C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1673 megawatts (thermal).

(2) Technical Specifications

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

(3) Fire Protection

The NMC shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the KNPP Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The NMC may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The NMC shall fully implement and maintain in effect all provisions of the Commission-approved "Kewaunee Nuclear Power Plant Security Manual," Rev. 1, approved by the NRC on December 15, 1989, the "Kewaunee Nuclear Power Plant Security Force Training and Qualification Manual," Rev. 7, approved by the NRC on November 17, 1987, and the "Kewaunee Nuclear Power Plant Security Contingency Plan," Rev. 1, approved by the NRC on September 1, 1983. These manuals include amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

(5) Fuel Burnup

The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

LIST OF FIGURES

Note:

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

j. MODES

MODE	REACTIVITY Δk/k	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	≤ -5%	≤ 140	~0
COLD SHUTDOWN	≤ -1%	≤ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	≥ 540	~0
HOT STANDBY	< 0.25%	$\sim T_{oper}$	< 2
OPERATING	< 0.25%	$\sim T_{oper}$	≥ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,673 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

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.

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 five days or complete depressurization and venting of the RCS through a ≥ 6.4 square inch vent within an additional eight 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 eight hours.

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[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

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).

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.

Note:

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

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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.

(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.

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 ensure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

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

The Low Temperature Overpressure Protection System must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 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 four 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, then the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within five days. If the isolated flowpath cannot be restored within five days, then the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional eight hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, then 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), then the vent path is considered secured in the open position.

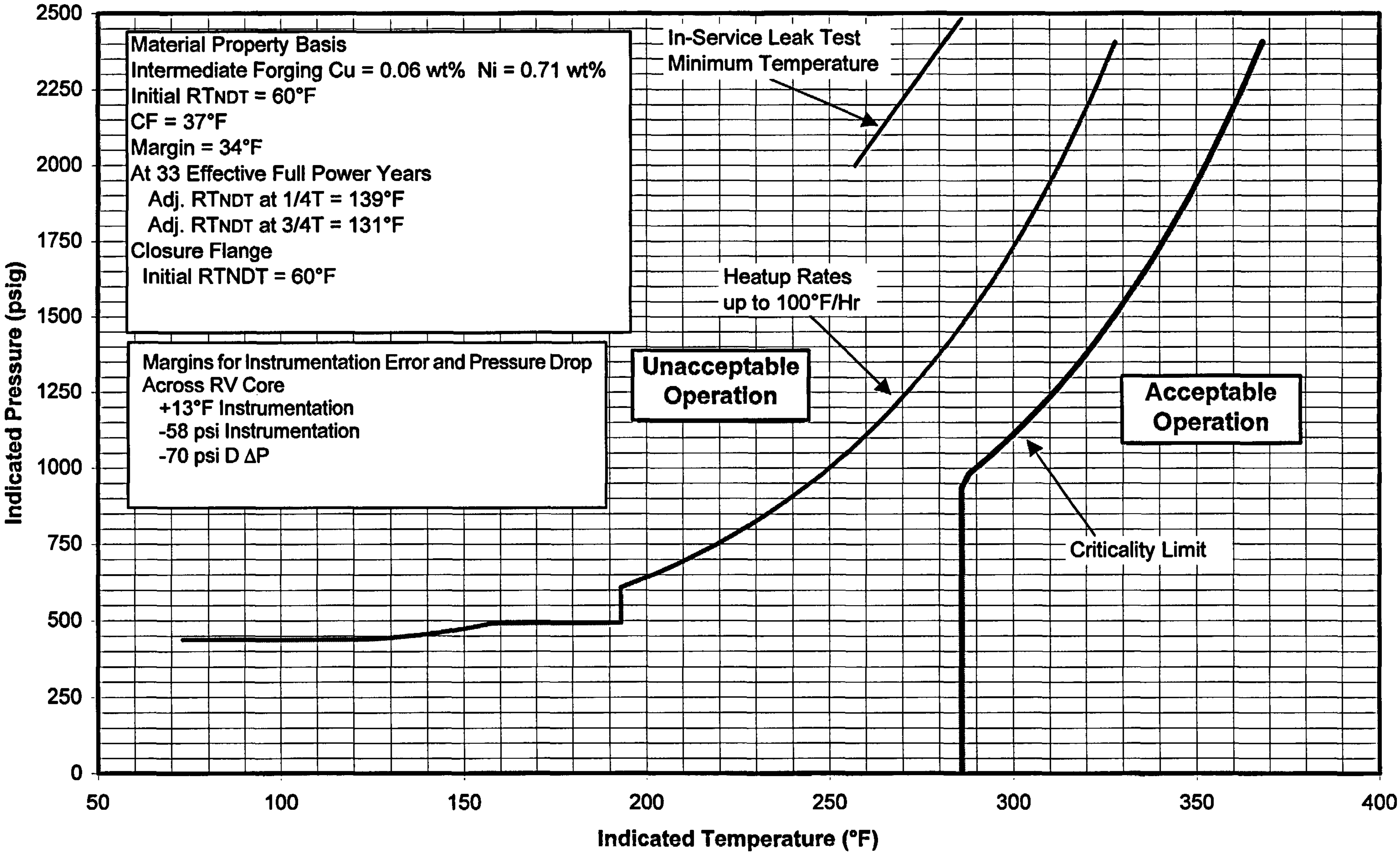
Note

[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

FIGURE TS 3.1-1

KEWAUNEE UNIT NO. 1 HEATUP LIMITATION CURVES

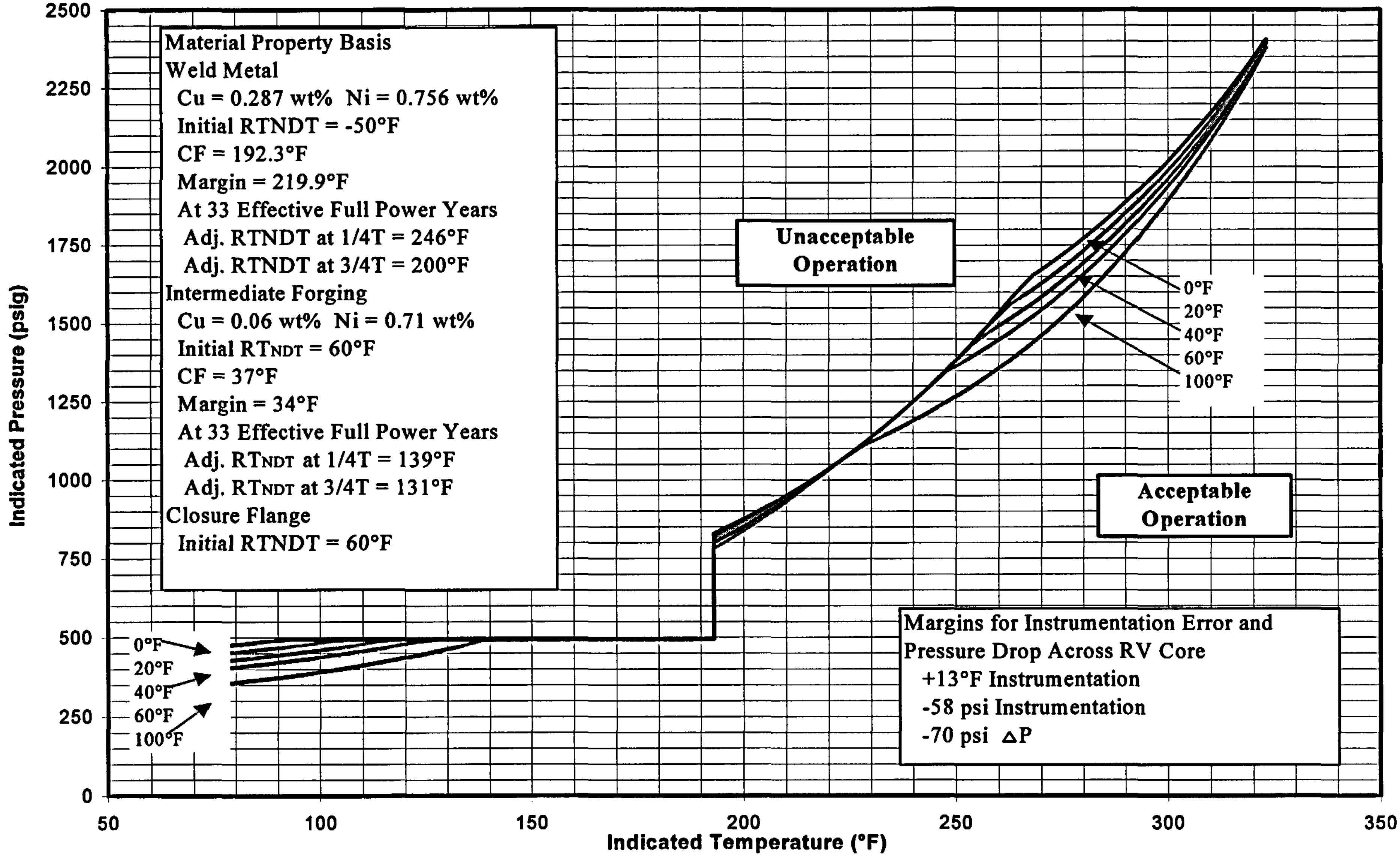
APPLICABLE FOR PERIODS UP TO 33[1] EFFECTIVE FULL-POWER YEARS



NOTE:

[1]The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

FIGURE TS 3.1-2
KEWAUNEE UNIT NO. 1 COOLDOWN LIMITATION CURVES
APPLICABLE FOR PERIODS UP TO 33 [1] EFFECTIVE FULL-POWER YEARS



[1] The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

3. Monthly OPERATING Report

Routine reports of OPERATING statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

4. Core Operating Limits Report (COLR)

A. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

(1)	TS 2.1	Reactor Core Safety Limit
(2)	TS 2.3.a.3.A	Overtemperature ΔT Setpoint
(3)	TS 2.3.a.3.B	Overpower ΔT Setpoint
(4)	TS 3.1.f.3	Moderator Temperature Coefficient (MTC)
(5)	TS 3.8.a.5	Refueling Boron Concentration
(6)	TS 3.10.a	Shutdown Margin
(7)	TS 3.10.b.1.A	$F_Q^N(Z)$ Limits
(8)	TS 3.10.b.1.B	$F_{\Delta H}^N$ Limits
(9)	TS 3.10.b.4	$F_Q^{EQ}(Z)$ Limits
(10)	TS 3.10.b.5.C.i	$F_Q^{EQ}(Z)$ penalty
(11)	TS 3.10.b.9	Axial Flux Difference Target Band
(12)	TS 3.10.b.11.A	Axial Flux Difference Envelope
(13)	TS 3.10.d.1	Shutdown Bank Insertion Limits
(14)	TS 3.10.d.2	Control Bank Insertion Limits
(15)	TS 3.10.k	Core Average Temperature
(16)	TS 3.10.l	Reactor Coolant System Pressure
(17)	TS 3.10.m.1	Reactor Coolant Flow

B. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% of the original rated power is specified in a previously approved method, 100.6% of uprated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow ultrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow System are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.

Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102% of the original rated power should include the condition given above allowing use of 100.6% of uprated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement.

The approved analytical methods are described in the following documents.

(1) Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods For Application To Kewaunee" Report, dated August 21, 1979, report date September 29, 1978

(2) Kewaunee Nuclear Power Plant – Review For Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC No. MB0306) dated September 10, 2001.

(3) S.M. Bajorek, et al., WCAP-12945-P-A (Proprietary), Westinghouse Code Qualification Document for Best-Estimate Loss-of –Coolant Accident Analysis, Volume I, Rev. 2, and Volume II-V, Rev.1, and WCAP-14747 (Non-Proprietary) March 1998.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (W Proprietary).

(10) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(11) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(12) S.I. Dederer, et al., WCAP-14449-P-A, Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Rev. 1 (Proprietary and WCAP-14450-NP-A, Rev. 1 (Non-Proprietary), October 1999.

(13) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

(14) WCAP-11397-P-A, "Revised Thermal Design Procedure, "April 1989.

(15) CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology," Rev. 1, May 2000.

C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.