Octobert 14, 1992

Docket No. 50-305

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Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Schrock:

SUBJECT: AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. M83536)

The Commission has issued the enclosed Amendment No. 96 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated May 27, 1992 and supplemented July 9, 1992.

The amendment revises the Pressure/Temperature limits in Technical Specification (TS) Section 3.1.b, "Heatup and Cooldown Limit Curves for Normal Operation;" and, TS Figures 3.1-1 and 3.1-2 are replaced with new heatup and cooldown limit curves. Also, administrative changes are made to correct typographical errors and format inconsistencies.

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

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

Enclosures:
1. Amendment No. 96 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

PD3-3:LA	INT:PD3-3:	PD3-3:PM	PD3-3:PD	OGC-OWF	
PKreutzer	CSkinner/bj CES	AHansen	JHannon	w/o MZOBLER	
9/12/92	9/18/92	10/13/92 9/29/92	9/29/92	9/30/92	

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DOCUMENT NAME: KEW83536.AMD

JBH-bf 10-14-92

9210230097 921014
PDR ADOCK 05000305
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Mr. C. A. Schrock
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

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

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
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 27, 1992, and supplemented July 9, 1992, 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.96 , 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

for Jon B. Hopkins

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: Octobert 14, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 96

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 vii	TS vii
Figure TS 3.1-1	Figure TS 3.1-1
Figure TS 3.1-2	Figure TS 3.1-2
TS 3.1-1 through TS 3.1-17	TS 3.1-1 through TS 3.1-20

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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.

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.

B. TWO residual heat removal trains shall be operable whenever the average reactor coolant temperature is $\leq$ 200°F and irradiated fuel is in the reactor, except when in the REFUELING mode one train may be inoperable for maintenance.

1. Each residual heat removal train shall be comprised of:

a) ONE operable residual heat removal pump

b) ONE operable residual heat removal heat exchanger

c) An operable flow path consisting of all valves and piping associated with the above train of components and required to remove decay heat from the core during normal shutdown situations. This flow path shall be capable of taking suction from the appropriate Reactor Coolant System hot leg and returning to the Reactor Coolant System.

2. If one residual heat removal train is inoperable, corrective action shall be taken immediately to return it to the operable status.

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be operable whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power-operated relief valves and the safety valve on the discharge of the charging pump are set for test pressure plus 35 psi to protect the system.

B. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressure Isolation Valves

   A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device 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. If a pressurizer PORV is inoperable, the PORV shall be restored to an operable condition within one hour or the associated block valve shall be closed and maintained closed by administrative procedures to prevent inadvertent opening.

      2. If a PORV block valve is inoperable, the block valve shall be restored to an operable condition within one hour or the block valve shall be closed with power removed from the valve; otherwise the unit shall be placed in the HOT SHUTDOWN condition using normal operating procedures.

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.

---

[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.

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 and TS 3.1-2 for the service period up to 20 equivalent 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.

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.

c. Maximum Coolant Activity

The total specific activity of the reactor coolant due to nuclides with half-lives of more than 30 minutes, excluding tritium, shall not exceed

$$A = \frac{91}{\overline{E}} \qquad \frac{\mu Ci}{cc}$$

whenever the reactor is critical or the average temperature is > 500°F ($\overline{E}$ is the average sum of the beta and gamma energies in Mev per disintegration).

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.[2]

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.

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.

[2] USAR Section 7.2.2

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.[(3)]

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 relief valve become inoperable.

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[(4)] 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.

[(3)] Order for Modification of License dated 4/20/81

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

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.[5]

<u>Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)</u>

<u>Fracture Toughness Properties - (TS 3.1.b.1)</u>

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[6], and the calculation methods of Footnote[7]. 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[8].

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(5) Letter from E. R. Mathews to S. A. Varga dated 5/21/82

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

(7) Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-86.

(8) 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)

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.

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[9][10].

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[11], weld metal Charpy test specimens from Capsule R indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (235°F).

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[9]"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.

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

[11]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.

The results of Irradiation Capsules V, R, and P analyses are presented in WCAP 8908[12], WCAP 9878, and WCAP-12020[13], 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.

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.

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[14] 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.[14]

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[12] 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.

[13] 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.

[14] USAR Section 14.2.4

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.

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.

Leakage of Reactor Coolant (TS 3.1.d)[15]

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 $X/Q = 2.0 \times 10^{-6}$ 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 and initiate closure of the vent line from the surge tank in the Component Cooling System, within less than one minute. In the case of failure of the closure of the vent line and resulting continuous discharge to the atmosphere via the component cooling surge tank vent, 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 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.

[15] 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.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

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.[16]

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[16] 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[17]. 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.[18][19]

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

[17] USAR Section 9.2

[18] USAR Table 3.2-1

[19] 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[20] 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.[21]

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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[20] USAR Figure 3.2-9

[21] USAR Table 3.2-1

KEWAUNEE UNIT NO. 1 COOLANT HEATUP LIMITATION CURVES

APPLICABLE FOR PERIODS UP TO 20 EFFECTIVE FULL-POWER YEARS

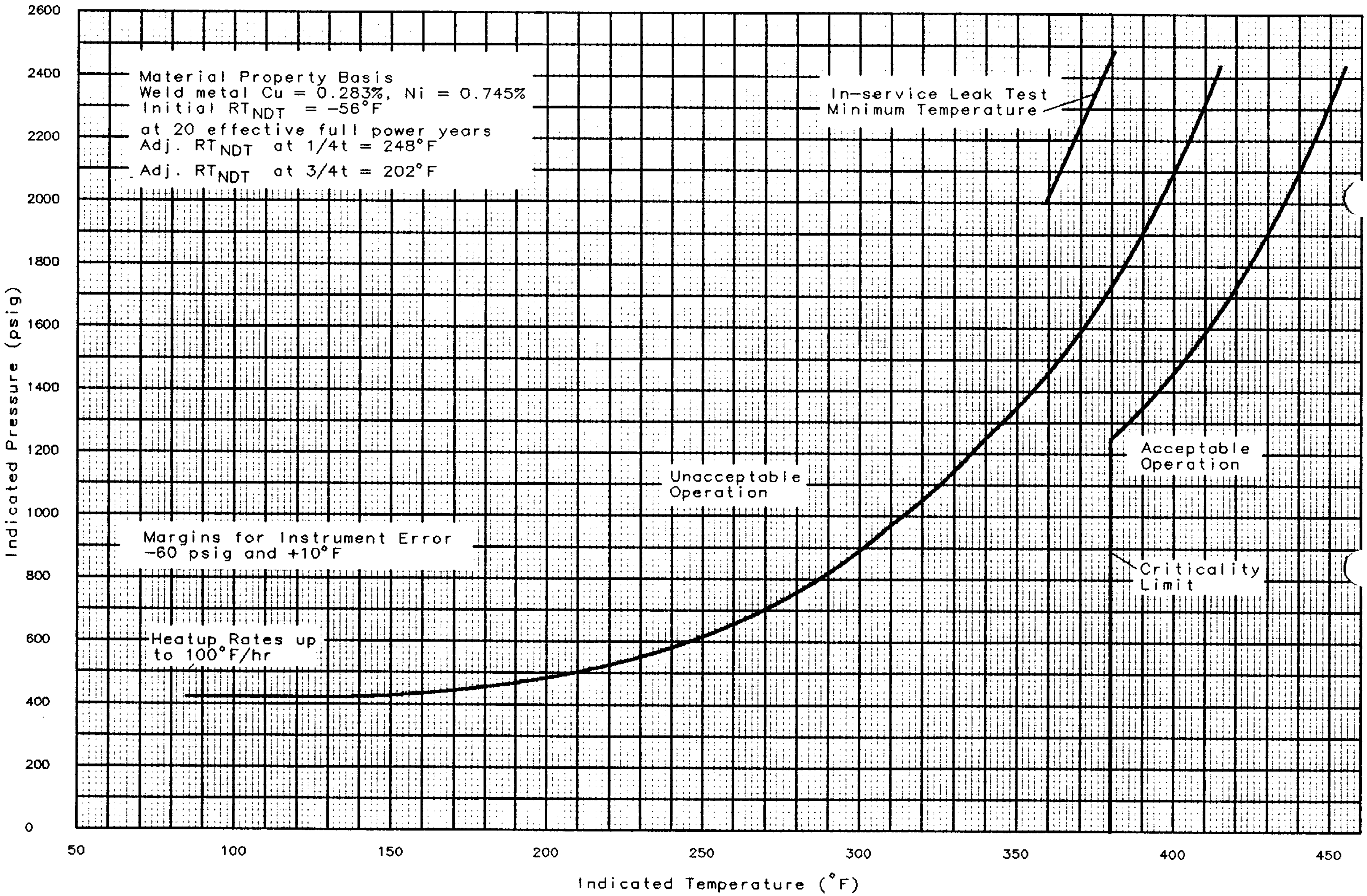

FIGURE TS 3.1-1

Amendment No. 4,70,96

KEWAUNEE UNIT NO. 1 COOLANT COOLDOWN LIMITATIONS

APPLICABLE FOR PERIODS UP TO 20 EFFECTIVE FULL-POWER YEARS

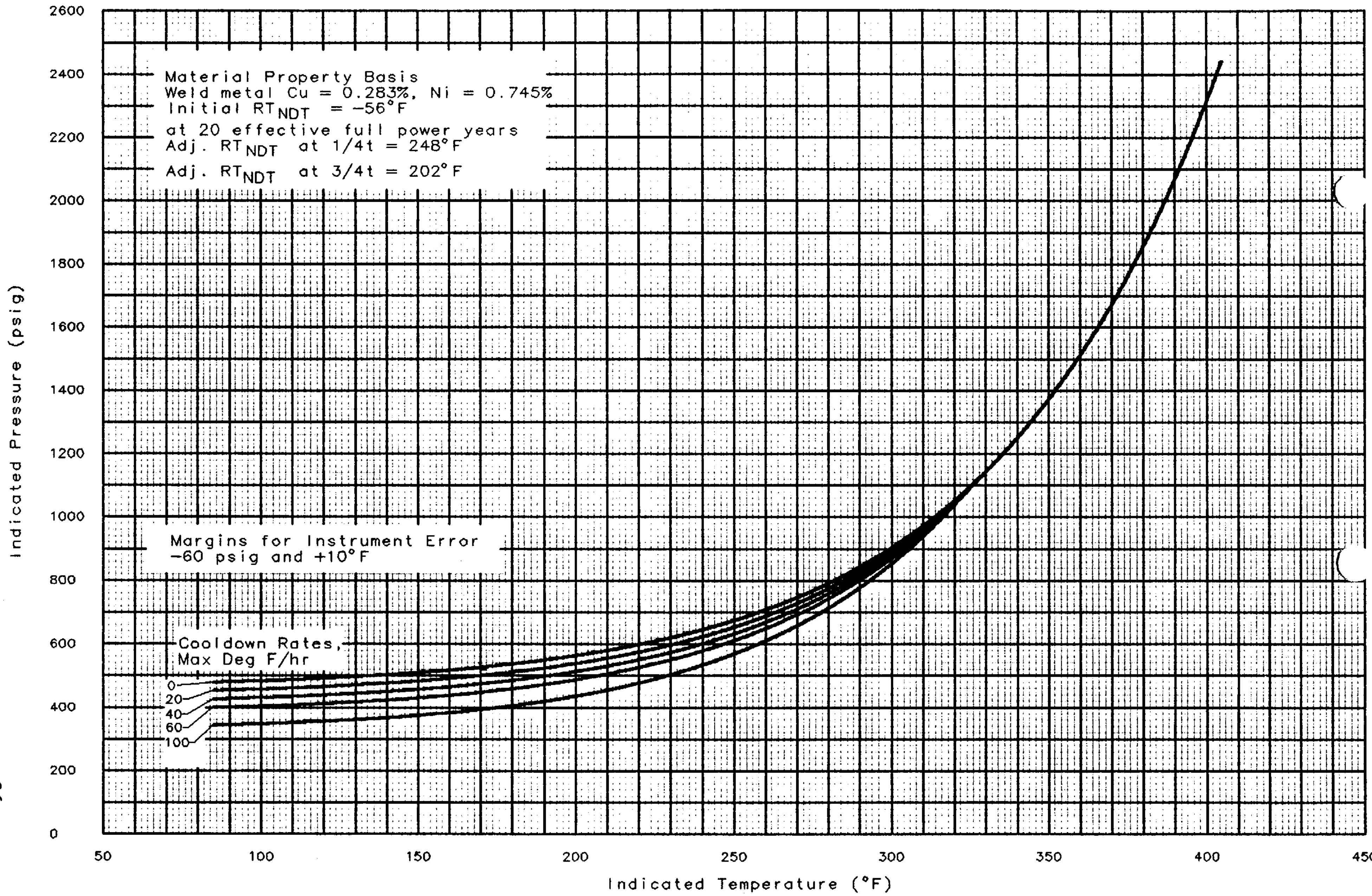

FIGURE TS 3.1-2

Amendment No. 40,70, 96

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 96 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 27, 1992, as supplemented July 9, 1992, the Wisconsin Public Service Corporation (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Kewaunee Nuclear Power Plant Technical Specifications, Section 3.1. The proposed P/T limits were requested for 20 effective full power years (EFPY). The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all nuclear plants. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE).

Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

(1) Analysis

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Kewaunee reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 20-EFPY was the intermediate to lower shell weld with 0.28% copper (Cu), 0.74% nickel (Ni), and an initial RT_{ndt} of -56°F.

For the limiting beltline material, the lower shell weld, the staff calculated the ART to be 203.8°F at 1/4-T (T = reactor vessel beltline thickness) and 162.3°F for 3/4-T at 20 EFPY. The staff used a neutron fluence of 1.59E19 n/cm^2 at 1/4-T and 7.3E18 n/cm^2 at 3/4-T. The ART was determined by the least squares extrapolation method using the Kewaunee surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Rev. 2.

The licensee used Section 1 of RG 1.99, Rev. 2, to calculate an ART of 248°F at 1/4-T and an ART of 202°F at 3/4-T for the same limiting material. Although the licensee is allowed to use the three sets of surveillance data to calculate ART according to the method described in Section 2 of RG 1.99, Rev. 2, the licensee decided to use, instead, the more conservative P/T curves generated by the Section 1 method. Using the Section 1 method and the licensee's figures for chemistry factor and margin, the staff calculated an ART of 247.2°F at 1/4-T and an ART of 201.5°F at 3/4-T. The staff judges that a difference of 0.8°F between the licensee's ART of 248°F and the staff's ART of 247.2°F is acceptable. Substituting the ART of 248°F into the equations in SRP Section 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50. In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that, when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 60°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end-of-life be above 50 ft-lb. The material with the highest Cu content and the lowest initial transverse USE is the intermediate-to-lower shell weld with 126 ft-lb. Using RG 1.99, Rev. 2, the staff calculated that the end-of-life USE will be 56.9 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

The licensee has removed three surveillance capsules from Kewaunee. The results from capsules V, R, and P were published in Westinghouse Reports WCAP-9808, WCAP-9878, and WCAP-12020, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

(2) Conclusions

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 20-EFPY because the proposed limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy Generic Letter 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Therefore, the staff finds the above TS change to be acceptable.

(3) Proposed Technical Specification Changes

Figures TS 3.1-1 and TS 3.1-2 are being revised to reflect the new heatup and cooldown curves discussed in the evaluation, and are acceptable. Section TS 3.1 and the bases for Section TS 3.1 are also being modified to reflect these revisions. Specifically, the service period for the heatup and cooldown curves on page TS 3.1-5 has been changed to 20 EFPY. In addition, the bases for TS 3.1.b.1, Fracture Toughness Properties, on pages TS 3.1-12 through TS 3.1-15 are being revised. The 1986 Edition of the ASME Boiler and Pressure Vessel Code is now being referenced, the referenced revision date of ASTM Standard E 262 is being changed to 1986, and Footnote 8 is being updated to reference the appropriate report. Furthermore, the third paragraph on page TS 3.1-13 is being modified to state that the allowable vessel pressure is obtained by taking the lesser of three values from the curves under consideration. On page TS 3.1-14, a reference to an outdated NRC Standard Review Plan is being deleted, Footnote 9 is being added, and Footnote 10 is being revised to reflect the current bases for fracture toughness properties. Finally, on page TS 3.1-15, a reference to WCAP-12020 is being added and the expiration date of the heatup and cooldown curves is being changed to 20 EFPYs. All of these changes updated the TS to reflect the new curves and, in some cases, provided additional clarification. Since these changes are consistent with the evaluation presented above, they are acceptable.

Administrative changes are being made to the Table of Contents (pages TS i and vii), Section TS 3.1, and the bases for Section TS 3.1. These changes include format modifications, correction of typographical errors, and other clarifications consistent with the staff evaluation presented above. Specifically, the format for Section TS 3.1 on page TS i is being changed, and the titles of Figures TS 3.1-1 and TS 3.1-2 on page TS vii and on the figures themselves are being modified to reflect the change to 20 EFPYs. In addition, format changes are being included on each page of Section TS 3.1, including the bases. The bases for all subsections are being located together at the end of Section TS 3.1. All of these changes are administrative, and are consistent with the evaluation presented above. Therefore, these changes are acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to requirements 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 surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 34591). 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.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (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.

Principal Contributor: Simon Sheng

Date: Octobert 14, 1992