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April 1, 1999

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
P.O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-43, KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA2284)

Dear Mr. Marchi:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 144 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 November 18, 1998, as supplemented by letters dated March 1, 1999, and March 9, 1999.

The amendment revises the pressure/temperature (P/T) limits and the low-temperature overpressure protection (LTOP) requirements in the facility technical specifications.

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,

Brenda Mozafari for:

William O. Long, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 144to
License No. DPR-43
2. Safety Evaluation

DFOI/1

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DOCUMENT NAME: KEWAUNEE\PA157.AMD *See Previous Concurrence

| OFFICE | PD31:PM | E | PD32:LA | E | EMCB:BC | OGC | | PD31:PD | |
|---|---|---|---|---|---|---|---|---|---|
| NAME | WLong* BM for | | EBarnhill EB | | ESullivan* | RBachman* | | GDick* | |
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| DATE | 03/16/99 4/1/99 | | 03/16/99 4/1/99 | | 03/10/99 | 03/18 /99 | | 03/31/99 | |

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

April 1, 1999

Mr. M. L. Marchi
Site Vice President-Kewaunee Plant
Wisconsin Public Service Corporation
P.O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-43, KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA2284)

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Sincerely,

Brenda Mozafari for

William O. Long, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 144 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

cc:

Foley & Lardner
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Chairman
Town of Carlton
Route 1
Kewaunee, WI 54216

Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, WI 54216

Attorney General
114 East, State Capitol
Madison, WI 53702

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
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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. 144
License No. DPR-43

1. The U.S. 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 November 18, 1998, as supplemented by letters dated March 1, 1999, and March 9, 1999, 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. 144 , 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

Tae Kim

T.K. Kim, Acting Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 1, 1999

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 144</u>

<u>FACILITY OPERATING LICENSE NO. DPR-43</u>

<u>DOCKET NO. 50-305</u>

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

| REMOVE | INSERT |
|---|---|
| TS vi | TS vi |
| TS 3.1-1 | TS 3.1-1 |
| TS 3.1-6 | TS 3.1-6 |
| TS 3.1-7 | TS 3.1-7 |
| TS B3.1-1 | TS B3.1-1 |
| TS B3.1-4 | TS B3.1-4 |
| TS B3.1-5 | TS B3.1-5 |
| TS B3.1-6 | TS B3.1-6 |
| TS B3.1-7 | TS B3.1-7 |
| TS B3.1-8 | TS B3.1-8 |
| TS B3.1-9 | TS B3.1-9 |
| TS B3.1-10 | TS B3.1-10 |
| TS B3-1-11 | TS B3.1-11 |
| TS B3.1-12 | TS B3.1-12 |
| TS B3.1-13 | TS B3.1-13 |
| TS B3.1-14 | TS B3.1-14 |
| TS B3.1-15 | TS B3.1-15 |
| FIGURE TS 3.1-1 | FIGURE TS 3.1-1 |
| FIGURE TS 3.1-2 | FIGURE TS 3.1-2 |
| FIGURE TS 3.1-4 | |

LIST OF FIGURES



Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

## 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 ≤ 200°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.

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. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33[1] 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. The isothermal curve in Figure TS 3.1-2 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 200°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.

<u>Note</u>:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

4. The overpressure protection system for low temperature operation shall be operable whenever one or more of the RCS cold leg temperatures are ≤ 200°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.

## 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 ≤ 200°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.

### 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.

---

[1] USAR Section 7.22

## 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.

---

[5]Section III and XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Protection Against Non-ductile Failure."

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

[7]WCAP-14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," T. Laubham and C. Kim, September 1998.

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. The heatup and cooldown limit curves have been developed by combining the most conservative pressure temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves. 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 for each of the limiting materials. At any given temperature, the allowable pressure is taken to be the lesser of the 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 including the pressure difference between the gage and beltline weld.

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 for each of the limiting materials. 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 $\Delta T$ 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 and limited application to ASME Boiler and Pressure Vessel Code Case N-588 to the circumferential beltline weld. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan[8] and Footnote.[9]

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 14279, Revision 1,[10] weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in $RT_{NDT}$ (250°F).

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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]1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

[10]C. Kim, et al., "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," WCAP-14279, Revision 1, September 1998.

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,[11] WCAP 9878,[12] WCAP-12020,[13] WCAP-14279,[14] and WCAP-14279, Revision 1[10] 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 33[1] effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33[1] effective full-power years of fluence (through the end of operating cycle 33[1]). 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.

Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

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(11)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.

(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, 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)E. Terek, et al., "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

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 as modified by ASME Boiler and Pressure Vessel Code Case N-514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33[1] effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 200°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 3.1-2.

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.

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 (≥ 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)

The limit on gross specific activity 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.

Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

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[15] 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.[15]

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:

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

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)

$\frac{X}{Q}$ = $2.9 \times 10^{-4}$ 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.

---

(15) USAR Section 14.2.4

Reactor coolant specific activity is further limited to $\leq$ 0.20 $\mu$Ci/gram DOSE EQUIVALENT I-131 to ensure that off-site thyroid dose does not exceed 10 CFR 100 guidelines and that the control room thyroid dose does not exceed GDC-19. To ensure the allowable doses are not exceeded, an evaluation was performed to determine the maximum allowable primary to secondary leak rate which could exist during a steam line break event. This analysis is described in the Basis for TS 3.4.d on secondary activity limits.

The action statement permitting power operation to continue for limited time periods with reactor coolant specific activity > 0.20 $\mu$Ci/gram 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)[16]

TS (TS 3.1.d.1)

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 0.1 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.

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[16]USAR Sections 6.5, 11.2.3, 14.2.4

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.

<u>TS 3.1.d.2</u>

The 150 gpd leakage limit through any one steam generator is specified to ensure tube integrity is maintained in the event of a main steam line break or under loss-of-coolant accident conditions. This reduced operational leakage rate is applicable in conjunction with the tube support plate voltage-based plugging criteria as specified in TS 4.2.b.5.

<u>TS 3.1.d.3</u>

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.

<u>TS 3.1.d.4</u>

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.

<u>TS 3.1.d.5</u>

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.[17]

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.[18] 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.

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[17]USAR Section 4.2

[18]USAR Section 9.2

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the fuel cycle, the moderator temperature coefficient may be calculated to be positive at ≤ 60% RATED POWER. The moderator coefficient 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.[19][20]

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.1 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.[19]

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.

The requirement that the reactor is not to be made critical when the moderator coefficient is > 5.0 pcm/°F 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. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than 5.0 pcm/°F provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient 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[21] and the small integrated Δk/k would limit the magnitude of a power excursion resulting from a reduction in moderator density.

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[19]USAR Table 3.2-1

[20]USAR Figure 3.2-8

[21]USAR Figure 3.2-9

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

Analysis has shown that maintaining the moderator temperature coefficient at criticality ≤ 5.0 pcm/°F will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports operating up to 60% power with a moderator temperature coefficient ≤ 5.0 pcm/°F. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than -8.0 pcm/°F for at least 95% of a cycle's time at HFP to ensure the limitations associated with and Anticipated Transient Without Scram (ATWS) event are not exceeded. NRC approved methods(22)(23) will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than -8.0 pcm/°F, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.

In the event that the limits of TS 3.1.f.3 are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit in TS 3.1.f.3. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

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(22)"NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.

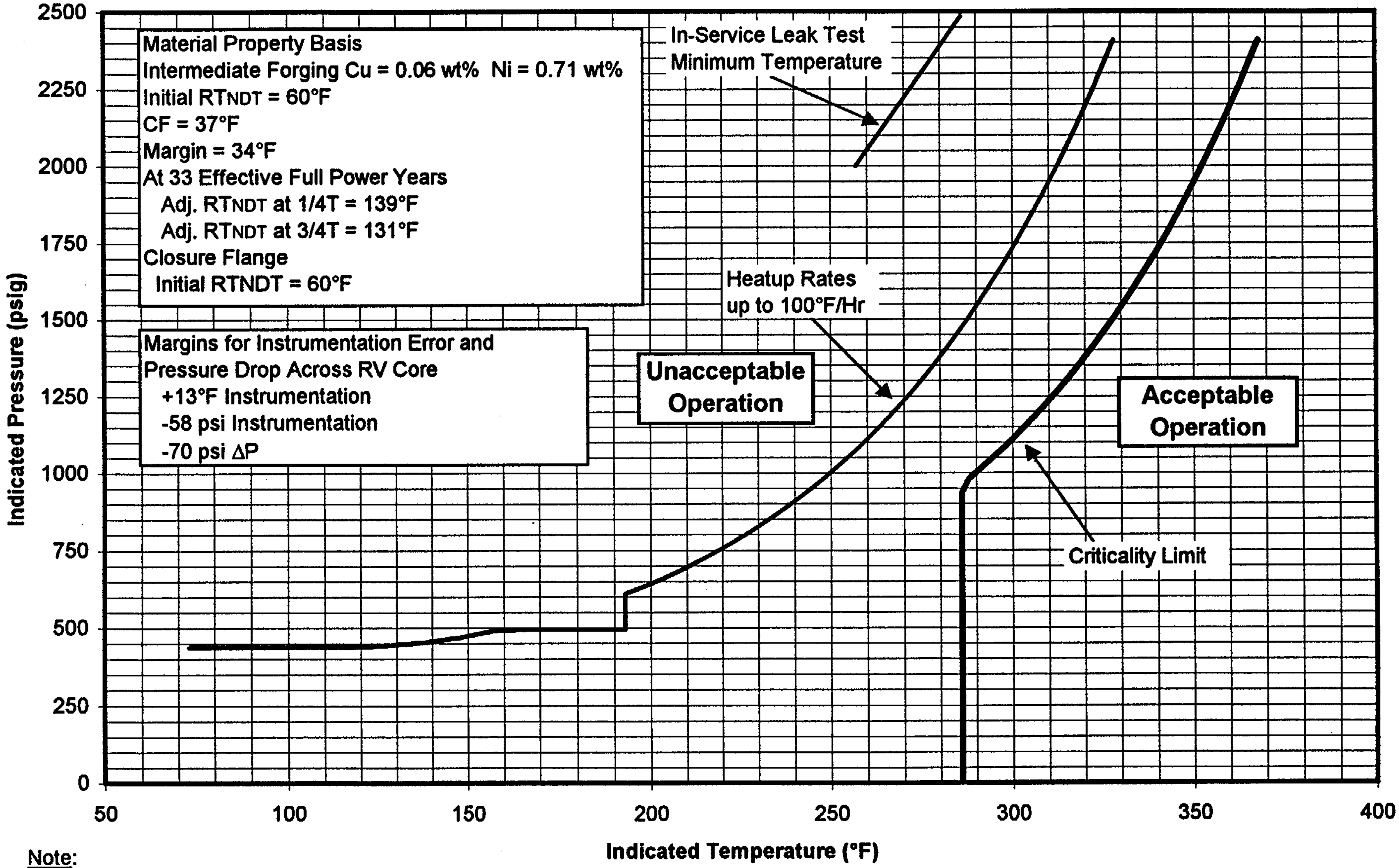
(23)"NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.

If after 24 hours, withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits of TS 3.1.f.3 are not developed, the plant shall be taken to HOT STANDBY until the moderator temperature coefficient is within the limits of TS 3.1.f. The reactor is allowed to return to criticality whenever TS 3.1.f is satisfied.

FIGURE TS 3.1-1

# KEWAUNEE UNIT NO. 1 HEATUP LIMITATION CURVES

## APPLICABLE FOR PERIODS UP TO 33[1] EFFECTIVE FULL-POWER YEARS



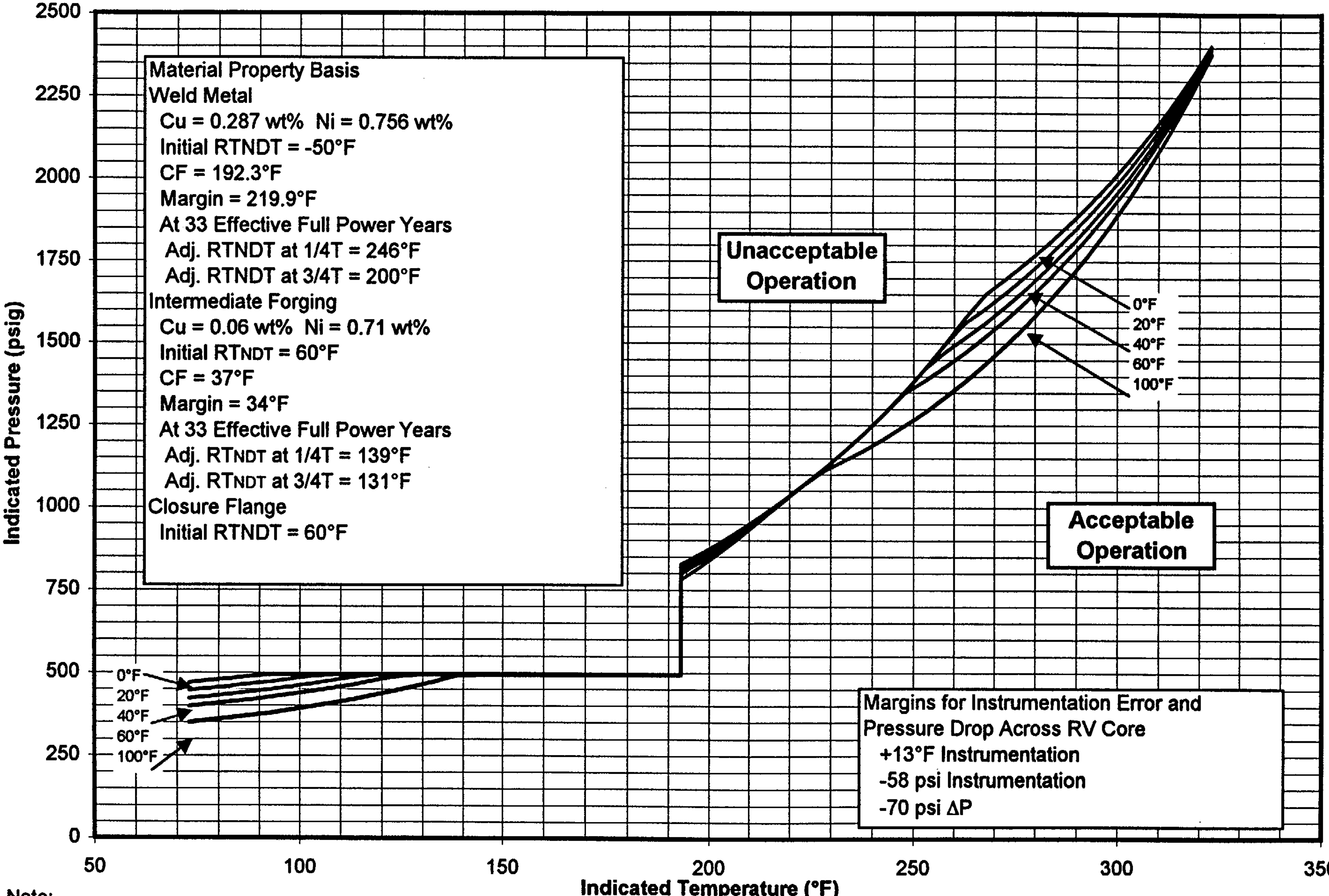

Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

FIGURE TS 3.1-2

# KEWAUNEE UNIT NO. 1 COOLDOWN LIMITATION CURVES

## APPLICABLE FOR PERIODS UP TO 33[1] EFFECTIVE FULL-POWER YEARS




Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

Figure TS 3.1-4
has been deleted

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 144 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 November 18, 1998, as supplemented by letters dated March 1, 1999, and March 9, 1999, the Wisconsin Public Service Corporation (WPSC or the licensee), requested a revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). The proposed amendment would revise TS 3.1 "Reactor Coolant System" specifications, and associated pressure/temperature (P/T) curves defining limitations for heatup and cooldown and low temperature overpressure protection (LTOP).

The March 1, 1999, letter provided clarifying information. The March 9, 1999, letter revised the application based on discussions with the staff, to limit the proposed applicability period of the new P/T curves to 28 effective full power years (EFPY). The initial application proposed that the curves be applicable to 33 EFPY. The March 1, 1999, and March 9, 1999, letters did not expand the scope of the changes beyond that described in the original Federal Register notice and thus did not change the initial no significant hazards consideration determination.

## 2.0 DISCUSSIONS AND EVALUATIONS

P/T limits are required by NRC regulations for the purpose of protecting the reactor pressure vessel (RPV) and primary system from brittle fracture. Ferritic steels exhibit, as an inherent material property, a significant change in fracture mode (brittle to ductile) over a modest temperature band. Brittle fracture is the mode of crack growth which occurs in ferritic steels at relatively low temperatures. Brittle fracture is characterized by rapid, low-energy cleavage of the ferritic steel which could lead to a significant amount of crack growth. If large-scale, through-wall cracking of an RPV were to occur, it could potentially challenge the ability of the RPV and the emergency core cooling systems to maintain an adequate water inventory in the reactor core.

P/T limit curves are incorporated into TSs to identify the acceptable pressure and temperature conditions within the RPV to be maintained by operators during heatup, cooldown, core critical operation, and in-service RPV hydrostatic and leak rate testing.

P/T limits are also used to determine TS operability requirements for LTOP systems that are used to protect the RCS during periods such as shutdown cooling when a mass or heat addition transient could cause a cold overpressure condition.

2.1 Regulatory Requirements

General Design Criterion 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and of gross rupture.

General Design Criterion 31 specifies that sufficient margin be provided to assure that the reactor coolant pressure boundary behaves in a non-brittle manner under the stresses of operating, maintenance, test and accident conditions, with a low probability of rapidly propagating fracture.

10 CFR 50.60 and 50.61 require that licensees demonstrate that the effects of progressive embrittlement by neutron irradiation do not compromise the integrity of the RPV. Two analyses must be performed: (1) an analysis of P/T limits for normal heatup and cooldown operations, and (2) an assessment of the ability to maintain integrity during an emergency shutdown (i.e., pressurized thermal shock or "PTS" event). 10 CFR 50.60 invokes 10 CFR Part 50 Appendices G and H. 10 CFR 50.61 is the "PTS Rule" which requires the PTS assessment.

10 CFR Part 50, Appendix G specifies fracture toughness requirements for ferritic materials of the reactor coolant boundary. It requires that P/T limits for the RCS be at least as conservative as those obtained by the methodology in the 1989 edition of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Code. 10 CFR 50.60 also states that alternatives to the requirements of 10 CFR Part 50, Appendix G may be used when the alternative has been approved via an exemption granted by the NRC.

10 CFR Part 50, Appendix H requires a Reactor Vessel Materials Surveillance Program. The purpose of the materials surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. These changes result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data are used as described in Section IV of Appendix G to Part 50.

The staff's evaluation of compliance with these requirements encompassed four distinct areas of review: (1) an evaluation of new P/T curves that would supersede the current P/T curves that are soon to expire, (2) an evaluation of the licensee's updated PTS assessment, (3) an evaluation of the licensee's fluence measurements, and (4) an evaluation of the licensee's proposed change to the LTOP technical specifications to reflect additional anticipated vessel embrittlement. The staff's evaluations for each of these areas is presented below.

## 2.2 Evaluation of Proposed P/T Limits

The proposed P/T limits reflect additional information about the KNPP RPV materials acquired by the licensee through the KNPP RPV surveillance program; additional RPV weld chemistry data; and the NRC staff's approval of WPSC's use of ASME Code Case N-588 (via letter dated November 25, 1998). The proposed P/T limits would supersede the current P/T limits which are valid through 20 EFPY of operation, with new limits valid through 28 EFPY.

### 2.2.1 Review Criteria

The staff evaluates licensees' proposed P/T limits using the guidance of Generic Letters (GL) 88-11 and 92-01, and their revisions and supplements; Regulatory Guide (RG) 1.99, Rev. 2; Standard Review Plan (SRP) Sections 5.2.2 and 5.3.2; and Branch Technical Position RSB 5-2.

SRP 5.2.2 "Overpressure Protection" provides review criteria for evaluation of the adequacy of overpressure protection for the reactor coolant pressure boundary to meet the requirements of GDC-31. Branch Technical Position RSB-5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," is part of SRP 5.2.2.

SRP 5.3.2 "Pressure-Temperature Limits" provides guidance on calculation of the P/T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section XI of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that have a depth of one-fourth of the section thickness (1/4T) and a length of 1-1/2 the section thickness. This flaw size must be postulated from both the inside and outside surfaces and are designated the "1/4T" and "3/4T" flaws, respectively. Branch Technical Position MTEB 5-2 "Fracture Toughness Requirements" is part of SRP 5.3.2. It summarizes and clarifies fracture toughness requirements and also states that TSs must include: (a) the P/T curves, (b) the basis for their determination, (c) information on intended operating procedures, and, (d) justification of adequacy of margins between expected conditions and limit conditions.

GL 88-11, 92-01, and their revisions and supplements; and RG 1.99, Rev. 2; provide additional guidance regarding the determination of parameters necessary for calculating P/T limits. GL 88-11 requests that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation when calculating the adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature ($IRT_{ndt}$) of the material, the increase in $RT_{ndt}$ caused by neutron irradiation ($\Delta RT_{ndt}$), and a margin (M) to account for uncertainties in the prediction method. The increase in $RT_{ndt}$ is calculated from the product of a chemistry factor (CF) and a fluence factor (FF). The CF is dependent upon the amount of copper and nickel in the vessel material or can be derived from credible surveillance data in accordance with RG 1.99, Revision 2. GL 92-01, its revision and supplements, requests that licensees submit reactor vessel materials data, which the staff will use as a basis in the review of the ART calculations.

Specific to the evaluation of the KNPP P/T limits, one additional regulatory provision applies. By letter dated August 6, 1998, WPSC requested that it be granted an exemption to the requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendix G for the purpose of applying ASME Code Case N-588 for their P/T limits calculations. The NRC approved this request via an exemption dated November 25, 1998. Normally, the methodology in the 1989 Edition of ASME Section XI, Appendix G would require the licensee to postulate a flaw in each RPV material from either the exterior or interior surface (whichever would be most limiting) of the vessel to a depth of 1/4 of the vessel thickness and oriented such that the flaw would extend vertically along the axis of the vessel. This orientation of the flaw results in the largest principal pressure loads being applied to the faces of the flaw which would conservatively analyze for the possibility of flaw propagation. ASME Code Case N-588 modifies this methodology by permitting WPSC to postulate a circumferentially-oriented flaw instead of an axially-oriented flaw when analyzing their circumferential RPV weld. The justification for this allowance is documented in the safety evaluation accompanying the exemption.

2.2.2 Licensee's Analysis

2.2.2.1 Limiting Materials and Their Properties

P/T limits are established by use of the relationship defined in the ASME Code as:

$$K_{Ia} \geq (SF)K_{IP} + K_{IT}$$

where $K_{Ia}$ is the critical material stress intensity factor at a given temperature, $K_{IP}$ is the stress intensity applied at the deepest point of the flaw due to the pressure loading, $K_{IT}$ is the stress intensity applied at the deepest point of the postulated flaw due to the thermal loading, and SF is the safety factor required by the ASME Code for normal operation or vessel hydrostatic testing. The licensee's analyses concluded that the material properties of two of the KNPP RPV beltline materials would define the KNPP P/T limits. This occurs because of the application of ASME Code Case N-588 to the analysis of the circumferential beltline weld. Normally, a single material exhibits the highest ART and serves as the limiting material for all P/T limit calculations throughout the entire range of P/T conditions. However, in the KNPP case, the circumferential weld (manufactured from weld wire heat 1P3571) exhibits the highest ART, but because the circumferentially-oriented flaw is subject to lower pressure stresses, this material was only found to be limiting at the high pressure - high temperature end of the cooldown limits (and for the leak test limit). In the low pressure - low temperature regime of the cooldown limits, intermediate shell forging 122K208VA1 was determined to be limiting because of the postulation of axial flaws. The intermediate shell forging was also limiting for the entire 100 °F/hr heatup limit and the criticality limit.

The information shown in attached Tables A-1, A-2, and A-3 was provided in WCAP-14278, Revision 1 as the licensee's assessment of the material properties of the limiting RPV materials. Surveillance data relevant to the assessment of the KNPP circumferential weld were evaluated in Table A-3. The data were applied to the circumferential RPV weld after accounting for the chemistry difference between the best-estimate value for the RPV weld and the value assigned to the surveillance material by use of the ratio procedure noted in RG 1.99, Revision 2. The use of the values shown in the tables results in ARTs for the circumferential weld at the clad-to-base metal interface, 1/4T location, and 3/4T location of 267 °F, 246 °F, and 200 °F, respectively. Likewise, for the intermediate shell forging the ARTs at the clad-to-base metal interface, 1/4T location, and 3/4T location were 143 °F, 139 °F, and 131 °F, respectively. An item of particular significance was the licensee's use of an $IRT_{NDT}$ value for the circumferential weld of -50 °F and an uncertainty on the $IRT_{NDT}$ value, $\sigma_I$, of 0 °F. The $\sigma_I$ contributes to the calculation of the margin term M as:

$$M = 2\sqrt{(\sigma_I^2 + \sigma_\Delta^2)}$$

where $\sigma_\Delta$ is the uncertainty associated with the shift in the reference temperature, $\Delta RT_{NDT}$. The staff's assessment of the $IRT_{ndt}$ and $\sigma_I$ for the KNPP circumferential weld is addressed in Section 4.1 below.

2.2.2.2 Development of P/T Curves

Based on the material properties data discussed above, the licensee submitted the P/T limits shown in Figures 1 and 2 as the curves to be incorporated into the KNPP TS. It should be noted that the P/T limits submitted by WPSC in Figures 1 and 2 also incorporated instrument uncertainties of +13 °F and -58 psi directly into the curves. For the purposes of its independent analysis, P/T limits were calculated by the staff without the inclusion of the pressure and temperature instrument uncertainties and were compared to the licensee's values on a like basis.

The proposed P/T limits were developed based on the application of the 1989 Edition of Section XI of the ASME Code, Appendix G and ASME Code case N-588. This was consistent with the approach used by the staff. The licensee's detailed methodology for evaluating the behavior of the RPV materials was, however, more detailed than the methodology given in the NRC's SRP Section 5.3.2. For example, when determining the through-wall thermal gradient during heatup and cooldown and in analyzing $K_{IT}$, the stress intensity due to the thermal gradient, the licensee's analysis utilized a finite element approach to arrive at these quantities at multiple points through the vessel wall. In addition, the licensee's analysis permitted the fundamental material properties of the RPV material (e.g. material yield strength, thermal conductivity, etc.) to vary with the wall temperature of the RPV for a more exact representation of the effect of the heatup or cooldown transient on the RPV. Additional information regarding these differences and data regarding the 1/4T and 3/4T temperature and $K_{IT}$ time histories for the heatup and cooldown transients were provided by the licensee via letter dated March 1, 1999.

2.2.3 Staff Confirmatory Analysis

The staff performed an independent analysis of the KNPP P/T limits using the methodology given in SRP 5.3.2, modified to use the more accurate temperature and $K_{IT}$ time histories of the licensee's methodology. This modification was used because the staff's standard methodology for determining $K_{IT}$ (based on the application of Welding Research Council Bulletin 175 correlations) and simplified assumptions for determining the through-wall temperature gradient have been demonstrated to be overly conservative, especially in the low pressure - low temperature regime.

2.2.3.1 Limiting Material Properties

The staff reviewed the information provided by the licensee in Tables A-1 through A-3 to characterize the material properties of the limiting KNPP RPV materials. The information provided was consistent with information submitted previously for the intermediate shell forging and the best-estimate chemistry data supplied regarding RPV weld wire heat 1P3571 were consistent with the most recent information collected regarding Combustion Engineering-fabricated welds in Combustion Engineering Owners Group report CE NPSD-1119, Revision 1. The chemistry and shift data provided for the 1P3571 surveillance material were also acceptable and the data were confirmed to be credible based on the credibility criteria of RG 1.99, Rev. 2. The licensee's application of the ratio procedure from RG 1.99 Rev. 2 for correlating the surveillance weld results to the best-estimate chemistry of the RPV weld was appropriate and the CF of 219.9 °F was determined to be correct. However, the staff does not accept the licensee's use of the -50 °F $IRT_{ndt}$ value and the 0 °F $\sigma_I$ value for the KNPP circumferential weld. These values were submitted by WPSC in a previous P/T limits submittal at which time the staff questioned their applicability given the existence of data from Maine Yankee which produced an $IRT_{ndt}$ for the same weld wire heat of -30 °F. It is the staff's position that, given the variability demonstrated in the determination of the $IRT_{ndt}$ for this material by means of Drop Weight and Charpy Impact testing, generic values ($IRT_{ndt}$ = -56 °F and $\sigma_I$ = 17 °F) characteristic of welds made using Linde 0091, 109, 124, and ARCOS B-5 weld fluxes should be used. The use of these values represents a 6 °F reduction in the $IRT_{ndt}$ and a 16 °F increase in the M term (from 28 °F to 44 °F). This results in staff calculated values for the ARTs at the clad-to-base metal interface, 1/4T, and 3/4T locations being 10 °F higher than those calculated by the licensee: 277 °F, 256 °F, and 210 °F, respectively. As noted above, the staff accepts the ARTs calculated at the clad-to-base metal interface, 1/4T, and 3/4T location for the intermediate shell forging: 143 °F, 139 °F, and 131 °F, respectively.

2.2.3.2 Development of P/T Limits

The staff's analysis generated independent P/T limits based on the use of the ART values cited in Section 2.2.3.1 above, the 1/4T and 3/4T temperature and $K_{IT}$ time-histories submitted by WPSC based on the Westinghouse analysis, a pressure stress/$K_{IP}$ evaluation based on thin shell theory assumptions (per SRP 5.3.2), and the use of the 1989 Edition of ASME Code, Section XI, Appendix G. In addition, when analyzing the circumferential weld, the staff also applied ASME Code Case N-588 for the analysis of the circumferentially-oriented flaw. For the 0, 20, 40, 60, and 100 °F/hr cooldown rates, the staff analyzed the 1/4T flaw for both the intermediate shell forging and the circumferential weld. For the 100 °F/hr heatup rate, the staff analyzed the 1/4T and 3/4T flaws for both the forging and the circumferential weld. This approach is acceptable since the 3/4T location can only be limiting when the thermal stresses are tensile on the outside surface of the vessel, as occurs during heatup.

2.2.4 Findings and Conclusion

Generally, the staff's results confirmed the results submitted by the licensee. For the cooldown curves, in those regions which were controlled by the intermediate shell forging analysis, the licensee's results were consistently more conservative than those determined by the staff. It was observed that this result was primarily due to a more conservative assumption in the licensee's analysis regarding the derivation of $K_{IP}$ from the pressure stresses. This is significant since the low pressure-low temperature end of the curves is controlled by the forging and is the region of greatest significance when protecting the vessel from brittle fracture. In addition, this assures that conservative values are being used by the licensee for establishing or verifying the acceptability of the LTOP system setpoints because they are based on the lower end of the isothermal cooldown curve.

The staff's results did, however, differ from the licensee's for the analysis of the circumferential weld and, as a byproduct of this difference, for the 100 °F/hr heatup curve. As noted previously, the staff's values for the 1/4T and 3/4T ARTs for the circumferential weld were 10 °F more conservative (i.e., greater) than those submitted by the licensee. When this difference was coupled with the slightly less conservative evaluation of $K_{IP}$ mentioned above, the result was that the curves developed by the staff for the circumferential weld were shifted between 5 °F to 7 °F in a direction more conservative than those proposed by the licensee. Again this result only affected the upper end (greater than 1000 psi or greater than 190 °F) end of the cooldown curves, and it resulted in the staff's analysis demonstrating that the 100 °F/hr curve was not completely controlled by the forging but was instead circumferential weld-limited at pressures exceeding 1750 psi or temperatures greater than 270 °F. Differences of the similar size were also observed for the hydrostatic and leak rate testing curve and for the criticality limit.

The staff determined that the effect of this 5 °F to 7 °F differential is to reduce the acceptability of the curves submitted by the licensee is a fluence of 2.85 x $10^{19}$ n/cm$^2$ or about 28 EFPY. The staff advised the licensee of the results of its independent analysis and the licensee subsequently, by letter dated March 9, 1999, revised its application to limit the proposed applicability of the new curves to 28 EFPY. It will be necessary for the licensee to submit a future amendment application for operation beyond 28 EFPY.

2.2.5 Summary - P/T Curves

The staff has determined, based on its independent assessment described above, that the proposed P/T curves, shown in Figures 1 and 2 of this SE, meet the requirements of 10 CFR Part 50, Appendix G to a fluence of 2.85 x $10^{19}$ n/cm$^2$ ($E \geq 1$ MeV) corresponding to 28 EFPY. The application, as modified by the March 9, 1999, supplement is acceptable.

2.3 <u>PTS Evaluation</u>

"PTS event" means an event or transient in a pressurized water reactor (PWR) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. The PTS Rule requires that PWR licensees calculate a projected $RT_{PTS}$ , accepted by the NRC, for each beltline material, for the predicted end-of-life fluence. This assessment must be updated whenever there is a significant change in the projected $RT_{PTS}$.

The circumferential weld seam is the limiting beltline material in the KNPP reactor vessel. The best estimate $RT_{PTS}$ was calculated by the licensee, using an $IRT_{ndt}$ value of -56 °F and a $\sigma_I$

value of 17 °F, to be 277 °F for the EOL fluence value of $3.34 \times 10^{19}$ n/cm$^2$. This value is below the screening criterion limit of 300 °F and is acceptable.

2.4 Fluence Evaluation

2.4.1 Background

The staff reviewed the validity of the fast neutron fluence data utilized in the estimation of the P/T and LTOP curves.

Attachment 3 of reference 1 (WCAP-14279) contains a reevaluation of Kewaunee surveillance capsules V, R, P, and S. In addition, the Maine Yankee surveillance capsule A-35 was also reevaluated. Both the Kewaunee and the Maine Yankee pressure vessels and surveillance welds were fabricated from the same weld wire heat, 1P3571, using lot 3958 of Linde flux 1092.

2.4.2 Methodology

Capsule reevaluation was required to put all of the capsule data on the same updated basis because the first three capsules were analyzed and evaluated well before recent cross section changes were introduced by the ENDF/B-VI data base. The BUGLE-93 library is based on ENDF/B-VI (Ref. 2) and is used with the two dimensional transport code DOT (Ref. 3). The methodology includes a forward transport calculation to establish neutron spectral distributions and an adjoint calculation. The spectral information is used to interpret dosimetry measurements and to estimate the energy integral for E > 1.0 MeV. The results of the adjoint calculations are used in conjunction with source distributions to yield the absolute values of neutron fluxes at points of interest.

Both the forward and the adjoint analyses were carried out using the $S_8$ quadrature and $P_3$ scattering cross section approximation. Both types of calculations were run in (r,θ) geometry and the sources were derived from fuel cycle design reports and used for the pinwise power distribution. The irradiation history (for both the dosimeters and the vessel) was obtained from the "Licensed Operating Reactor Status Summary Report." Up to this point, the proposed methodology complies with staff recommendations as described in the draft regulatory guide DG-1053.

However, in addition the report utilizes the FERRET (Ref. 4) code for a least squares averaging of the dosimetry results. The licensee states that the FERRET code utilizes a priori calculated group fluxes to produce a best estimate fluence value. In the effort to adjust the parameters to fit the measured values FERRET also could adjust the cross sections. The FERRET code has not been approved by the NRC.

2.4.3 Results - Kewaunee

Reevaluation of Capsules V, R, P, and S resulted in lower estimated fluence values by 5 %, 7%, 5% and 3% respectively. The new estimates are within the uncertainty limits of the previous estimates. The dosimeter consistency for the calculated to measured values is good. More important though is the consistency of the high energy dosimeter (Cu-63) to Fe-54 and Ni-58

(E > 1.0 MeV) for which the values are within the uncertainty limits. Finally, let us note that the overall E > 1.0 MeV capsule fluence values are higher than the calculated values by 10%, 7%, and 9% for capsules V, R, and P respectively, and lower by 4% for capsule S.

We find the proposed fluence values acceptable because the proposed values are higher than the calculated values and we consider them conservative.

2.4.4 Results: Maine Yankee

The calculated to measured value ratio is not reported for Maine Yankee; however, the report states that the calculated value is used, which is also the staff's recommendation, thus the staff finds the proposed values acceptable.

2.4.5 Fluence Evaluation - Summary and Conclusions

The staff reviewed the information submitted by Wisconsin Public Service Corporation regarding revised and updated fluence methodology. The purpose of the reevaluation was to apply consistent and uniform values for a request to extend the applicability of the vessel pressure temperature curves from 20 to 33 EFPYs of operation. Capsule A-35 from the Maine Yankee reactor was also reevaluated (for use of the material properties). The proposed values are acceptable because the methodology utilized is acceptable and/or is conservative. However, due to technical differences between the NRC staff and the licensee concerning the unirradiated material properties, these curves are limited to 28 EFPY of operation

2.4.6 Fluence Evaluation - References

1. Marchi, Mark L , Wisconsin Public Service Corporation,letter to USNRC "Kewaunee Nuclear Power Plant, Proposed Technical Specification Amendment 157 - Revised Heatup and Cooldown Limit Curves," November 18, 1998. Attachment 3: WCAP-14279 Revision 1, "Evaluation of Capsules from the Kewaunee and Capsule A-35 from the Maine Yankee Nuclear Plant Reactor Vessel Radiation Surveillance Program," by C. C. Kim et. al., September, 1998, Attachment 5: WCAP-14280 Revision 1, "Evaluation of Pressurized Thermal Shock for the Kewaunee Reactor Vessel" by E. Terek et. al., September 1998.

2. ORNL RSIC Data Library Collection DLC-175, Bugle-93 Production and Testing of the VITAMIN-B6 Fine-Group and the Bugle-93 Broad Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data.

3. ORNL RSIC Code Package CCC-543, "TORT-DORT Two and Three Dimensional Discrete Ordinates Transport Version 2.7.3," May 1993.

4. HEDL-TME-79-40, "FERRET Data Analysis Code," By F.A. Schmittroth, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

## 2.5 LTOP Evaluation

### 2.5.1 Introduction

Section 5.2.2 of the Standard Review Plan (SRP) specifies that the LTOP system be designed in accordance with the guidance of Reactor System Branch Technical Position (RSB) 5-2. The RSB 5-2 guidance specifies that the LTOP system be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to prevent RCS pressure from exceeding the Appendix G limits of 10 CFR Part 50 while operating at low temperatures. The existing LTOP system for KNPP is provided by the relief valve in the suction line of the residual heat removal (RHR) system. The RHR relief valve discharges to the containment sump when it opens. Administrative controls and plant procedures aid in controlling RCS pressure during low-temperature operations. Normal plant procedures maximize the use of a steam or gas bubble in the pressurizer during operations of low-pressure and low-temperature conditions. In the current TS, the pressure setpoint for the LTOP system to open the RHR relief valve is 500 psig and the enabling temperature of the LTOP system is 355 °F. These LTOP setpoints were developed to protect the pressure/temperature (P/T) limits established in the current TS Figure 3.1-4.

As a result of the new P/T limits, the licensee proposed to change the enabling temperature in TS 3.1 for the LTOP system from 355 °F to 200 °F. In the proposed TS, the licensee kept the pressure setpoint of 500 psig for the RHR relief valve unchanged. In support of the TS changes application, the licensee provided analyses (Refs. 2 through 4) to demonstrate adequacy of the new LTOP setpoints for protection against the P/T limits developed in accordance with the requirements of the Appendix G limits. The staff has reviewed the proposed TS changes and the supporting analysis and prepared the following evaluation.

### 2.5.2 LTOP Evaluation

The proposed TS 3.1.b.4 specifies the new LTOP enabling temperature of 200 °F, which is changed from 355 °F in the current TS. The proposed TS 3.1.b.4.A specifies operability of the RHR suction relief valve for LTOP. The relief valve is required to automatically open for LTOP when the RCS pressure exceeds the RHR relief valve setpoint of 500 psig, which is the same as the RHR valve pressure setpoint in the current TS.

The calculations to determine the LTOP pressure setpoint for the relief RHR valve were documented in references 2 through 4. The setpoint calculations were performed to show adequacy of the minimum opening pressure of the RHR relief valve to prevent the RCS pressure from exceeding the reactor P/T limits calculated in accordance with the requirements of Appendix G of 10 CFR 50. In the calculations, two types of events were analyzed:

(1) mass addition transient caused by a makeup/letdown mismatch
(2) the heat addition transient caused by an inadvertent starting of one inactive RCP

These events were previously identified by the licensee as the limiting mass and energy input events for design of the LTOP system at KNPP. The setpoint calculations assumed that the transients occurred while the pressurizer was in water-solid condition. In the analysis of heat addition events, the temperature of water in the steam generator (SG) secondary side was assumed to be less than or equal to 100 °F above the RCS cold leg temperature. A sensitivity study was performed to assess the effect of the RCS initial temperature on the pressure responses during the transients for a temperature range varying from 70 to 350 °F. The results of the analyses of the mass addition and heat addition transients (pages 31 and 32 of Ref. 3 and Ref. 4) showed that the mass addition transient with the initial RCS temperature of 100 °F is limiting. In the analyses, the pressure setpoint of the RHR relief valve capacity was assumed to be 500 psig. The results of the analysis for the limiting case showed that the RHR relief valve would mitigate the limiting LTOP transient while maintaining the RCS pressure (with the calculated peak pressure of 604 psig) less than the P/T limit of 621 psig.

The RCS P/T limit of 621 psig was obtained from the bounding P/T limits for the KNPP reactor vessel specified in Table 5-6 of reference 2, which were derived by the licensee in accordance with the requirements of the Appendix G with inclusion of uncertainties associated with the instrumentation errors and RHR relief valve setpoint corrections. The low temperature portion of the bounding P/T limits are applicable to the KNPP operation for service periods of up to 33 EFPY. To support the assumption made in the analysis of the heat addition events, the licensee has included a restriction in the current TS 3.1.a.1.c to prevent starting of any RCP when the temperature of water in the SG secondary side is greater than 100 °F above the RCS temperature during low-temperature operating conditions.

Based on its review, the staff finds that (1) the methodology used to support the adequacy of the proposed setpoints for the LTOP system is the same as that previously used for design of the LTOP system at KNPP, (2) the effect of uncertainties associated with the instrumentation errors and RHR relief valve setpoint corrections is appropriately included in the setpoint analysis, (3) the results of the analysis have shown that the calculated peak RCS pressure (604 psig) is within the P/T limits (Table 5-6 of reference 2) established by the licensee in accordance with the Appendix G requirements, and, thus have met the BTP RSB 5-2 guidance. Therefore, the staff concludes that the analysis used by the licensee to determine the setpoints for the LTOP system at KNPP is acceptable. The proposed TS 3.1.b.4 specifies the new LTOP enabling temperature of 200 °F, and the proposed TS 3.1.b.4.A specifies the setpoint pressure of 500 psig for the RHR relief valve to prevent the RCS from overpressurization during low-temperature operations. The staff finds that the LTOP setpoints in the proposed TS appropriately reflect the results of the acceptable setpoint analysis for the LTOP system at KNPP. Therefore, the staff concludes that the proposed changes to TS 3.1 relating to the LTOP setpoints are acceptable.

### 2.5.3 Conclusions from LTOP Evaluation

The staff has reviewed the licensee's proposed TS changes with respect to the new setpoints for the LTOP system applicable to service periods of up to 33 EFPY at KNPP. Based on this review, the staff finds that adequate analyses have shown that the setpoint pressure of 500 psig for the RHR relief valve and the enabling temperature of 200 °F for the LTOP system specified in TS 3.1 provide sufficient margins to ensure that the P/T limits, established in accordance with the Appendix G requirements, will not be exceeded, and thus, have met the RSB 5-2 guidance. Therefore, the staff concludes that the proposed TS 3.1 with respect to the LTOP setpoints (the LTOP enabling temperature of 200 °F and the RHR relief valve opening pressure of 500 psig ) is acceptable.

2.5.4 References - LTOP Evaluation

1. Marchi, M. L. (WPPC), letter to NRC, "Proposed Technical Specification Amendment 157 - Revised Heatup and Cooldown Limit Curves," November 18,1998.
2. Attachment 6 to Reference 1, WCAP-14278, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," September 1998.
3. Marchi, M. L. (WPPC), letter to NRC, "Proposed Amendment 139a to the Kewaunee Nuclear Power Plant Technical Specifications," October 25, 1996.
4. Marchi, M. L. (WPPC), letter to NRC, "Proposed Amendment 139a to the Kewaunee Nuclear Power Plant Technical Specifications," November 18, 1996.

## 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 changes a requirement with respect to 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 effluent 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 (63 FR 71978). 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.

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