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

April 7, 1994

Docket No. 50-305

Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service
Corporation
Post Office Box 19002
Green Bay, Wisconsin 54037-9002

Dear Mr. Schrock:

SUBJECT: AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NOS. M77357 AND M77427)

The Commission has issued the enclosed Amendment No. 108to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications (TS) in response to your application dated May 5, 1993, as supplemented March 4, 1994. This submittal superseded previous submittals on the same subject dated May 9, 1991, as supplemented June 25 and June 26, 1991, and July 24, 1992.

The amendment was submitted as a result of NRC recommendations pertaining to NRC Generic Letter 90-06 for Generic Issue (GI) 70, "Power-Operated Relief Valve (PORV) and Block Valve Reliability," and GI 94, "Additional Low-Temperature Overpressure Protection (LTOP) for Light Water Reactors." The amendment revises TS Section 3.1 by adding restrictions on the restart of an inactive reactor coolant pump, modifying the limiting conditions for operation of the pressurizer power-operated relief valves (PORVs) and associated block valves, and adding provisions to ensure that adequate low-temperature overpressure protection (LTOP) is available. Additionally, this amendment modifies the limiting conditions for operation for reactor coolant temperature and pressure by adding Figure TS 3.1-4 to define 10 CFR 50 Appendix G pressure and temperature limitations for LTOP evaluation through the end of operating cycle 20. This change was submitted as a result of a Westinghouse Nuclear Safety Advisory Letter (NSAL) concerning LTOP setpoints received by Wisconsin Public Service Corporation (WPSC) on March 22, 1993. This issue was also discussed at a meeting between WPSC and the NRC staff held on April 22, 1993.

DF01

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

Sincerely,

Original signed by Richard J. Laufer

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

Enclosures:
1. Amendment No. 108 to License No. DPR-43
2. Safety Evaluation

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See next page

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Wisconsin Public Service Corporation   Kewaunee Nuclear Power Plant

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

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated May 5, 1993, as supplemented March 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.108, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett for

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

Attachment:
Changes to the Technical Specifications

Date of issuance: April 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
TS i	TS i
TS vi	TS vi
TS 3.1-1	TS 3.1-1
TS 3.1-3	TS 3.1-3
TS 3.1-4	TS 3.1-4
TS 3.1-5	TS 3.1-5
TS 3.1-6	TS 3.1-6
TS 3.1-7	TS 3.1-7
TS 3.1-8	TS 3.1-8
TS 3.1-9	TS 3.1-9
--------	TS 3.1-10
--------	TS 3.1-11
TS B3.1-1 through TS B3.1-11 (11 pages)	TS B3.1-1 through TS B3.1-13 (13 pages)
---------	FIGURE TS 3.1-4

TABLE OF CONTENTS
TECHNICAL SPECIFICATIONS
APPENDIX A

LIST OF FIGURES

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

APPLICABILITY

Applies to the Operating status of the Reactor Coolant System (RCS).

OBJECTIVE

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

SPECIFICATIONS

a. Operational Components

1. Reactor Coolant Pumps

   A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   B. When the reactor is in the OPERATING mode, except for low power tests, both reactor coolant pumps shall be in operation.

   C. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures ≤ 338°F unless the secondary water temperature of each steam generator is < 100°F above each of the RCS cold leg temperatures.

2. Decay Heat Removal Capability

   A. At least TWO of the following FOUR heat sinks shall be operable whenever the average reactor coolant temperature is ≤ 350°F but > 200°F.

      1. Steam Generator 1A
      2. Steam Generator 1B
      3. Residual Heat Removal Train A
      4. Residual Heat Removal Train B

      If less than the above number of required heat sinks are operable, corrective action shall be taken immediately to restore the minimum number to the operable status.

4. Pressure Isolation Valves

   A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device during OPERATING and HOT STANDBY modes, except as specified in 3.1.a.4.B. Valve leakage shall not exceed the amounts indicated.

   B. In the event that integrity of any pressure isolation valve as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.[1]

   C. If TS 3.1.a.4.A and TS 3.1.a.4.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the HOT SHUTDOWN condition within the next 4 hours, the INTERMEDIATE SHUTDOWN condition in the next 6 hours and the COLD SHUTDOWN condition within the next 24 hours.

5. Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves

   A. Two PORVs and their associated block valves shall be operable during HOT STANDBY and OPERATING modes.

      1. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, action shall be initiated to:

         - Achieve HOT STANDBY within 6 hours
         - Achieve HOT SHUTDOWN within the following 6 hours

      2. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or action shall be initiated to:

         - Achieve HOT STANDBY within 6 hours
         - Achieve HOT SHUTDOWN within the following 6 hours

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[1]Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position with their power breakers locked out.

3. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours

4. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the block valve to OPERABLE status within 72 hours; otherwise action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours

5. With both block valves INOPERABLE, within 1 hour restore the block valves to OPERABLE status or place their associated PORVs in manual control. Restore at least one block valve to OPERABLE status within the next hour; otherwise, action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours

6. Pressurizer Heaters

   A. At least one group of pressurizer heaters shall have an emergency power supply available when the average RCS temperature is > 350°F.

7. Reactor Coolant Vent System

A. A reactor coolant vent path from both the reactor vessel head and pressurizer steam space shall be operable and closed prior to the average RCS temperature being heated > 200°F except as specified in TS 3.1.a.7.B and TS 3.1.a.7.C below.

B. When the average RCS temperature is > 200°F, any one of the following conditions of inoperability may exist:

1. Both of the parallel vent valves in the reactor vessel vent path are inoperable.

2. Both of the parallel vent valves in the pressurizer vent path are inoperable.

If operability is not restored within 30 days, then within one hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours
- Achieve HOT SHUTDOWN within the following 6 hours
- Achieve COLD SHUTDOWN within an additional 36 hours

C. If no Reactor Coolant System vent paths are operable, restore at least one vent path to operable status within 72 hours. If operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours
- Achieve HOT SHUTDOWN within the following 6 hours
- Achieve COLD SHUTDOWN within an additional 36 hours

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1, TS 3.1-2, and TS 3.1-4. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 20 effective full-power years. Figure TS 3.1-4 is applicable through the end of operating cycle 20 or 17.14 effective full-power years.

    A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

    B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

    C. Figure TS 3.1-4 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 338°F.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.

4. The overpressure protection system for low temperature operation shall be operable whenever one or more of the RCS cold leg temperatures are ≤ 338°F, and the reactor vessel head is installed. The system shall be considered operable when at least one of the following conditions is satisfied:

   A. The overpressure relief valve on the Residual Heat Removal System (RHR 33-1) shall have a set pressure of ≤ 500 psig and shall be aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open.

      1. With one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. Restore the inoperable flow path within 5 days or complete depressurization and venting of the RCS through a ≥ 6.4 square inch vent within an additional 8 hours.

      2. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS through at least a 6.4 square inch vent pathway within 8 hours.

   B. A vent pathway shall be provided with an effective flow cross section ≥ 6.4 square inches.

      1. When low temperature overpressure protection is provided via a vent pathway, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position. If the vent path is provided by any other means, verify the vent pathway every 12 hours.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

   A. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

   B. $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\bar{E}$ is the average sum of the beta and gamma energies in Mev per disintegration)

   whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

   A. With the specific activity of the reactor coolant > 1 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding the limit shown on Figure TS 3.1-3, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within 6 hours.

   B. With the specific activity of the reactor coolant > $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within 6 hours.

   C. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item f, once every 4 hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

d. Leakage of Reactor Coolant

1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 500 gallons per day through any one steam generator. With tube leakage greater than the above limit, reduce the leakage rate within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.

5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is operable.

e. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

1. Concentrations of contaminants in the reactor coolant shall not exceed the following limits when the reactor coolant temperature is > 250°F.

CONTAMINANT	NORMAL STEADY-STATE OPERATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	0.10	1.00
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in TS 3.1.e.1 above are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants cannot be controlled within the transient limits of TS 3.1.e.1 above or returned to the normal steady-state limit within 24 hours, the reactor shall be brought to the COLD SHUTDOWN condition, utilizing normal operating procedures, and the cause shall be ascertained and corrected. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee shall be made before starting.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is ≤ 250°F.

CONTAMINANT	NORMAL CONCENTRATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	Saturated	Saturated
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

5. If the transient limits of TS 3.1.e.4 are exceeded or the concentrations cannot be returned to normal values within 48 hours, the reactor shall be brought to the COLD SHUTDOWN condition and the cause shall be ascertained and corrected.

6. To meet TS 3.1.e.1 and TS 3.1.e.4 above, reactor coolant pump operation shall be permitted for short periods, provided the coolant temperature does not exceed 250°F.

f. Minimum Conditions for Criticality

1. Except during low-power physics tests, the reactor shall not be made critical unless the moderator temperature coefficient is negative.

2. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line shown in Figure TS 3.1-1.

3. Except during low-power physics tests, when the reactor coolant temperature is in a range where the moderator temperature coefficient is positive, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least 1% $\Delta k/k$ until normal water level is established in the pressurizer.

BASES - Operational Components (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part 1 of the specification requires that both reactor coolant pumps be operating when the reactor is in power operation to provide core cooling. Planned power operation with one loop out of service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this mode of operation. The flow provided in each case in Part 1 will keep DNBR well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[1]

The RCS will be protected against exceeding the design basis of the LTOP system by restricting the starting of a RXCP to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ 338°F is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the low temperature overpressure protection (LTOP) system. The LTOP enable temperature of 338°F is based on the 20 effective full-power year curves.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

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[1]USAR Section 7.2.2

The requirement for at least one train of residual heat removal when in the REFUELING MODE is to ensure sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel < 140°F. The requirement to have two trains of residual heat removal operable when there is < 23 feet of water above the reactor vessel flange ensures that a single failure will not result in complete loss-of-heat removal capabilities. With the reactor vessel head removed and at least 23 feet of water above the vessel flange, a large heat sink is available. In the event of a failure of the operable train, additional time is available to initiate alternate core cooling procedures.

Pressurizer Safety Valves (TS 3.1.a.3)

Each of the pressurizer safety valves is designed to relieve 325,000 lbs. per hour of saturated steam at its setpoint. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against overpressurization.

Pressure Isolation Valves (TS 3.1.a.4)

The Basis for the Pressure Isolation Valves is discussed in the Reactor Safety Study (RSS), WASH-1400, and identifies an intersystem loss-of-coolant accident in a PWR which is a significant contributor to risk from core melt accidents (EVENT V). The design examined in the RSS contained two in-series check valves isolating the high pressure Primary Coolant System from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the EVENT V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.(2)

PORVs and PORV Block Valves (TS 3.1.a.5)

The pressurizer power-operated relief valves (PORVs) operate as part of the pressurizer pressure control system. They are intended to relieve RCS pressure below the setting of the code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

The pressurizer PORVs and associated block valves must be operable to provide an alternate means of mitigating a design basis steam generator tube rupture. Thus, an inoperable PORV (for reasons other than seat leakage) or block valve is not permitted in the HOT STANDBY and OPERATING modes for periods of more than 72 hours.

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(2)Order for Modification of License dated 4/20/81

Pressurizer Heaters (TS 3.1.a.6)

Pressurizer heaters are vital elements in the operation of the pressurizer which is necessary to maintain system pressure. Loss of energy to the heaters would result in the inability to maintain system pressure via heat addition to the pressurizer. Hot functional tests[3] have indicated that one group of heaters is required to overcome ambient heat losses. Placing heaters necessary to overcome ambient heat losses on emergency power will assure the ability to maintain pressurizer pressure. Annual surveillance tests are performed to ensure heater operability.

Reactor Coolant Vent System (TS 3.1.a.7)

The function of the high point vent system is to vent noncondensible gases from the high points of the RCS to assure that core cooling during natural circulation will not be inhibited. The operability of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow capacity of one charging pump.[4]

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[3]Hot functional test (PT-RC-31)

[4]Letter from E. R. Mathews to S. A. Varga dated 5/21/82

Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties - (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code[5], and the calculation methods of Footnote[6]. The postirradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and are discussed in detail in Footnote[7].

The method specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{IR} \qquad (3.1b\text{-}1)$$

where

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

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[5]ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, 1986 Edition, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

[6]Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-86.

[7]WCAP-13229, "Heatup and Cooldown Limit Curves for Normal Operation for Kewaunee," M. A. Ramirez and J. M. Chicots, March 1992 (Westinghouse Proprietary Class 3)

From equation (3.1b-1) the variables that affect the heatup and cooldown analysis can be readily identified. K_{Im} is the stress intensity factor due to membrane (pressure) stress. K_{It} is the thermal (bending) stress intensity factor and accounts for the linearly varying stress in the vessel wall due to thermal gradients. During heatup K_{It} is negative on the inside and positive on the outer surface of the vessel wall. The signs are reversed for cooldown and, therefore, an ID or an OD one quarter thickness surface flaw is postulated in whichever location is more limiting. K_{IR} is dependent on irradiation and temperature and, therefore, the fluence profile through the reactor vessel wall and the rates of heatup and cooldown are important. Details of the procedure used to account for these variables are explained in the following text.

Following the generation of pressure-temperature curves for both the steady-state (zero rate of change of temperature) and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location. The pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan[8][9]. Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above. The derivation of the limit curves is consistent with Footnotes[10][11].

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 9878[12], weld metal Charpy test specimens from Capsule R indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (235°F).

The results of Irradiation Capsules V, R, and P analyses are presented in WCAP 8908[13], WCAP 9878, and WCAP-12020[14], respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 20 effective full-power years.

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[8] "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

[9] ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, 1986 Edition, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

[10] NRC Regulatory Standard Review Plan Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits" 1974

[11] ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, Summer 1984 Addenda, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

[12] S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

[13] S. E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

[14] S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

A limit curve (Figure TS 3.1-4) for evaluation of low temperature overpressure protection (LTOP) events has been calculated using the methodology of Regulatory Guide 1.99, Revision 2, Position C.2. The derivation of the LTOP evaluation curve is consistent with Footnotes[15][16]. This curve is applicable for 17.14 effective full-power years of fluence (through the end of operating cycle 20). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

Pressurizer Limits - (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Low Temperature Overpressure Protection - (TS 3.1.b.4)

The low temperature overpressure protection system must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2. Based on the Kewaunee Appendix G pressure-temperature limits calculated through 20 effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 338°F and the head is on the reactor vessel. The LTOP system is considered operable when all 4 valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-4.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of ± 3% (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

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[15]NRC Regulatory Standard Review Plan Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits," 1974

[16]ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, Summer 1984 Addenda, Non-Mandatory Appendix G - "Protection Against Non-ductile Failure."

If one train of RHR suction piping to RHR 33-1 is isolated, the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within 5 days. If the isolated flowpath cannot be restored within 5 days, the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional 8 hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, the system can still be considered operable if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve ($\geq$ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), the vent path is considered secured in the open position.

Maximum Coolant Activity (TS 3.1.c)

This specification is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is at the allowable limit. The potential release of activity to the atmosphere has been evaluated to insure that the public is protected.

Rupture of a steam generator tube would allow reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases[17] which would be released to the atmosphere from the air ejector or a relief valve. Activity could continue to be released until the operator could reduce the Reactor Coolant System pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, followed by isolation of the faulty steam generator by the operator within one-half hour after the event. During this period, 120,000 lbs. of reactor coolant are discharged into the steam generator.[17]

The limiting off-site dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, for purposes of analysis, the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose, has been used. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

[17]USAR Section 14.2.4

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$Dose,\ rem = 1/2\ [\bar{E} \cdot A \cdot V \cdot \frac{X}{Q} \cdot (3.7 \times 10^{10})\ (1.33 \times 10^{-11})]$$

Where: $\bar{E}$ = average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m^3)

$\bar{E}A$ = 91 Mev Ci/dis m^3 (the maximum per this specification)

$\frac{X}{Q}$ = 2.9×10^{-4} sec/m^3, the 0-2 hr. dispersion coefficient at the site boundary prescribed by the Commission

V = 77 m^3, which corresponds to a reactor coolant liquid mass of 120,000 lbs.

The resultant dose is < 0.5 rem at the site boundary.

The action statement permitting power operation to continue for limited time periods with reactor coolant specific activity > 1 μCi/grams DOSE EQUIVALENT I-131, but within the allowable limit shown in Figure TS 3.1-3, accommodates the possible iodine spiking phenomenon which may occur following changes in thermal power.

Reducing average coolant to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

Leakage of Reactor Coolant (TS 3.1.d)[18]

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}\,\mu$ Ci/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average X/Q = 2.0×10^{-6} sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.5 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve (12) hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of 10 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.

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[18] USAR Sections 6.5, 11.2.3, 14.2.4

The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.

If leakage is to the containment, it may be identified by one or more of the following methods:

A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.

B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.

C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

   If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

<u>Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)</u>

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[19]

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[19] USAR Section 4.2

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank[20]. Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[21][22]

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

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[20] USAR Section 9.2

[21] USAR Table 3.2-1

[22] USAR Figure 3.2-8

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[23] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.2 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[24]

The requirement that the pressurizer is partly voided when the reactor is < 1% subcritical assures that the Reactor Coolant System will not be solid when criticality is achieved.

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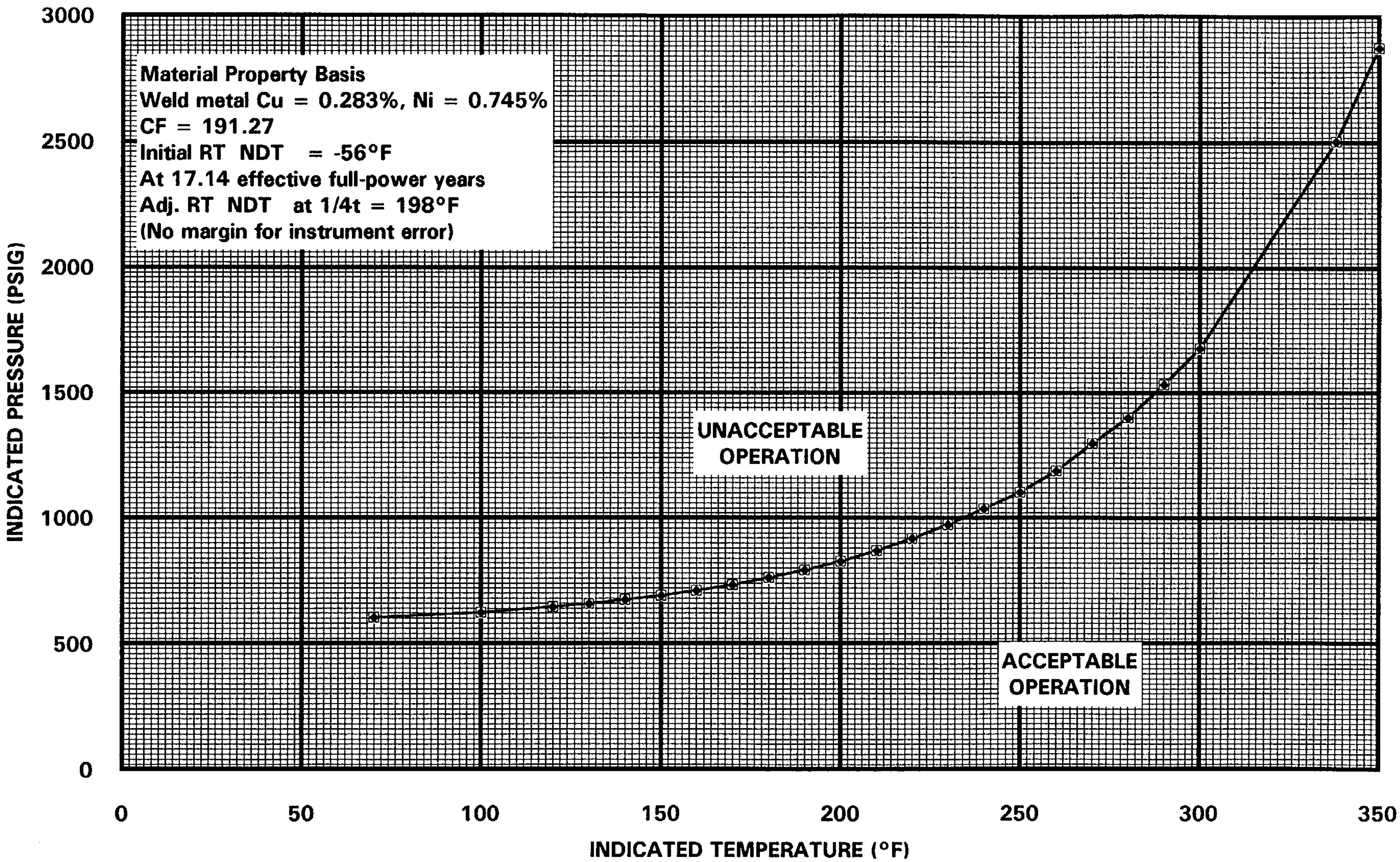
[23] USAR Figure 3.2-9

[24] USAR Table 3.2-1

FIGURE TS 3.1-4
LOW TEMPERATURE OVERPRESSURE PROTECTION CURVE
APPLICABLE FOR PERIODS UP TO END OF OPERATING CYCLE 20



Amendment No.108

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO.108 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated May 5, 1993, as supplemented March 4, 1994, the Wisconsin Public Service Corporation (WPSC), the licensee, submitted a request for revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). This submittal was made in response to Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1990. This submittal superseded previous submittals on the same subject dated May 9, 1991, as supplemented June 25 and June 26, 1991, and July 24, 1992.

The proposed amendment would revise TS Section 3.1 by adding restrictions on the restart of an inactive reactor coolant pump, modifying the limiting conditions for operation of the pressurizer power-operated relief valves (PORVs) and associated block valves, and adding provisions to ensure that adequate low-temperature overpressure protection (LTOP) is available. Additionally, the proposed amendment would modify the limiting conditions for operation for reactor coolant temperature and pressure by adding Figure TS 3.1-4 to define 10 CFR 50 Appendix G pressure and temperature limitations for LTOP evaluation through the end of operating cycle 20. This proposed change was submitted as a result of a Westinghouse Nuclear Safety Advisory Letter (NSAL) concerning LTOP setpoints received by Wisconsin Public Service Corporation (WPSC) on March 22, 1993. This issue was also discussed at a meeting between WPSC and the NRC staff held on April 22, 1993.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's technical specifications were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

2.0 EVALUATIONS

2.1 Evaluation for Generic Issue 70

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

The proposed TS changes in response to Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," consist of the following changes to TS 3.1, "Reactor Coolant System."

TS 3.1.a.5.A is being replaced and the associated bases are being modified. The new sections 3.1.a.5.A.1 through 3.1.a.5.A.4 describe the actions required to be taken when one or both PORVs are inoperable due to excessive leakage and due to causes other than excessive leakage. The new TS also describe the actions required to be taken when one or both PORV block valves are inoperable. The associated bases have been expanded by stating that the PORVs and associated block valves must be operable to provide an alternate means of mitigating a design basis steam generator tube rupture.

Since the proposed changes discussed above are consistent with the modified Standard TS (STS) included with the GL, the staff finds them acceptable.

The licensee's proposal did not adopt the PORV surveillance requirements of the modified STS included with the GL. The licensee's surveillance requirement states that the PORVs will be tested in accordance with the inservice testing (IST) program. In Attachment 2 to their May 5, 1993, submittal, the licensee committed to revise the IST plan to perform the 18 month stroke test of the PORVs during Kewaunee modes of Hot Shutdown or Intermediate Shutdown.

Since this commitment meets the intent of the GL requirement, the staff finds it acceptable.

Attachment 1 to the licensee's May 5, 1993, submittal also addressed staff concerns regarding the testing of PORV control air valves raised in a January 21, 1993, letter. The licensee states that the PORV accumulators are not required for compliance with previous regulatory positions. The PORV accumulators were installed to ensure PORV availability should the normal air supply valve, IA 101, close on a containment isolation signal and to eliminate the need for operator action to reopen IA 101 to restore the PORV air supply. Subsequent to this modification, the instrument air (IA) system was modified to remove the containment isolation signal from IA 101 and change the operation of IA 101 from a fail close to a fail open valve. These modifications eliminated the design features that provided the licensee's basis for installing the PORV accumulators.

The IA system is supplied by redundant emergency powered components, is not isolated by a containment isolation signal or failure of the essential power to the IA supply valve (IA 101), and provides a highly reliable normal IA supply to the PORVs. The IA system is not subject to credible single active failures given the redundancy of components, emergency power supplies, and header piping. Additionally, IA valves 101, 102, and 103 are included in the licensee's IST program.

The PORVs at Kewaunee are not used for two of the three design basis events discussed in NUREG-1316. Of the three events, the PORVs are used as the second choice for RCS depressurization in a steam generator tube rupture (SGTR). The licensee has examined the core melt frequency attributable to a SGTR and found it acceptably low.

Given the highly reliable configuration of the normal IA supply system and the limited reliance on PORVs for mitigation of design basis events, the licensee has determined that the accumulator air supply system is not necessary for PORV operability.

The staff has reviewed the licensee's justification summarized above, and finds that the valves in their PORV control air system are adequately included within the scope of a program covered by Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," of Section XI of the ASME Boiler and Pressure Vessel Code.

2.2 <u>Evaluation for Generic Issue 94</u>

The actions proposed by the NRC staff to improve the availability of the low-temperature overpressure protection (LTOP) system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

The proposed TS changes in response to Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," consist of the following changes to TS 3.1, "Reactor Coolant System."

TS 3.1.a.1.C is being added and the associated bases are being supplemented. TS 3.1.a.1.C reads:

> "A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 372 degrees F unless the secondary water temperature of each steam generator is less than 100 degrees F above each of the RCS cold leg temperatures."

This restriction on pump startup has been administratively enforced at Kewaunee. It is now being added to the TS to provide additional control to ensure that the limiting energy input transient assumed in the design basis of the LTOP system is not violated. The 100 degree F temperature difference restriction is part of the Kewaunee LTOP design basis which has been previously accepted by the staff. The inclusion of this restriction in the TS is consistent with the guidance provided in the GL and is therefore acceptable.

TS 3.1.b.4 is being added and the associated bases are being supplemented. This TS specifies that the LTOP system shall be operable whenever the RCS cold leg temperature is less than or equal to 338 degrees F. Operability is defined as: (1) having the Residual Heat Removal (RHR) system overpressure relief valve (RHR 33-1) set at less than or equal to 500 psig with the system aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open; or (2) by providing a vent pathway. This TS also specifies that with one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. The inoperable flow path must be restored within 5 days or a complete depressurization and venting of the RCS shall be completed within an additional 8 hours. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS is required within 8 hours.

This proposed TS is consistent with the guidance provided in the GL for plants which do not use PORVs for LTOP with the exception of the allowable outage time (AOT) for an inoperable flowpath. The model STS provided with the GL specify a 7-day AOT for an inoperable PORV (LTOP flowpath) with RCS temperature greater than 200 degrees F (MODE 4) and a 24-hour AOT for an inoperable PORV (LTOP flow path) with RCS temperature less than 200 degrees F (MODES 5 and 6).

Attachment 3 to the licensee's May 5, 1993, submittal, as modified by the March 4, 1994, submittal, provides a detailed justification for their proposed 5 day AOT for an inoperable LTOP flowpath with RCS temperature less than 200 degrees F. The licensee cites (1) their system design; (2) historical LTOP usage time; (3) existing TS; and (4) the time required to depressurize and vent the RCS as justification for their proposed 5-day AOT.

(1) The licensee states that the LTOP system at Kewaunee consists of a Crosby Size 4P6 ASME Section III, Class 2 safety relief valve (RHR 33-1) located within containment on the common normal RHR Suction line downstream of the isolation valves from the RCS. Each flowpath has two motor operated RHR isolation valves powered from separate Class 1E sources. Therefore, with one flowpath closed, and the valves in the other path open with the motor breakers locked in the off position, no single active failure could be assumed that would prevent the LTOP system from performing its intended function.

(2) The licensee states that regulatory analysis for the resolution of GI 94 provided in NUREG-1326, assumed an LTOP usage time that was unacceptably conservative for Kewaunee. A typical average time for Kewaunee in shutdown mode is 45-days per year. This includes a typical 41-day refueling outage during which the RCS is typically either vented through an adequate flowpath or above the LTOP enable temperature for approximately 37 days. An assumption of 4 days reliance on LTOP during a typical refueling outage, and 50% of other shutdown days per year, results in a more accurate representation of Kewaunee reliance on LTOP of 6 days per reactor year.

(3) The licensee cites their existing TS 3.1.a.2.B for the RHR system as further justification for their proposed LTOP TS. TS 3.1.a.2.B requires two RHR trains to be operable whenever the average RCS temperature is less than or equal to 200 degrees F. If one RHR train is inoperable, corrective action shall be taken immediately to restore it to operable status.

(4) The licensee states that depressurization and venting require approximately 24 hours to complete at Kewaunee. Therefore, the GL proposed TS requirement of a 24-hour AOT or complete depressurization and venting within a total of 32 hours actually only allows 8 hours before depressurization must start. This does not provide adequate time to attempt to diagnose and repair potential problems with an LTOP flowpath or provide time for a crud burst and cleanup evolution before depressurization and cooldown to reduce personnel dose rates during the outage.

In their justification, the licensee provides a hypothetical example of attempting to repair a problem with valve RHR 1A before starting the depressurization. In the licensee's example it would take up to 76 hours to attempt to repair and test the valve, perform the crud burst cleanup evolution, and depressurize and vent the plant.

The staff has reviewed the licensee's proposed LTOP TS and their justification for a 5-day AOT instead of the GL recommended 24 hour AOT. While the proposed TS does not adopt the GL recommendations completely, it does meet the GL intent of improving the availability of the LTOP system, and is reasonable based on the licensee's justification. The proposed TS represents a significant improvement over the current Kewaunee TS which do not even address the LTOP system. Based on the above, the staff finds the licensee's proposed TS acceptable.

2.3 Evaluation of LTOP Design Curve

On March 22, 1993, the licensee received a NSAL from Westinghouse which identified a potential issue regarding the establishment of a nonconservative LTOP setpoint. The licensee's engineering assessment concluded that neither the Westinghouse methodology used for calculating the LTOP setpoints, nor the Kewaunee plant setpoint calculation had properly accounted for flow induced differential pressures across the reactor core, or applicable piping losses. As a result, the licensee's engineering staff completed an evaluation of the LTOP system design requirements, licensing basis, and setpoint methodology.

Calculations of revised Appendix G pressure and temperature limits for isothermal LTOP events indicated the existing setpoint would adequately protect the reactor coolant pressure boundary for two additional annual operating cycles. The licensee concluded that the existing LTOP setpoint could be justified for two cycles of operation under a 10 CFR 50.59 safety evaluation.

In subsequent discussions with the NRC staff, including an April 22, 1993, meeting, the licensee was requested to include an LTOP design curve with their follow up response to GL 90-06.

Based on the above, the licensee has proposed the following modifications to the limiting conditions for operation for reactor coolant temperature and pressure:

1. Figure TS 3.1-4 will be added to define 10 CFR 50 Appendix G pressure and temperature limitations for LTOP evaluation through the end of operating cycle (EOC) 20. This is equivalent to 17.14 effective full power years.

2. Figure TS 3.1-4 will be used to evaluate LTOP events when RCS temperature is less than or equal to the enable temperature of 338 degrees F.

3. Existing TS 3.1.b.1 will be modified to reflect the incorporation and use of Figure TS 3.1-4.

The text of the existing TS 3.1.b.1 has been modified to reflect the incorporation and use of Figure TS 3.1-4. The figure is to be used for evaluation of reactor vessel integrity should an LTOP event occur. Compliance with the pressure and temperature limits of Figure TS 3.1-4 provides assurance that adequate reactor vessel fracture toughness properties are maintained, and assure the integrity of the RCS pressure boundary.

As proposed, Figure TS 3.1-4 expires at EOC 20, which is equivalent to 17.14 EFPY of operation. EOC 20 is currently scheduled to occur about April 1995. Without further action by the licensee, LTOP events occurring after EOC 20 would be evaluated to the zero cooldown rate curve of Figure TS 3.1-2.

The use of Figures TS 3.1-1 and 3.1-2 is not changed. These curves are to be complied with during reactor coolant system heatup and cooldown evolutions.

The proposed TS changes reflect 10 CFR 50 Appendix G pressure and temperature limitations for a limited period of neutron irradiation (i.e., EOC 20). The use of predicted fluence values through EOC 20 was appropriately considered within the calculations in accordance with standard industry methodology previously docketed under WCAP-13227, "Evaluation of Pressurized Thermal Shock for Kewaunee," March 1992. Revised flux values were used for Cycles 16, 17, and 18 based on actual core reload designs. All other flux values were taken from WCAP-12333, "Kewaunee Reactor Vessel Life Attainment Plan," August 1989.

The pressure and temperature limits were calculated in accordance with the approved regulatory methodology of Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, Position C.2. The calculation of pressure temperature limits in accordance with approved regulatory methods provides assurance that reactor pressure vessel fracture toughness requirements are met and the integrity of the RCS pressure boundary is maintained.

Compliance with RG 1.99, Revision 2, methods is an acceptable approach for evaluating predictions of radiation embrittlement needed to implement Appendices G and H to 10 CFR 50. The use of Regulatory Position C.2 meets previously established criteria for protection of the health and safety of the public.

The staff has reviewed the licensee's proposal to add the LTOP design curve, Figure TS 3.1-4, to the TS, and finds it acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes 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 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there have been no public comments on such finding (58 FR 39062). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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

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