December 18, 1986

Docket No. 50-305

Mr. D. C. Hintz
Manager, Nuclear Power
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Hintz:

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment is in response to your application dated April 29, 1986.

The amendment revises the heatup and cooldown Technical Specifications (TS). In addition, editorial corrections and minor administrative changes are made to the TS. This action completes our TAC No. 61411.

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's next regular biweekly notice publication in the Federal Register.

Sincerely,

/S/

Morton B. Fairtile, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 70 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

Office:	LA/PAD#1:DPLA	PM/PAD#1:DPLA	OGC	PD/PAD#1
Surname:	PShuttleworth	MFairtile/		GLear
Date:	11/20/86	11/20/86	11/ /86	12/18/86

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PDR ADOCK 05000305
P PDR

Mr. D. C. Hintz
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

cc:
David Baker, Esquire
Foley and Lardner
P. O. Box 2193
Orlando, Florida 32082

Stanley LaCrosse, Chairman
Town of Carlton
Route 1
Kewaunee, Wisconsin 54216

Mr. Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, Wisconsin 54216

Chairman
Public Service Commission of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Attorney General
114 East, State Capitol
Madison, Wisconsin 53702

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #1, Box 999
Kewaunee, Wisconsin 54216

Regional Administrator - Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Robert S. Cullen
Chief Engineer
Wisconsin Public Service Commission
P.O. Box 7854
Madison, Wisconsin 53707

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. 70
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 April 29, 1986 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. 70 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile

Morton B. Fairtile, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 18, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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 the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
viii	viii
3.1-3	3.1-3
3.1-4	3.1-4
3.1-5	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7
3.1-8 (blank)	3.1-8
3.1-9	3.1-9
3.1-10	3.1-10
3.1-11	3.1-11
3.1-11a	------
Figure TS 3.1-1	Figure TS 3.1-1
Figure TS 3.1-2	Figure TS 3.1-2
Figure TS 3.1-3	------
Figure TS 3.1-4	------

LIST OF FIGURES

b. HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

Specification

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 15 equivalent fullpower 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 above 200 psig if the temperature of the steam generator is below 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 greater than 320°F.

Basis

Fracture Toughness Properties

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 [1], and the calculation methods of reference [2]. The post-irradiation 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 Nonmandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and are discussed in detail in Reference (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 is 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 two 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 ıd to produce tensile stresses 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 NRC Regulatory Standard Review Plan Directorate of Licensing, Section 5.3.2 "Pressure-Temperature Limits" 1974 and Reference (1).

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

The results of Irradiati~~~ Capsules V and R analyses are ~esented in WCAP 8908 and WCAP 9878, 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 15 effective fullpower years.

Pressurizer Limits

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.

REFERENCES

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

2. Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-70, 1975 Book of ASTM Standards, Part 45, pp. 756-763.

3. P. K. Nair and E. B. Norris, "Pressure/Temperature Operating Curves and Assessment of RT_{PTS} Concerns for Kewaunee Nuclear Plant," SWRI Project 06-8919, April, 1986.

4. 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.

5. 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.

6. Letter from P. S. VanTeslaar (Westinghouse) to C. W. Giesler (WPS) dated April 30, 1981, transmitting KNPP Heatup and Cooldown curves based on Capsule R results.

c. MAXIMUM COOLANT ACTIVITY

Specification

The total specific activity of the reactor coolant due to nuclides with halflives of more than 30 minutes, excluding tritium, shall not exceed $A = \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ whenever the reactor is critical or the average temperature is greater than 500°F ($\bar{E}$ is the average sum of the beta and gamma energies in Mev per disintegration).

Basis

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 (1) 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 set point 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.(1) 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 [illegible]e because of the short range [illegible] 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:

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

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 less than 0.5 rem at the site boundary.

References:

(1) FSAR Section 14.2.4

d. LEAKAGE OF REACTOR COOLANT

Specification

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.

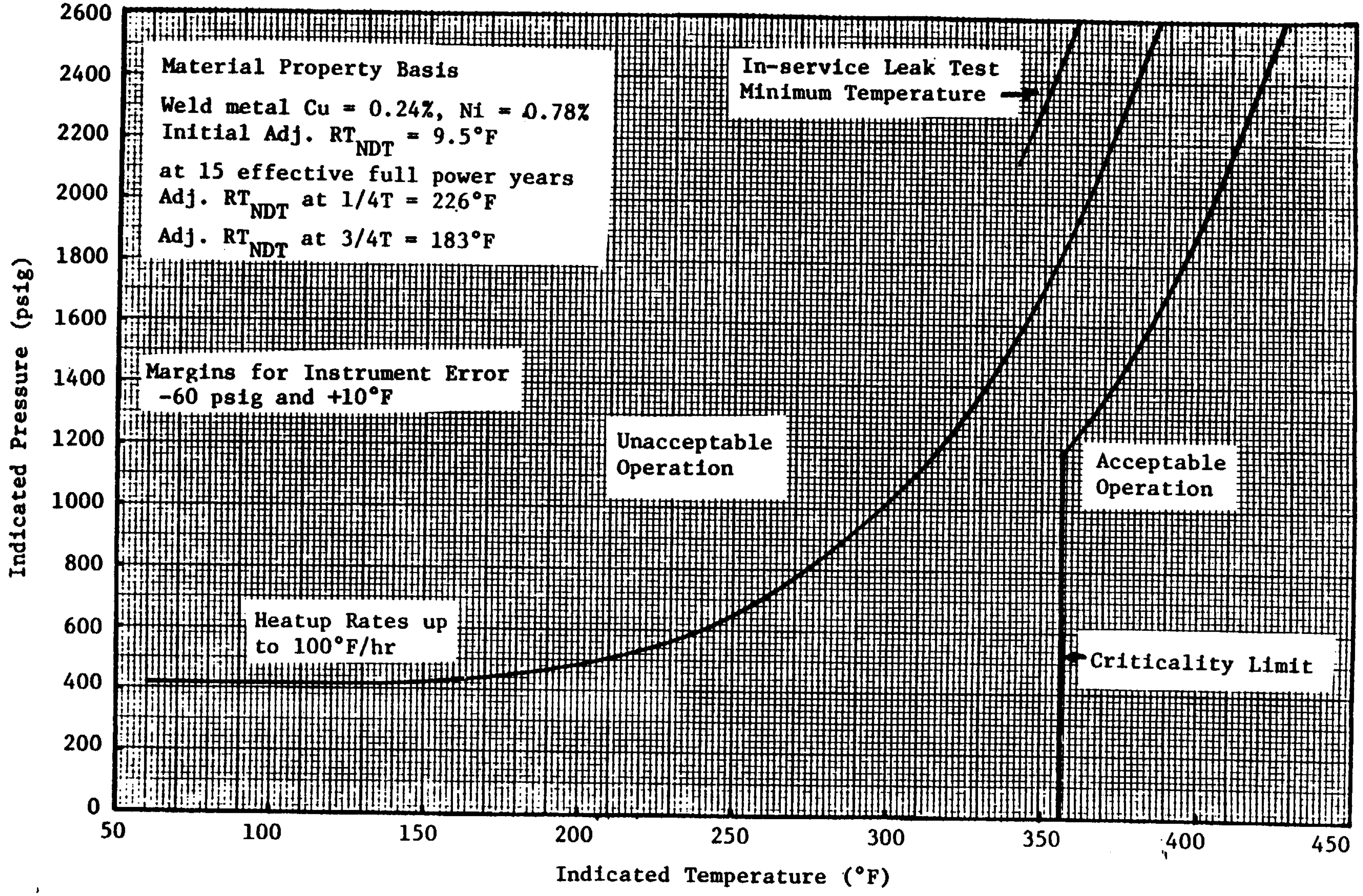
3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed 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 i ritical and above 2% power, tw 'eactor 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.

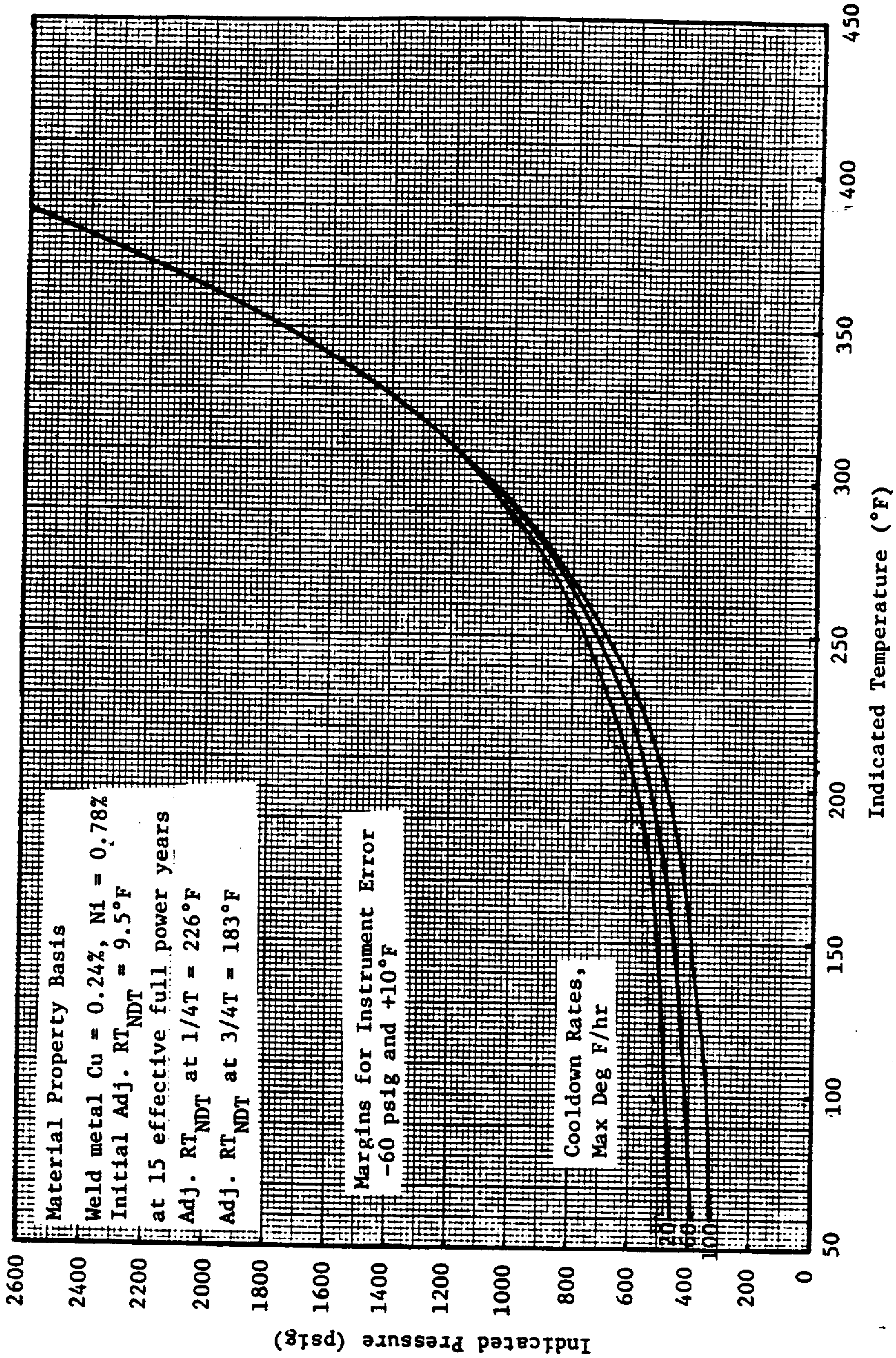
Basis

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



KEWAUNEE UNIT NO. 1 COOLANT HEATUP LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

Figure TS 3.1-1
Amendment No. 4, 70



KEWAUNEE UNIT NO. 1 COOLANT COOLDOWN LIMITATIONS
APPLICABLE FOR PERIODS UP TO 15 EFFECTIVE FULL POWER YEARS

Figure TS 3.1-2

Amendment No. ~~40~~. 70

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 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

Introduction

In a letter from C.W. Giesler to H.R. Denton, dated April 29, 1986, the Wisconsin Public Service Corporation, et al. (the licensees) proposed changes to Kewaunee Nuclear Power Plant (Kewaunee) Technical Specification (T.S.) Section 3.1(b), "Heatup and Cooldown Limit Curves For Normal Operation," and its bases. The licensees requested changes to the pressure temperature limits described in Figures 3.1-1 and 3.1-2 and deletion of Figures 3.1-3 and 3.1-4 from the Technical Specifications. The bases for the changes in the heatup and cooldown limits are the test results from the Kewaunee surveillance program, which are contained in References 1 and 2. References 1 and 2 were submitted for staff review in letters from the licensees dated July 8, 1977, and August 7, 1981, respectively.

Discussion

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR Part 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR Part 50, are dependent upon the initial reference temperature (RT_{NDT}) for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material. The Kewaunee reactor vessel was procured to ASME Code requirements which did specify fracture toughness testing to determine the initial RT_{NDT} for each vessel material. The licensee indicates that the initial RT_{NDT} for the limiting materials in the closure flange and beltline regions of the Kewaunee vessel was estimated using the method recommended by the staff in Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," and Commission Report SECY 82-465, "Pressurized Thermal Shock." These methods result in an initial RT_{NDT} for the limiting beltline base metal and weld metal of 60°F, and -56°F, respectively, and an initial RT_{NDT} for the limiting closure flange material of 60°F.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensees using the empirical relationship documented in Proposed Revision 2 to Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials," which was issued for public comment during February 1986, and was reviewed by NRR, CRGR, and ACRS. The proposed revision has two methods for calculating the increase in RT_{NDT}. One method is to be used when test data from surveillance material is available and the other when test data from surveillance material is unavailable. Both methods are dependent upon the predicted neutron fluence. An evaluation of the licensees' neutron fluence estimates is discussed in Enclosure 1, which was contained in Reference 3.

The beltline material with the highest RT_{NDT} at the end of the effective period of the curves is the limiting beltline material. For Kewaunee the limiting material is the beltline weld, which was fabricated using Linde 1092 flux from heat 3958 and B-4 Mod. weld wire from heat 1P3571. This combination of flux and weld wire was used to fabricate the weld samples that are in the Kewaunee surveillance capsules (References 4 and 5).

In addition to weld metal, the Kewaunee surveillance capsules also contain material from two Kewaunee beltline forgings and a correlation monitor plate. The correlation monitor plate is from an experimental heat of material that was not used in the fabrication of the Kewaunee reactor vessel. The increase in RT_{NDT} for Kewaunee measured from surveillance material is compared in Table I to the value predicted by the regulatory guide. Except for the correlation monitor material, the proposed regulatory guide provides conservative predictions for the increase in RT_{NDT} resulting from neutron irradiation damage.

The proposed regulatory guide permits the use of surveillance material to determine the increase in RT_{NDT} when the five criteria in the discussion section of the guide have been satisfied. Based on the information in References 1 through 5, the Kewaunee surveillance material satisfies these criteria. Using the method of calculating the increase in RT_{NDT} recommended by the proposed guide when surveillance material is available, the RT_{NDT} at the vessel inside surface for the weld metal at 15 effective full power years (EFPY) is 225°F. The Kewaunee heatup and cooldown curves are based upon the increase in RT_{NDT} calculated by the method recommended when surveillance material is not available. Using this method of calculating the increase in RT_{NDT}, the RT_{NDT} at the vessel inside surface for the weld metal at 15 EFPY is 251°F. Since the weld metal value of RT_{NDT} used by the licensees exceeds the value calculated using the surveillance data, the heatup and cooldown curves are based on a conservative value for the increase in RT_{NDT} resulting from neutron irradiation damage.

The information previously contained in T.S. Figures 3.1-3 and 3.1-4 is either presented in the updated FSAR, in regulatory guides issued by the NRC, or in surveillance reports docketed by the licensees. The figures are not plant operation conditions. Hence, they may be deleted from the plant's T.S. The list of figures, TS viii, was changed to reflect the deletions.

Evaluation/Conclusion

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage was calculated using design basis calculated neutron fluences and the Regulatory Guide 1.99, Rev. 2, prediction method. Our conclusion is that the proposed pressure-temperature limits as shown in T.S. Figures 3.1-1 and 3.2-2 meet the safety margins of Appendix G, 10 CFR Part 50 for at least 15 EFPY, and the changes to T.S. Section 3.1(b) and bases are acceptable.

The licensees increased the service period from 10 to 15 equivalent full power years on TS page 3.1-3 to agree with the revised heatup and cooldown curves. This is an editorial correction and we find it acceptable. The new pages 3.1-8 through 3.1-11 were renumbered to reflect the deletion of a previously blank page. We find this change acceptable.

Environmental Consideration

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change to the 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. 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.

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; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Acknowledgement

Principal Contributors:

L. Lois
B. Elliot
M. Fairtile

Dated: December 18, 1986.

References:

(1) Westinghouse Report WCAP 8908, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," Yanichko et.al., January 1977.

(2) Westinghouse Report WCAP 9878, "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program, "Yanichko et.al., March 1981.

(3) Memorandum from C.E. Rossi to M. Fairtile, dated June 16, 1986.

(4) Memorandum from Wisconsin Public Service Corportion to USNRC, "Reactor Vessel Material Surveillance Program," dated February 1, 1978.

(5) Telecon between staff and Wisconsin Public Service Corporation, dated September 25, 1986.

Table I

Comparison of Increase in Reference Temperature (WRT_{NDT})
Measured from Kewaunee Surveillance Material
to the WRT_{NDT} Predicted by Proposed
Regulatory Guide 1.99, Revision 2

Capsule/Surveillance Material	Neutron Fluence ($x10^{19}$ n/cm²)	WRT_{NDT} Measured From Surveillance Material (°F)	WRT_{NDT} (mean Value) Predicted By Reg. Guide 1.99, Rev. 2 (°F)
Capsule V[(a)]			
Forging 122x208VA1	.599	0	27
Forging 123x167VA1	.599	0	27
Weld Metal	.599	175	174
Correlation Monitor	.599	95	87
Capsule R[(b)]			
Forging 122x208VA1	2.07	15	37
Forging 123x167VA1	2.07	20	37
Weld Metal	2.07	235	243
Correlation Monitor	2.07	140	122

(a) Data from Reference (1)
(b) Data from Reference (2)

Enclosure 1

KEWAUNEE NUCLEAR POWER PLANT, FAST NEUTRON FLUENCE ESTIMATES
FOR REVISED HEATUP AND COOLDOWN LIMIT CURVES FOR PROPOSED AMENDMENT 71
(TAC NO. 61411)

By letter dated April 29, 1986 the Wisconsin Public Service Corporation, licensee for the Kewaunee nuclear power plant, submitted information on the material composition of the critical peripheral beltline pressure vessel weld and the current and projected fluence at the inside surface of the vessel (Reference 1). The information is to be used in the revision of the Pressure-Temperature heatup and cooldown curves as required by 10 CFR 50 Appendix G and the estimation of the RT_{PTS} as required by 10 CFR 50.61.

The Reactor Systems Branch reviewed section III entitled "Vessel Fluence Development" in the submitted report. The fluence was determined using the DOT-IV code which has been accepted for fast neutron pressure vessel calculations. The cross sections are from the "Bugle" library which is based on the ENDF/B-IV and is acceptable. In the estimate they used the P_3 scattering approximation and an S_8 angular quadrature both of which are adequate and acceptable. The neutron source was based on measured past cycle power distribution assembly averages, which is acceptable for the present fluence estimate. However, the report does not state any assumptions for future core loadings which determine future neutron sources. The extrapolation is linear which presupposes the same neutron source strength and future cores similar to those in the past. Given the trend to lower leakage cores it is a conservative assumption and, hence, acceptable. The estimate of the azimuthal distribution was based on adequate radial and angular mesh points and is a realistic representation.

We conclude that the current and projected fluence estimates were performed with acceptable methodology, cross sections, approximations and spatial meshings and are therefore, acceptable.

Reference:

1. Letter from C. W. Giesler Wisconsin Public Service Corporation to H. R. Denton Director, NRR, dated April 29, 1986.

DEC 30 1986

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