

(3) Steam-Feedwater Flow Mismatch Trip - ≤1.0 x 10^6 lb/hr

(4) Turbine Trip (Not a protection circuit)

(5) Safety Injection Signal

(6) Manual Trip

* This setting limit applies to Unit 1 until the narrow range lower tap is changed to the lower position consistent with Unit 2.

power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figures 15.2.1-1 and 15.2.1-2. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced[6][7].

The overpower, overtemperature and pressurizer pressure system setpoints include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints will not exceed the core safety limits as shown in Figures 15.2.1-1 and 15.2.1-2.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident[4].

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis[8]. The low loop flow signal is caused by a condition of less than 90 percent flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening

C. MAXIMUM COOLANT ACTIVITY

Specification:

The specific activity of the reactor coolant shall be limit to:

1. Less than or equal to 0.8 microcurie per gram Dose Equivalent I-131.
   a. If the specific activity of the reactor coolant is greater than 0.8 microcuries per gram Dose Equivalent I-131 but within the allowable limit (below and to the left of the line) shown on Figure 15.3.1-5, operation may continue for up to 48 hours. Reactor coolant sampling shall be in accordance with Table 15.4.1-2.
   b. If the specific activity of the reactor coolant is greater than 0.8 microcuries per gram Dose Equivalent I-131 for more than 48 hours during one continuous time interval or exceeds the allowable limit (above and to the right of the line) shown on Figure 15.3.1-5, the reactor will be shut down and the average reactor coolant temperature will be less than 500 °F within 6 hours.
2. Less than or equal to $100/\bar{E}$ microcuries per gram.
   a. If the specific activity of the reactor coolant is greater than $100/\bar{E}$ microcuries per gram, the reactor will be shut down and the average reactor coolant temperature will be less than 500 °F within 6 hours. Reactor coolant sampling shall be in accordance with Table 15.4.1-2.

Basis:

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 500 gpd in either steam generator. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative for Point Beach Nuclear Plant.

Continued power operation for limited time periods with the reactor coolant's specific activity greater than 0.8 microcurie/gram Dose Equivalent I-131, but within the allowable limit shown on Figure 15.3.1-5, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power. Operation with specific activity levels exceeding 0.8 microcurie/gram Dose Equivalent I-131 but within the limits shown on Figure 15.3.1-5 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to less than 500 °F normally prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within the limits shown during Rated Power operation:

1. T_{avg} shall be maintained ≥557°F and ≤573.9°F.

2. Reactor Coolant System (RCS) pressurizer pressure shall be maintained:

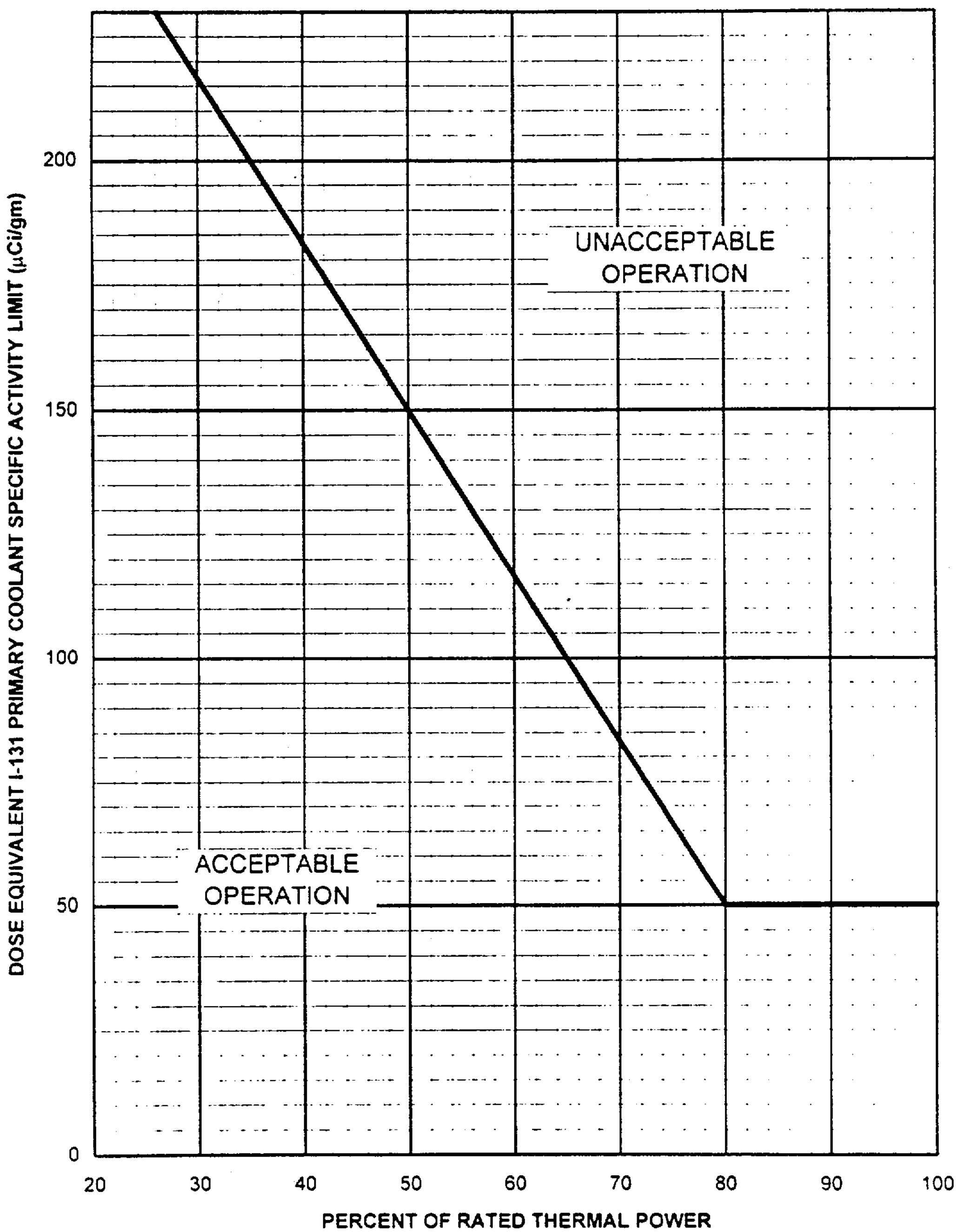
   ≥2205 psig during operation at 2250 psia, or
   ≥1955 psig during operation at 2000 psia.

3. Reactor Coolant System raw measured Total Flow Rate shall be maintained ≥181,800 gpm.

Basis:

The reactor coolant system total flow rate of 181,800 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimetric at the beginning of each cycle.

FIGURE 15.3.1-5



DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.8 μCi/gm Dose Equivalent I-131

3. A minimum of 13,000 gallons of water per operating unit in the condensate storage tanks and an unlimited water supply from the lake via either leg of the plant Service Water System.

4. System piping and valves required to function during accident conditions directly associated with the above components operable.

5. Both atmospheric steam dump lines shall be operable. If either of the atmospheric steam dump lines is determined to be inoperable, restore the inoperable line to an operable status within 24 hours. If operability cannot be restored, be in hot shutdown within six hours and cold shutdown within 24 hours.

B. The dose equivalent I-131 activity on the secondary side of the steam generator shall not exceed 1.0 μCi/g.

C. During power operation the requirements of 15.3.4.A.2.a and b may be modified to allow the following components to be inoperable for a specified time. If the system is not restored to meet the requirements of 15.3.4.A.2.a and b within the time period specified, the specified action must be taken. If the requirements of 15.3.4.A.2.a and b are not satisfied within an additional 48 hours, the appropriate reactor(s) shall be cooled down to less than 350 °F.

1. Two Unit Operation - One of the four operable auxiliary feedwater pumps may be out-of-service for the below specified times. A turbine driven auxiliary feedwater pump may be out of service for up to 72 hours. If the turbine driven auxiliary feedwater pump cannot be restored to service within the 72 hour time period the associated reactor shall be in hot shutdown within the next 12 hours. A motor driven auxiliary feedwater pump may be out of service for up to 7 days. If the inoperable motor driven auxiliary feedwater pump cannot be restored to service within the 7 day time period both of the reactors shall be in hot shutdown within the next 12 hours.

For the purposes of determining a maximum allowable secondary coolant activity, the steam break accident is based on a postulated release of the contents of one steam generator to the atmosphere using a site boundary dose limit. The limiting dose for this accident results from iodine in the secondary coolant. I-131 is the dominant isotope because of its low derived air concentration and because the other iodine isotopes have shorter half-lives and therefore cannot buildup to significant concentrations in the secondary coolant, given the limitations on primary system leak rate and activity. It is assumed that the accident occurs at zero load, which is when the maximum amount of water is contained in one steam generator. One tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate-out and retention in water droplets. It is conservative to measure gross beta-gamma activity except when the gross activity exceeds or equals 1.0 μCi/g. At this time the iodine-131 activity must be measured.

The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = \frac{C \times V}{10} \times B(t) \times \frac{\chi}{Q} \times DCF$$

where:

C = secondary coolant activity (1.0 μCi/g = 0.001 Ci/kg)

V = water mass in one steam generator (2877 $ft^3 \approx$ 62,250 kg)

B(t) = breathing rate (3.47×10^{-4} m^3/sec)

χ/Q = 5.0×10^{-4} sec/m^3 (4)

DCF = 1.07×10^6 rem/Ci I-131 inhaled

The resultant dose is approximately 1.2 rem.

References:
FSAR Section 10
FSAR Section 14

TABLE 15.3.5-1
(PAGE 1 OF 2)
ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety Injection*	≤6 psig
2	High Containment Pressure (Hi-Hi)	a. Containment Spray b. Steam Line Isolation of Both Lines	≤ 30 psig ≤ 20 psig
3.	Pressurizer Low Pressure	Safety Injection*	≥ 1715 psig
4	Low Steam Line Pressure	Safety Injection* Lead Time Constant Lag Time Constant	≥ 500 psig ≥ 12 seconds ≤ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{AVG}	Steam Line Isolation of Affected line	≤ d/p corresponding to 0.66×10^6 lb/hr at 1005 psig ≥ 540°F
6	High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	≤ d/p corresponding to 4×10^6 lb/hr at 806 psig
7	Low-low Steam Generator Water Level	Auxiliary Feedwater Initiation	≥ 20% of narrow range instrument ≥ 5% of narrow range instrument (Unit 1)**
8	Undervoltage on 4 KV Busses	Auxiliary Feedwater Initiation	≥ 75% of normal voltage

* Initiates also containment isolation, feedwater line isolation and starting of all containment fans.
** This setting limit applies to Unit 1 until the narrow range lower tap is changed to the lower position consistent with Unit 2.

d/p means differential pressure

Unit 1 - Amendment No. ~~114~~, 173
Unit 2 - Amendment No. ~~117~~, 177

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 0.8μCi/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 and after each BAST concentration change when they are being relied upon as a source of borated water.
5.	Spray Additive Tank	NaOH Concentration	Monthly
6.	Accumulator	Boron Concentration	Monthly

TABLE 15.4.1-2 (Continued)

	Test	Frequency
7. Spent Fuel Pit	a) Boron Concentration b) Water Level Verification	Monthly Weekly
8. Secondary Coolant	Gross Beta-gamma Activity or gamma isotopic analysis	Weekly(6)
	Iodine concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.0 μCi/g(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 5 years(11)
12. Main Steam Safety Valves	Set Point	Every 5 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 and piping temperatures ≥ temperature required by Table 15.3.2-1	Daily(19)

b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal Reactor Coolant System volume (both liquid and steam) at rated operating conditions and zero percent steam generator tube plugging is:

Unit 1 - 6500 ft^3

Unit 2 - 6643 ft^3

References

(1) FSAR Section 3.2.3

(2) Deleted

(3) Deleted

(4) FSAR Section 3.2.3

(5) Deleted

(6) FSAR Table 4.1-9

e. Reactor coolant activity

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 15.3.1.C. The following information shall be included:

1. Reactor power history starting 48 hours prior to the first sample in which the activity limit was exceeded;
2. Results of the last isotopic analysis for radioiodine analysis prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include the date and time of sampling and the radioiodine concentrations;
3. Clean-up flow history starting 48 hours prior to the first sample in which the activity limit was exceeded.
4. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and
5. The time duration when the specific activity of the primary coolant exceeded 0.8 microcuries per gram DOSE EQUIVALENT I-131.

C. Monthly Operating Reports

1. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis under the titles "Operating Data Report," "Average Daily Power Levels" and "Unit Shutdowns" and "Power Reduction". In addition, the report shall contain a narrative summary of operating experience that describes the operation of the facility, including major safety-related maintenance for the monthly report period.

2. Completed reports shall be sent by the tenth of each month following the calendar month covered by the report.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 173 AND 177 TO

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 INTRODUCTION

By applications dated June 4, 1996 (two), as supplemented August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997, the Wisconsin Electric Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant (PBNP), Unit Nos. 1 and 2. The proposed amendments, Change Request (CR) 188 and CR-189, would revise the TS to reflect new parameters associated with replacement steam generators in Unit 2 and changes in analyses that affect both Units 1 and 2. Additional information related to TS CR-192, included in the licensee's application dated September 30, 1996, as supplemented on November 26 and December 12, 1996, and January 16, March 20, April 2, April 16, May 9, June 3, and June 13 (two), 1997, was used to independently assess if the radiological consequences of the proposed TS changes related to the new steam generators remained less than the radiological consequences of a design-basis loss-of-coolant accident (LOCA). The proposed changes affect TS 15.1, "Definitions;" TS 15.2.1, "Safety Limit, Reactor Core;" TS 15.2.3, "Limiting Safety System Settings, Protective Instrumentation;" TS 15.3.1, "Reactor Coolant System," Section C, "Maximum Coolant Activity," and Section G, "Operational Limitations;" TS 15.3.4, "Steam and Power Conversion System;" TS 15.3.5, "Instrumentation System;" TS 15.4.1, "Operational Safety Review;" TS 15.5.3, "Design Features-Reactor;" and TS 15.6.9, "Plant Reporting Requirements" of both units and are listed below:

a. TS 15.2.1 (page 15.2.1-1): Revise references to Figures 15.2.1-1 and 15.2.1-2 and add footnote concerning the applicability of the figures to each unit.

   Figures 15.2.1-1 and 15.2.1-2: Combine Figures 15.2.1-1 and 15.2.1-2 into a revised Figure 15.2.1-1, renumber existing Figure 15.2.1-1 as 15.2.1-2, and add a footnote to each figure concerning the applicability of the figures to each unit.

   Basis for TS 15.2.3 (page 15.2.3-6): Revise references to Figures 15.2.1-1 and 15.2.1-2.

b. TS 15.2.2 Basis (page 15.2.2-1): Remove specific value for the reactor high-pressure trip and clarify that the reactor coolant system (RCS) pressure for the hypothetical locked rotor and rod ejection accidents use the faulted condition stress limit acceptance criterion of 3105 psig.

c. TS 15.2.3.1.B(2) (page 15.2.3-1): Revise the high pressurizer pressure trip setpoint to allow operation at either 2000 psia or 2250 psia.

TS 15.3.1.G.2 (page 15.3.1-19): Add an RCS pressurizer pressure operating limit of ≥ 2205 psig for operation at 2250 psia for Unit 2.

d. TS 15.2.3.1.B(3) (page 15.2.3-2): Revise the low pressurizer pressure trip setpoint for operation at either 2000 psia or 2250 psia.

e. TS 15.2.3.1.B(4) (page 15.2.3-2): Revise the values of overtemperature ΔT input parameters corresponding to operation at both 2000 psia and 2250 psia.

TS 15.3.1.G.1 (page 15.3.1-19): Revise the average RCS temperature (T_{avg}) from below 578 °F to a range of ≥ 557 °F and ≤ 573.9 °F.

TS 15.2.3.1.B(5) (page 15.2.3-3): Revise the values of the overpower ΔT input parameters corresponding to operation at both 2000 psia and 2250 psia.

f. TS 15.2.3.1.C(2) (page 15.2.3-3a): Add a low-low steam generator water level trip setpoint of ≥ 20 percent.

g. Table 15.3.5-1(7): Add a low-low steam generator water level trip setpoint of ≥ 20 percent.

h. TS 15.5.3.B.3 (page 15.5.3-3) Modify the nominal liquid volume of the RCS at rated operating conditions from 6040 cubic feet to a nominal RCS volume (both liquid and steam) of 6500 ft^3 for Unit 1 and 6643 ft^3 for Unit 2 at rated operating conditions and zero percent steam generator tube plugging.

i. TS 15.1, "Definitions;" TS 15.3.1, "Reactor Coolant System," Section C,"Maximum Coolant Activity," which includes Figure 15.3.15, "TS 15.3.4, "Steam and Power Conversion System;" TS 15.4.1, "Operational Safety Review;" and TS 15.6.9, "Plant Reporting Requirements" (pages 15.1-6, 15.3.1-9, 15.3.1-10, 15.3.4-2, 15.3.4-3, 15.6.9-3, Figure 15.3.1-5, Table 15.3.5-1 (page 1 of 2), and Table 15.4.1-2 (pages 1 and 2 of 5)): Revise the definition of Dose Equivalent I-131 from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," to Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," dated September 1988.

Revise the limits of RCS specific activity from 1.0 to 0.8 microcurie per gram Dose Equivalent I-131 and secondary steam generator coolant specific activity from 1.2 microcuries per cubic centimeter to 1.0 microcurie per gram Dose Equivalent I-131.

2.0 EVALUATION

The proposed changes are based on recalculated setpoints and uncertainties using a methodology in accordance with the guidance of ISA-S67.04, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," which has been endorsed by the staff in Regulatory Guide 1.105, Revision 2, "Instrumentation Setpoints for Nuclear Safety Related Instrumentation." The effect of the revised setpoints on applicable accident analyses was reviewed by the staff. In addition, the staff performed an independent dose assessment for the following analyzed accidents, included in Chapter 14 of the Final Safety Analysis Report (FSAR), potentially affected by the changes in steam generators: control rod ejection (CRE), reactor coolant pump locked rotor (RCPLR), steam generator tube rupture (SGTR), and main steamline break (MSLB). The staff evaluation of the proposed changes is discussed below.

2.1 TS 15.2.1 - Reactor Core Safety Limits

TS 15.2.1 states that the combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown on Figures 15.2.1-1 and 15.2.1-2, "Reactor Core Safety Limits," for Units 1 and 2, respectively.

The reactor core safety limits are designed to maintain the integrity of the fuel cladding and represent the combination of thermal power, RCS pressure, and average temperature for which the calculated departure from nucleate boiling ratio (DNBR) is no less than the design limit DNBR or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The reactor core safety limits were recalculated to include a slightly higher pressure for the highest pressure safety limit (2425 psia as compared to the current 2400 psia) because of the change in the high pressurizer pressure trip, and to correct a previously evaluated departure from nucleate boiling (DNB) analysis discrepancy. The new limit is consistent with the high pressurizer trip point assumed in the loss of load accident. In order to maintain one set of reactor core safety limits for both units, the Unit 2 safety analyses performed by Westinghouse with the new steam generators also encompass operation of Unit 1. The analyses were performed under the operating conditions associated with the new steam generators (i.e., an average coolant temperature window between 557 °F and 573.9 °F, an operating pressure of either 2000 psia or 2250 psia, a reduced thermal design flow, a slightly larger primary volume, and a slightly smaller secondary volume). Therefore, the licensee has proposed combining the "Reactor Core Safety Limits," given in TS Figure 15.2.1-1 for Unit 1 and in TS Figure 15.2.1-2 for Unit 2 into one figure (Figure 15.2.1-1). Existing Figure 15.2.1-2 would be deleted. The analyses were performed in accordance with NRC-approved methodologies, the trip setpoints were consistent with the TS values, and the results indicate that all design-basis acceptance criteria continue to be met.

Therefore, the proposed changes to the reactor core safety limits are acceptable.

The Unit 2 change will be made immediately. However, in order to avoid the possibility of an inadvertent reactor trip, Unit 1 will continue to operate with the existing safety limit values until the 1997 fall refueling outage (R24). Therefore, a footnote will be added to the appropriate TS to clarify that Figure 15.2.1-1 applies to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, revised Figure 15.2.1-2 applies to Unit 1.

2.2 TS 15.2.2 Basis - Safety Limit, Reactor Coolant System Pressure

The current TS 15.2.2 Basis specifies the reactor high pressure trip of 2385 psig. The basis states that the nominal settings of the power-operated relief valves, the reactor high-pressure trip, and the safety valves have been established to assure never reaching the RCS pressure safety limit. The initial hydrostatic test was conducted at 3110 psig to assure the integrity of the RCS.

The licensee proposes to remove the reactor high-pressure trip setpoint from the bases since the new setpoints are included in TS Section 15.2.3. In addition the licensee proposes to clarify the basis to reflect that the RCS pressure for the hypothetical locked rotor and rod ejection accidents use the faulted condition stress limit acceptance criterion of 3105 psig. Since the 3105 psig is the design basis for these reactor vessels, the staff finds the change acceptable.

2.3 TS 15.2.3.1.B(2) - High Pressurizer Pressure Reactor Trip

The licensee proposed to establish a new TS high pressurizer pressure reactor trip setpoint limit of ≤ 2210 psig for operation at a primary system pressure of 2000 psia. The existing TS setpoint limit of ≤ 2385 psig will remain applicable for operation at a primary system pressure of 2250 psia. The proposed trip setpoint and allowable value of 2191.78 psig ensures the analytical limit of 2235 psig is met with excess margin for operation at 2000 psia.

The high pressurizer pressure reactor trip function is utilized in the loss of load transient as described in Section 14.1.9 of PBNP's FSAR. During a loss of load transient, the steam load on the plant's secondary side is greatly reduced, leading to a condition of decreased heat removal by the secondary system. If the reactor continues to generate heat, the reactor coolant will heat up, expand, compress the bubble in the pressurizer, and pressurize the RCS. To limit the maximum pressure that the RCS experiences during such an event, the high pressurizer pressure reactor trip function is utilized to trip the reactor and, thereby, significantly reduce the heat input from the core into the reactor coolant. In Section 14.1.9 of the FSAR, this scenario is analyzed for the following four cases:

Case a - Total loss of steam load with minimum reactivity feedback, assuming full credit for pressurizer pressure control (i.e., pressurizer spray and pressurizer power operated relief valves). Initial conditions are the normal full power operating conditions at 2000 psia.

Case b - Total loss of steam load with maximum reactivity feedback, assuming full credit for pressurizer pressure control. Initial conditions are the normal full power operating conditions at 2000 psia.

Case c - Total loss of steam load with minimum reactivity feedback and no credit for pressurizer pressure control. Initial conditions are the normal full power operating conditions at 2250 psia.

Case d - Total loss of steam load with maximum reactivity feedback and no credit for pressurizer pressure control. Initial conditions are the normal full power operating conditions at 2250 psia.

The trip setpoints used in the above analyses were 2250 psia for cases a and b and 2425 psia for cases c and d. These values allow a 25-psi margin between the analysis setpoints and the TS-required setpoints. Therefore, for instrument uncertainties less than 25 psi, this allowance ensures conservative TS limits with respect to the setpoints used in the analyses.

In cases a and b the reactor was tripped by the overtemperature delta-T reactor trip signal (a different protective signal). In cases c and d the reactor was tripped by the pressurizer high pressure reactor trip signal. Case a was the limiting case from a DNBR perspective. However, the minimum DNBR reached in this case was within the DNBR limit and, therefore, fuel cladding integrity was not challenged. Case c was the limiting case from a maximum RCS pressure perspective. In this case, the maximum RCS pressure reached was 2740 psia, which was within the NRC acceptance criterion of 110 percent of the RCS design pressure. The analyses further show that the main steam system pressure is also maintained below the limit of 110 percent of its design pressure. In addition, from a maximum RCS pressure perspective, the proposed TS high pressurizer pressure reactor trip setpoint of ≤ 2210 psig for operation at 2000 psia is more conservative than the current TS value of ≤ 2385 psig. Based on the above discussion, the staff finds this change acceptable.

2.4 <u>TS 15.2.3.1.B(3) - Low Pressurizer Pressure Reactor Trip</u>

The licensee proposed to change the TS low pressurizer pressure reactor trip setpoint limit from ≥ 1865 psig to ≥ 1905 psig for operation at a primary system pressure of 2250 psia and from ≥ 1790 psig to ≥ 1800 psig for operation at a primary system pressure of 2000 psia. The proposed trip setpoints are in the conservative direction and ensure that the analytical limit will be met with excess margin.

The low pressurizer pressure reactor trip function is utilized to protect the core against excessive steam voids during transients and accidents that lead to depressurization of the RCS at normal operating temperatures. In addition, the low pressurizer pressure reactor trip setpoint is higher than the low pressurizer pressure safety injection setpoint and, therefore, trips the reactor in anticipation of further depressurization and subsequent safety injection. Analyses that take credit for the low pressurizer pressure reactor trip include those for the small-break loss-of-coolant accident (SBLOCA), SGTR, and MSLB accidents. These accidents are analyzed in FSAR Sections 14.3.1, 14.2.4, and 14.2.5, respectively.

The proposed TS setpoints of ≥ 1905 psig for operation at a primary system pressure of 2250 psia and ≥ 1800 psig for operation at primary system pressure of 2000 psia are more conservative than the existing TS setpoints. The licensee has shown that the acceptance criteria contained in 10 CFR 50.46 and 10 CFR Part 100 continue to be met for the SBLOCA, SGTR, and steam line break accidents. In addition, low pressurizer pressure reactor trip values used in the analyses were a minimum of 40 psi lower (i.e., more conservative) than the proposed TS required setpoints. Therefore, for instrument uncertainties less than 40 psi, this allowance ensures that the TS limits are conservative with respect to the analyses. Based on the above discussion, the staff finds this change acceptable.

2.5 TS 15.2.3.1.B(4) - Overtemperature Delta T Reactor Trip
TS 15.2.3.1.B(5) - Overpower Delta T Reactor Trip

TS 15.2.3, Specification 1.B(4) gives the overtemperature delta-T (OTΔT) reactor trip setpoint function and parameter values. Specification 1.B(5) gives the overpower delta-T (OPΔT) reactor trip setpoint function and parameter values. The overtemperature ΔT signal initiates a reactor trip in the event of an uncontrolled rod withdrawal and loss of load. The overtemperature ΔT calculation inputs have been revised to accommodate operation at both 2000 psia and 2250 psia and maintain the safety margin.

Revisions to the OTΔT and OPΔT reactor trips specified in TS 15.2.3 are proposed as a result of the replacement steam generators and to provide for operation at either 2000 psia or 2250 psia primary pressure. The proposed revisions include changing the OTΔT T′, P′, K_1, K_2, and K_3 terms and the OTΔT T′ and K_4 terms as follows:

$T' \leq 572.9\ ^{\circ}F$
$P' = 2235$ psig (2250 psia operation only)
$P' = 1985$ psig (2000 psia operation only)
$K_1 \leq 1.19$ (2250 psia operation only)
$K_1 \leq 1.14$ (2000 psia operation only)
$K_2 = 0.025$ (2250 psia operation only)
$K_2 = 0.022$ (2000 psia operation only)
$K_3 = 0.0013$ (2250 psia operation only)
$K_3 = 0.001$ (2000 psia operation only)
$K_4 \leq 1.09$ of rated power

The analyses to support the proposed changes were performed in accordance with NRC-approved methodologies and the results indicate that all design-basis acceptance criteria continue to be met. Therefore, the proposed changes provide adequate protection over the full range of expected RCS operation and maintain the safety margins for Unit 2 with the replacement steam generators. Since the safety analyses and evaluations were performed to cover both units, the proposed changes are acceptable for Unit 1 operation as well as Unit 2 operation. However, in order to eliminate the possibility of an inadvertent reactor trip while adjusting the setpoints, the adjustments will be made during the next scheduled refueling outage for each unit. A footnote will be added to clarify this situation.

2.6 <u>TS 15.2.3.1.C(2) and TS Table 15.3.5-1 - Low-Low Steam Generator Level Setting Limit Changes</u>

The low-low steam generator level reactor trip function is utilized to protect the steam generators and reactor in the case of a sustained steam/feedwater flow mismatch of insufficient magnitude to cause a flow mismatch reactor trip. The purpose of the steam/feedwater flow mismatch trip is to protect the reactor from a sudden loss of its heat sink.

The licensee proposed to change the TS low-low steam generator level trip setpoint limit from ≥ 5 percent to ≥ 20 percent. This change was necessary because (1) the lower taps for the level instruments on the newly installed steam generators are located at a lower elevation than those on the original steam generators, and (2) the level instruments on the new steam generators have wider spans than those on the original steam generators. These changes in configuration of the level instruments make the original 5 percent setpoint limit nonconservative with respect to the analyses of record, if applied to the new steam generators. That is, an indication of 5 percent in the new steam generators represents a lower level of inventory than the same indication of 5 percent in the old steam generators. However, the proposed 20 percent limit for the new steam generators is also lower than the original 5 percent limit for the old steam generators. Therefore, the licensee reanalyzed the associated FSAR Chapter 14 accident scenarios for the proposed limit of 20 percent. Reanalyses were performed for loss of normal feedwater (FSAR Section 14.1.10) and loss of AC power to the station auxiliaries (FSAR Section 14.1.11), for both Point Beach units.

During a loss of normal feedwater event, the normal path of supplying water to the steam generators is lost. This causes a reduction in the capabilities of the steam generators to remove heat from the RCS. If the reactor core is allowed to generate heat in excess of the capabilities of the steam generators, the reactor coolant will heat up, expand, compress the bubble in the pressurizer, and pressurize the closed RCS. Therefore, the low-low steam generator level reactor trip function is utilized to trip the reactor and, thereby, significantly reduce the amount of heat generated in the core. However, if sufficient RCS cooling is not provided, residual heat generated after the reactor trip can still cause the RCS to pressurize to the point where pressurizer relief valves are actuated, leading to a loss of reactor coolant. Therefore, the auxiliary feedwater (AFW) system is also actuated at the low-low steam generator level setpoint. The AFW system provides emergency feedwater to the steam generators and allows for heat removal to continue after the loss of normal feedwater.

The loss of AC power to the station auxiliaries event is similar to the loss of normal feedwater event and, itself, results in a loss of normal feedwater. However, whereas the reactor coolant pumps (RCPs) are assumed to continue to run throughout the loss of normal feedwater event, the RCPs are assumed to lose AC power and coast down during the loss of AC power to the station auxiliaries event.

In the analyses of these events, the licensee used a value of 10 percent for the low-low steam generator level trip setpoint. This value allows a 10-percent margin between the analyses setpoint of 10 percent and the proposed TS setpoint of ≥ 20 percent. Therefore, for instrument uncertainties less than 10 percent steam generator level, the proposed TS setpoint ensures conservative TS limits with respect to the analyses.

The analyses of both events show that flow from one motor-driven AFW pump (i.e., 200 gpm) to a single steam generator is sufficient to keep the pressurizer pressure well below the lift setpoints of the pressurizer relief valves. The pressure in the main steam system has been shown to remain below the limit of 110 percent of the system's design pressure. These analyses further show that the fuel cladding integrity is not challenged by these events for either Point Beach unit. Therefore, the staff finds the proposed change acceptable. In addition, since the Unit 1 steam generators were not replaced, the staff also finds acceptable, based on the approved previous analyses, the licensee's request to maintain the 5 percent setting limit for Unit 1 until such time as the narrow range level instrumentation is modified to be consistent with that of Unit 2.

2.7 TS 15.3.1.G - Full Power Average RCS Temperature Operating Range

The proposed full power average RCS temperature operating range is between 557 °F and 573.9 °F. The upper limit (573.9 °F) required further evaluation. The licensee decided to address the affect of an increase of 3.9 °F over the previously analyzed temperature of 570 °F by taking a peak clad temperature penalty. The licensee's reanalysis of the FSAR Chapter 14 events considered the effect of the proposed lower full power average temperature. The only non-LOCA event determined to be affected is the uncontrolled rod cluster control assembly (RCCA) withdrawal event from full power conditions. The results indicate that the acceptance criterion for minimum DNBR continues to be met for this event.

2.8 TS 15.5.3.B.3 - Nominal Reactor Coolant System Volume

TS 15.5.3.B.3 currently states that the nominal liquid volume of the RCS at rated operating conditions is 6040 cubic feet.

The nominal RCS volume (both liquid and steam) at rated operating conditions and zero percent steam generator tube plugging specified in TS 15.5.3.B.3 is being modified to read 6500 ft^3 (Unit 1) and 6643 ft^3 (Unit 2). The existing wording describes only an approximate liquid volume of the RCS for both units. The proposed changes include the pressurizer steam space volume and will more accurately describe the total volume for each unit. The volume for Unit 1 is not changing and that for Unit 2 is increasing due to the higher volume associated with the new steam generators. The smaller volume associated with Unit 1 is typically more limiting with respect to the FSAR safety analyses associated with core cooling and the Unit 1 coolant volume is not being changed.

Revised radiological consequences resulting from (1) steam generator tube rupture (SGTR), (2) main steam line break (MSLB), (3) control rod ejection (CRE), and (4) reactor coolant pump locked rotor (RCPLR) were evaluated by the licensee for the new steam generator volumes and current system operation based on emergency operating procedures (EOPs). The calculations submitted included an increase in power level even though the power level change was not part of the application. The licensee concluded that the radiological consequences of SGTR, MSLB, CRE, and RCPLR are acceptable for the replacement steam generators.

Additionally, the licensee has submitted for review the radiological consequences of a LOCA in support of an application to amend TS to reflect revised system requirements to ensure post-accident containment cooling capability (TS CR-192, September 30, 1996). The revised analysis included the parameters associated with this application for TS changes proposed for replacement steam generators and system operation in accordance with the EOPs. The licensee's applications indicate that the radiological consequences for a LOCA are higher than the radiological consequences for the SGTR, MSLB, CRE, and RCPLR.

The staff reviewed the licensee's analyses and compared the potential radiological consequences to the current licensing basis, proposed radiological consequences associated with a LOCA, and the acceptance criteria presented in 10 CFR Part 100 and the dose limits included in General Design Criterion 19 of Appendix A to 10 CFR Part 50 (GDC 19). The licensee's commitment to meet the dose limits specified in GDC 19 was made as a result of NUREG-0737, Section III.D.3.4. The LOCA analysis used for comparison was the analysis included in PBNP's TS CR-192 for revised system requirements to ensure post-accident cooling capability which included the changes to system operation as specified in the EOPs and the parameters associated with the Delta 47 replacement steam generators.

The staff independently assessed the postulated radiological doses for individuals located at the Exclusion Area Boundary (EAB), Low-Population Zone (LPZ), and control room for SGTR, MSLB, CRE, RCPLR, and LOCA. The LOCA analysis still remains limiting for the proposed changes associated with steam generator replacement, revised system operation in accordance with EOPs, and proposed changes to post-accident cooling capability. The staff's evaluation of LOCA radiological consequences will be included in the safety evaluation associated with PBNP's TS CR-192 which is required prior to restart of Unit 2.

The staff has assessed those accidents for which the change to the Delta 47 replacement steam generators have an impact upon the offsite and control room operator doses and determined that the doses would not exceed the dose limits in 10 CFR Part 100 or GDC 19 of 10 CFR Part 50, Appendix A, for either offsite locations or control room operators. Therefore, the staff finds the proposed replacement of the existing steam generators with the Delta 47 steam generators acceptable from a radiological standpoint at a core reactor power level of 1518.5 megawatts thermal.

2.9 TS 15.1, "Definitions;" TS 15.3.1, "Reactor Coolant System," Section C, "Maximum Coolant Activity," "TS 15.3.4, "Steam and Power Conversion System;" TS 15.4.1, "Operational Safety Review;" and TS 15.6.9, "Plant Reporting Requirements"

The current TS define dose equivalent iodine as 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 determination are listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." TS 15.3.1.C, TS 15.3.4, TS 15.4.1, and TS 15.6.9 include a limit for RCS specific activity of 1.0 microcurie per gram Dose Equivalent I-131.

The proposed TS changes will revise the limits of RCS specific activity to 0.8 microcurie per gram Dose Equivalent I-131 and revise the secondary side steam generator dose equivalent I-131 activity to 1.0 microcurie per gram. The licensee proposes to use the dose equivalent I-131 defined in Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," dated September 1988. The proposed change ensures consistency between the TS and new radiological analyses. The effect of this change causes the Dose Equivalent I-131 limit to be reduced by 20 percent.

The staff finds the proposed changes acceptable since the net effect of both the change in limits and the change in the standard result in equivalent doses consequences.

2.10 TS 15.6.9.C, Monthly Operating Reports

TS page 15.6.9-3 was updated to reflect the licensee's pen and ink deletion of conflicting submittal directions for the Monthly Operating Reports. This change is permitted by NRC's Final Rule on "Domestic Licensing of Production and Utilization Facilities; Communications Procedures Amendments," dated November 6, 1986 (51 FR 40303). Therefore, the staff finds this change acceptable.

2.11 Evaluation Summary

Based on the above, the staff concludes that the licensee's proposed changes to the setpoints and TS limits for steam generator replacement are consistent with the guidance of Regulatory Guide 1.105, Revision 2, "Instrumentation Setpoints for Nuclear Safety Related Instrumentation," reflect the parameters for both Unit 1 and 2 steam generators, and reflect the new analyzed conditions. Therefore, the staff finds the changes acceptable.

3.0 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.

4.0 <u>ENVIRONMENTAL CONSIDERATION</u>

These amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change 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 effluent 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 (61 FR 34903, 61 FR 34904, and 62 FR 17243). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also change recordkeeping, reporting or administrative procedures or requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 <u>CONCLUSION</u>

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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