ENCLOSURE 3

Joseph M. Farley Nuclear Plant Unit 1
Pressure Temperature Limits Report
Technical Specification Changes

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DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System Component body, pipe wall or vessel wall.

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

1.21a The PRESSURE TEMPERATURE LIMITS REPORT (PTLR) is the unit specific document that provides the reactor vessel pressure and temperature (P/T) limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These P/T limits shall be determined for each fluence period or effective full-power years (EFPYs) in accordance with Specification 6.9.1.15. Plant operation within these operating limits is addressed in LCO 3.4.10.1, RCS Pressure/Temperature Limits.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71; State regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limits specified in the PRESSURE TEMPERATURE LIMITS REPORT (PTLR) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.

ACTION:

With any of the above limits specified in the PTLR exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation or inspection to determine the effects of the out-of-limit condition on the fracture toughness of the Reactor Pressure Vessel; determine that the Reactor Pressure Vessel remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits specified in the PTLR at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.10.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10CFR50, Appendix H.

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Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, as required per 10 CFR Part 50, Appendix G.

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 the PRESSURE TEMPERATURE LIMITS REPORT (PTLR).

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

    b) The PTLR defines limits to assure prevention of nonductile 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) These limit lines shall be calculated periodically using methods provided in WCAP-14040-NP-A, Revision 2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.

3) 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.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°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.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{ndt}, at the end of 36 effective full power years (EFPY) of service life. The 36 EFPY service life period is chosen such that the limiting RT_{ndt} at the 1/4T location in the core region is greater than the RT_{ndt} of the limiting unirradiated material. The selection of such a limiting RT_{ndt} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

Values of ΔRT_{ndt} determined in accordance with WCAP-14040-NP-A, Revision 2, may be used until the next results from the material surveillance program, evaluated according to ASTM E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the PTLR. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the surveillance capsule exceeds the calculated ΔRT_{ndt} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-14040-NP-A, Revision 2.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of either RHR relief valve or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F, or (2) the start of all operable charging pumps and their injection into a water solid RCS. In the case of the injection by the charging pumps, the analysis is based on the start of the maximum number of operable charging pumps allowed by the Technical Specifications.

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3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

ANNUAL REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY REPORT

6.9.1.13 This annual report is only required when the results of specific activity analyses of the primary coolant have exceeded the limits of Specification 3.4.9 during the year. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (micro Ci/gm) and one other radioiodine isotope concentration (micro Ci/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL SEALED SOURCE LEAKAGE REPORT

6.9.1.14 A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.15 The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.10.1.

The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," approved by NRC SER dated October 16, 1995.

The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Commission in accordance with the requirements of 10CFR50.4 within the time period specified for each report. Reports should be submitted to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
c. ALL REPORTABLE EVENTS submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to the procedures required by Specification 6.8.1.
f. Records of radioactive shipments.
g. Records of sealed source and fission detector leak tests and results.

ENCLOSURE 4

Joseph M. Farley Nuclear Plant Unit 2
Pressure Temperature Limits Report
Technical Specification Changes

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DEFINITIONS

SECTION PAGE

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ADMINISTRATIVE CONTROLS

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1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71; State regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

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1.23 PURGE and PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

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3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limits specified in the PRESSURE TEMPERATURE LIMITS REPORT (PTLR) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.

ACTION:

With any of the above limits specified in the PTLR exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation or inspection to determine the effects of the out-of-limit condition on the fracture toughness of the Reactor Pressure Vessel; determine that the Reactor Pressure Vessel remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

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PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

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The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," approved by NRC SER dated October 16, 1995.

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c. All REPORTABLE EVENTS submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to the procedures required by Specification 6.8.1.
f. Records of radioactive shipments.
g. Records of sealed source and fission detector leak tests and results.

ENCLOSURE 5

Joseph M. Farley Nuclear Plant Units 1 and 2
Pressure Temperature Limits Report
Technical Specification Changes

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DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

ADMINISTRATIVE CONTROLS

INSERT 2 PRESSURE TEMPERATURE LIMITS REPORT (PTLR) 6-20a

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1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

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1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

FARLEY NUCLEAR PLANT - UNIT 1 PTLR SUBMITTAL

TECHNICAL SPECIFICATIONS MARKUPS

INSERT 3

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

1.21a The PRESSURE TEMPERATURE LIMITS REPORT (PTLR) is the unit specific document that provides the reactor vessel pressure and temperature (P/T) limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These P/T limits shall be determined for each fluence period or effective full-power years (EFPYs) in accordance with Specification 6.9.1.15. Plant operation within these operating limits is addressed in LCO 3.4.10.1, RCS Pressure/Temperature Limits.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

LIMITS SPECIFIED IN THE PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the ~~limit lines shown on Figures 3.4-2 and 3.4-3~~ during heatup, cooldown, criticality, and inservice leak and hydrostatic testing. ~~with:~~

~~a. A maximum heatup of 100°F in any one hour period.~~

~~b. A maximum cooldown of 100°F in any one hour period.~~

~~c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.~~

APPLICABILITY: At all times.

ACTION:

SPECIFIED IN THE PTLR

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation or inspection to determine the effects of the out-of-limit condition on the fracture toughness of the Reactor Pressure Vessel; determine that the Reactor Pressure Vessel remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SPECIFIED IN THE PTLR

4.4.10.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.10.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10CFR50, Appendix H.

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MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	LOWER SHELL (PLATE NO. B6919-2)
COPPER CONTENT	:	0.14 WT%
NICKEL CONTENT	:	0.56 WT%
INITIAL RT_{NDT}	:	5° F
RT_{NDT} AFTER 16 EFPY	:	1/4T, 146.4° F
	:	3/4T, 121.5° F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60° F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY

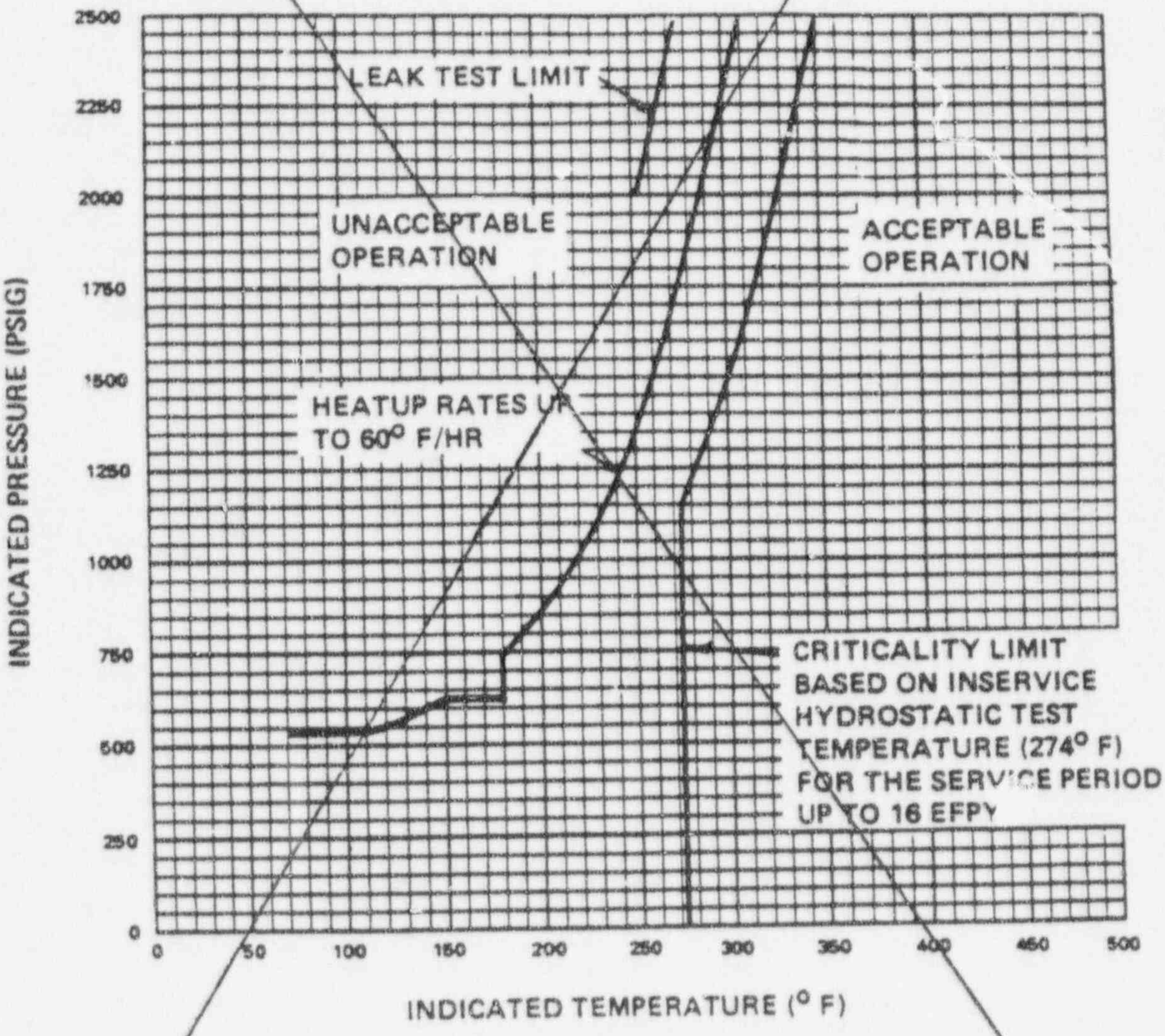

FIGURE 3.4-2 FARLEY UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 16 EFPY

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MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	LOWER SHELL (PLATE NO. B6919-2)
COPPER CONTENT	:	0.14 WT%
NICKEL CONTENT	:	0.56 WT%
INITIAL RT_{NDT}	:	5° F
RT_{NDT} AFTER 16 EFPY	:	1/4T, 146.4° F
	:	3/4T, 121.5° F

CURVES APPLICABLE FOR COOLDOWN UP TO 100° F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY

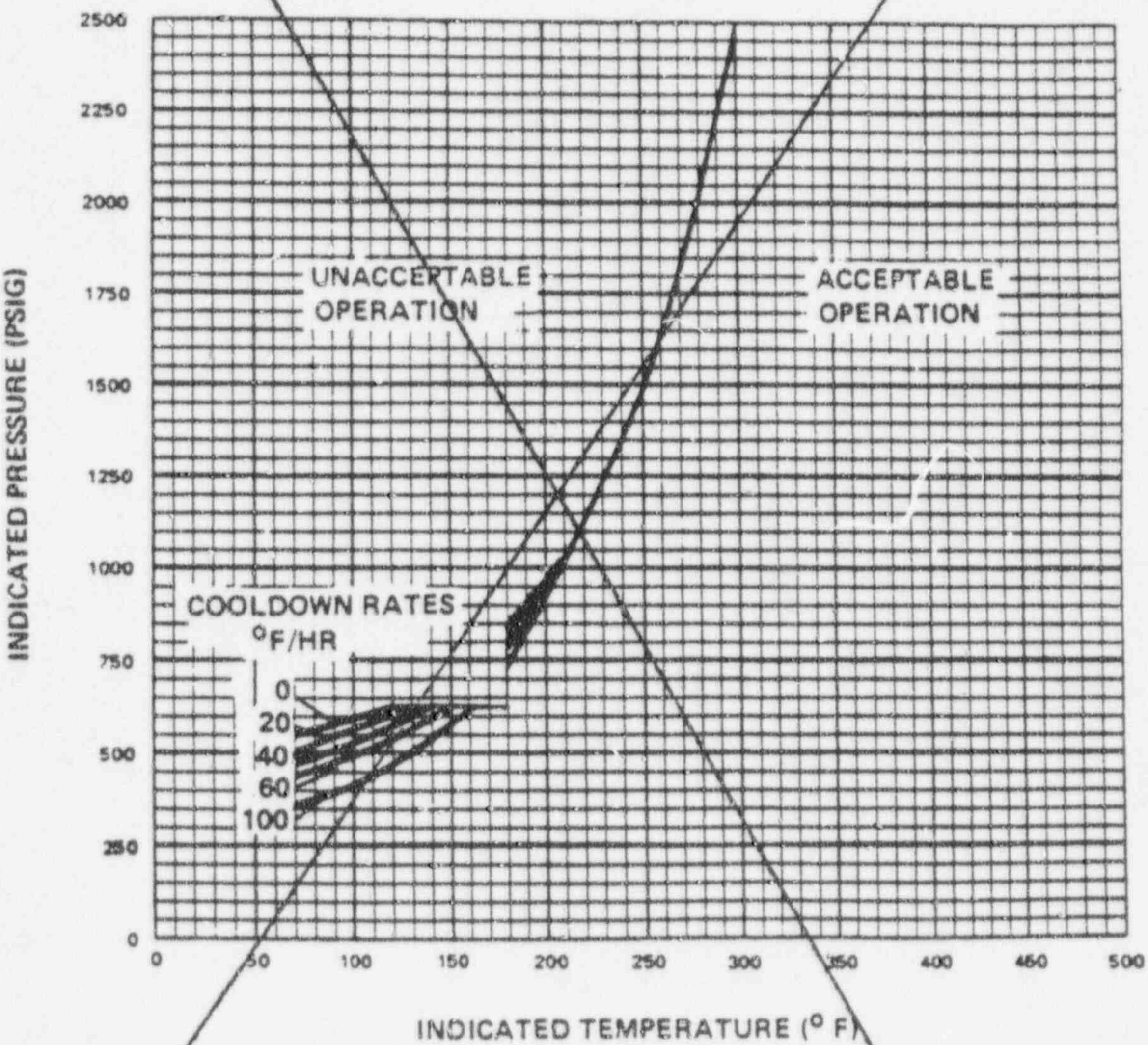

FIGURE 3.4-3 FARLEY UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 16 EFPY

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section ~~III,~~ XI Appendix G as required per 10 CFR Part 50 Appendix G.

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 3.4-2 and 3.4-3.~~ THE PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

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

   b) THE PTLR ~~Figures 3.4-2 and 3.4-3~~ defines limits to assure prevention of nonductile 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) These limit lines shall be calculated periodically using methods provided ~~below~~ IN WCAP-14040-NP-A, REVISION 2, METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES.

3) 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.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°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.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

~~The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with ASTM E185-82 and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."~~

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{ndt}, at the end of ~~16~~ 36 effective full power years (EFPY) of service life. The ~~16~~ EFPY service life period is chosen such that the limiting RT_{ndt} at the 1/4T location in the core region is greater than the RT_{ndt} of the limiting unirradiated material. The selection of such a limiting RT_{ndt} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

~~The reactor vessel materials have been tested to determine their initial RT_{ndt}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{ndt}. Therefore, an adjusted reference temperature, based upon the fluence and copper and nickel content of the material in question, can be predicted using Figure B 3/4.4-1, Figure B 3/4.4-2 and the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{ndt} at the end of 16 EFPY.~~

Values of ΔRT_{ndt} determined in ~~this manner~~ ACCORDANCE WITH WCAP-14040-NP-A, REVISION 2, may be used until the next results from the material surveillance program, evaluated according to ASTM E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in ~~FSAR Section 5.4~~ THE PTLR. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the surveillance capsule exceeds the calculated ΔRT_{ndt} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section ~~III~~ XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 ~~CFR Part 50 and these methods~~ are discussed in detail in ~~WCAP-7924-A.~~ WCAP-14040-NP-A, REVISION 2.

~~The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT~~ ~~$_{ndt}$~~ ~~, is used and this includes the radiation induced shift, ΔRT~~ ~~$_{ndt}$~~ ~~, corresponding to the end of the period for which heatup and cooldown curves are generated.~~

NOTE: THE TWO REMAINING PARAGRAPHS OF BASES SECTION 3/4.4.10, AS MODIFIED, ARE MOVED TO THIS PAGE FROM PAGE B3/4 4-11 FOR CONTINUITY.

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~~TABLE B 3/4.4-1~~

~~FARLEY UNIT 1 REACTOR VESSEL TOUGHNESS PROPERTIES~~

Component	Code No.	Material Type	Cu (%)	P (%)	Ni (%)	T_{ndt} (°F)	RT_{ndt} (°F)	Upper Shelf Energy MWD[c]	NMWD[d]
Closure head dome	B6901	A533,B,Cl.1	0.16	0.009	0.50	-30	-20[a]	140	-
Closure head segment	B6902-1	A533,B,Cl.1	0.17	0.007	0.52	-20	-20[a]	138	-
Closure head flange	B6915-1	A508, Cl.2	0.10	0.012	0.64	60[a]	60[a]	75[a]	-
Vessel flange	B6913-1	A508, Cl.2	0.17	0.011	0.69	60[a]	60[a]	106[a]	-
Inlet nozzle	B6917-1	A508, Cl.2	-	0.010	0.83	60[a]	60[a]	-	110
Inlet nozzle	B6917-2	A508, Cl.2	-	0.008	0.80	60[a]	60[a]	-	80
Inlet nozzle	B6917-3	A508, Cl.2	-	0.008	0.87	60[a]	60[a]	-	98
Outlet nozzle	B6916-1	A508, Cl.2	-	0.007	0.77	60[a]	60[a]	-	96.5
Outlet nozzle	B6916-2	A508, Cl.2	-	0.011	0.78	60[a]	60[a]	-	97.5
Outlet nozzle	B6916-3	A508, Cl.2	-	0.009	0.78	60[a]	60[a]	-	100
Nozzle shell	B6914-1	A508, Cl.2		0.010	0.68	30	30[a]	148	-
Inter. shell	B6903-2	A533,B,Cl.1	0.13	0.011	0.60	0	0	151.5	97
Inter. shell	B6903-3	A533,B,Cl.1	0.12	0.014	0.56	10	10	134.5	100
Lower shell	B6919-1	A533,B,Cl.1	0.14	0.015	0.55	-20	15	133	90.5
Lower shell	B6919-2	A533,B,Cl.1	0.14	0.015	0.56	-10	5	134	97
Bottom head ring	B6912-1	A508, Cl.2	-	0.010	0.72	10	10[a]	163.5	-
Bottom head segment	B6906-1	A533,B,Cl.1	0.15	0.011	0.52	-30	-30[a]	147	-
Bottom head dome	B6907-1	A533,B,Cl.1	0.17	0.014	0.60	-30	-30[a]	143.5	-
Inter. shell long. weld seam	M1.33	Sub Arc Weld	0.25	0.017	0.21	0[a]	0[a]	-	-
Inter. to lower shell weld seams	G1.18	Sub Arc Weld	0.22	0.011	<0.20[b]	0[a]	0[a]	-	-
Lower shell long. weld seams	G1.08	Sub Arc Weld	0.17	0.022	<0.20[b]	0[a]	0[a]	-	-

[a] Estimate per NUREG-0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
[b] Estimated (low nickel weld wire used in fabricating vessel weld seams).
[c] Major working direction.
[d] Normal to major working direction.

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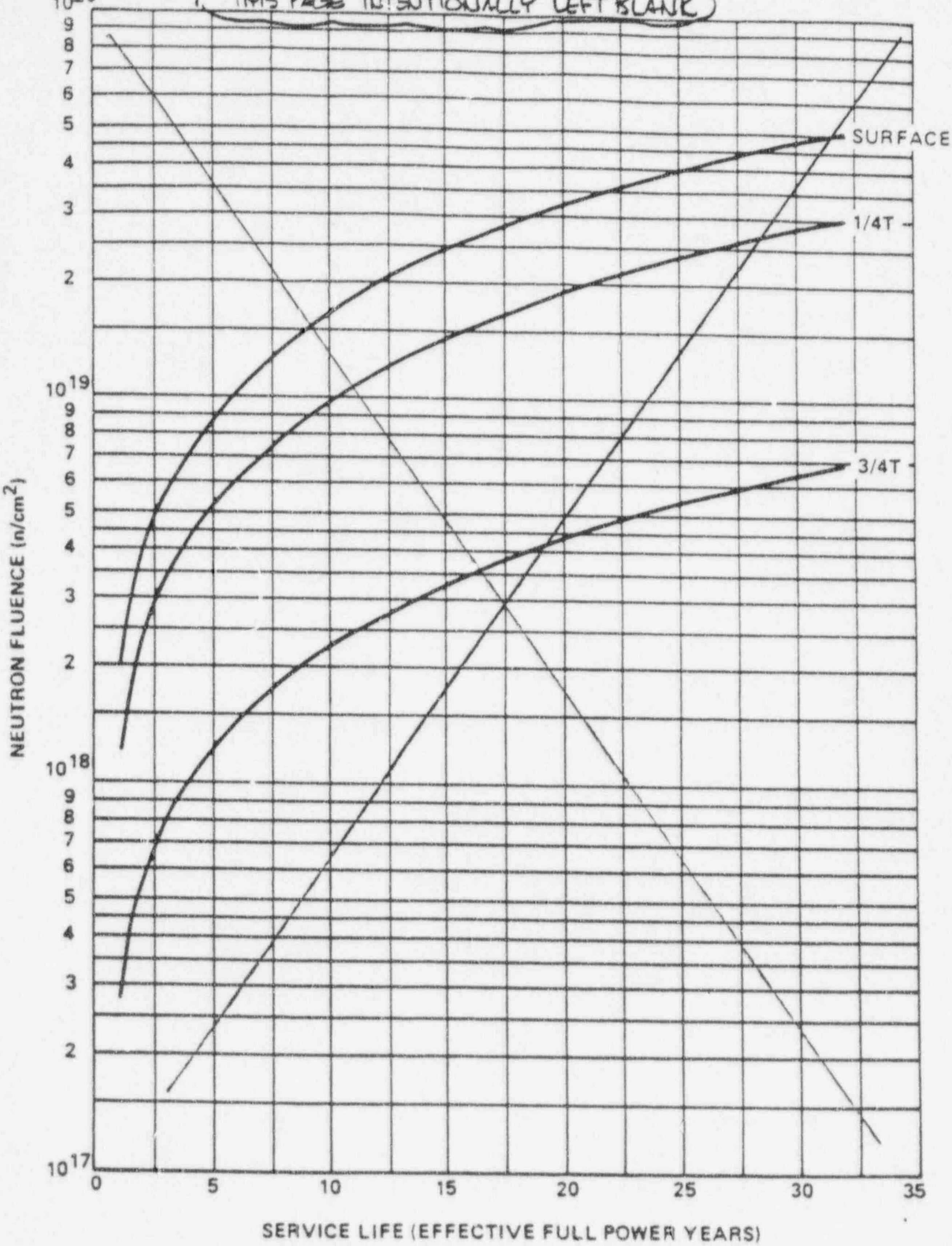

~~FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)~~

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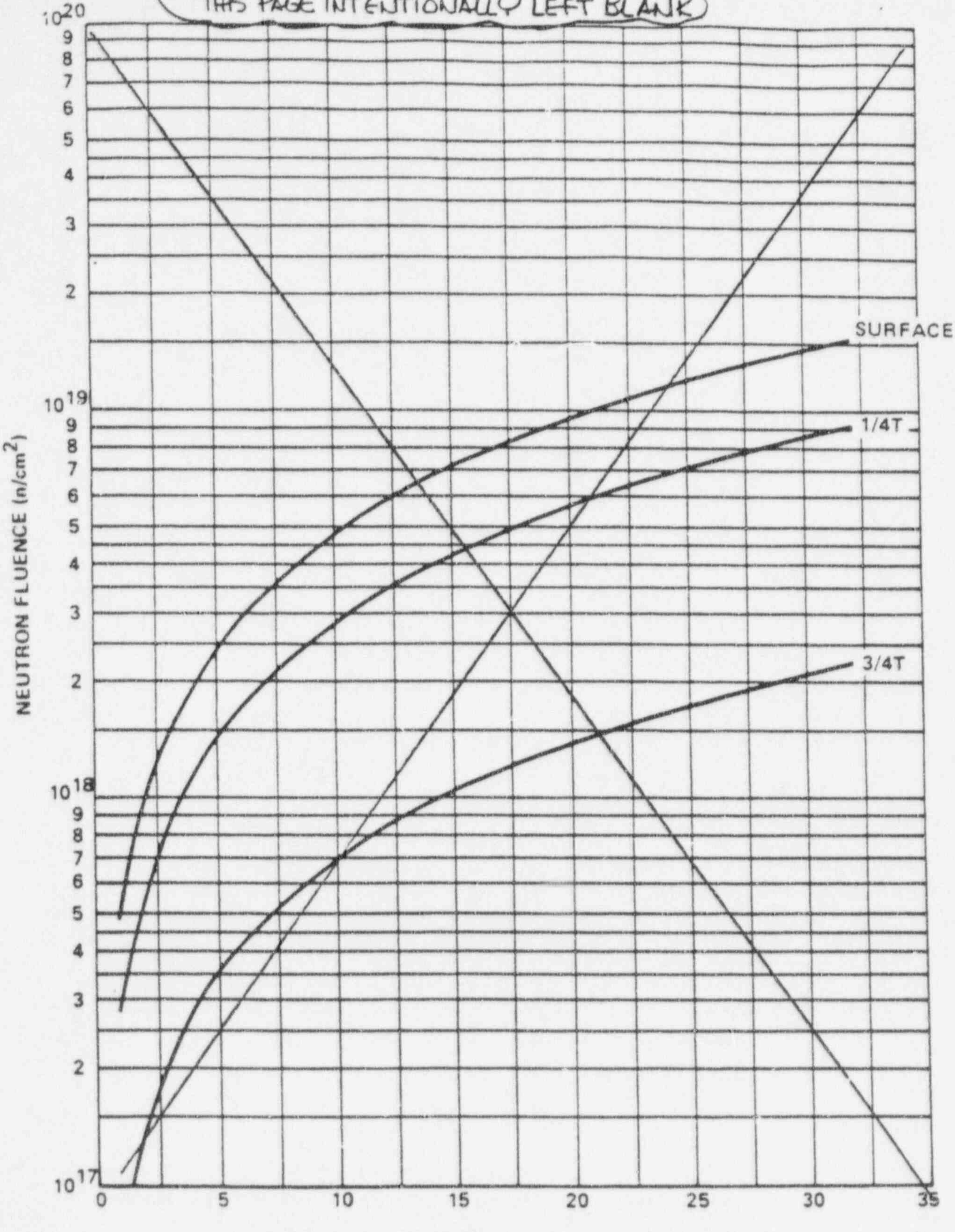

~~FIGURE B 3/4.4-2 FAST NEUTRON FLUENCE (E>1 MeV) AT 45° AS A FUNCTION OF FULL POWER SERVICE (EFPY)~~

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~~The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:~~

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

~~where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:~~

$$C\ K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

~~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.~~

~~C = 2.0 for level A and B service limits, and~~

~~C = 1.5 for inservice hydrostatic and leak test operations.~~

~~At any time during the heautp or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are~~

calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed 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 at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

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HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

~~The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.~~

~~Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 1). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. As a result, such a fracture analysis was performed for Farley Unit 2. These Farley Unit 2 fracture analysis results are applicable to Farley Unit 1 since the pertinent parameters are identical for both plants. Based upon this fracture analysis, the 16 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.~~

SEE NOTE BELOW

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of either RHR relief valve or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F, or (2) the start of ~~3 charging pumps~~ ALL OPERABLE and their injection into a water solid RCS.

INSERT 4

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

INSERT 4 IN THE CASE OF THE INJECTION BY THE CHARGING PUMPS, THE ANALYSIS IS BASED ON THE START OF THE MAXIMUM NUMBER OF OPERABLE CHARGING PUMPS ALLOWED BY THE TECHNICAL SPECIFICATIONS.

NOTE: THE PARAGRAPHS INCLUDED IN THE BUBBLE, AS MODIFIED, ARE MOVED TO PAGE B 3/4 4-8 FOR CONTINUITY OF BASES 3/4.4.10.

ANNUAL REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY REPORT

6.9.1.13 This annual report is only required when the results of specific activity analyses of the primary coolant have exceeded the limits of Specification 3.4.9 during the year. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (micro Ci/gm) and one other radioiodine isotope concentration (micro Ci/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL SEALED SOURCE LEAKAGE REPORT

6.9.1.14 A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

INSERT 5

See FOLLOWING PAGE

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Commission in accordance with the requirements of 10CFR50.4 within the time period specified for each report. Reports should be submitted to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
c. ALL REPORTABLE EVENTS submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to the procedures required by Specification 6.8.1.
f. Records of radioactive shipments.
g. Records of sealed source and fission detector leak tests and results.

NOTE: SECTIONS 6.9.1.15, 6.9.2, AND 6.10 MOVED TO PAGE 6-20a.

INSERT 5

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.15 The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.10.1.

The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," approved by NRC SER dated October 16, 1995.

The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

SECTION

PAGE

1.0 DEFINITIONS

ADMINISTRATIVE CONTROLS

INSERT 2 PRESSURE TEMPERATURE LIMITS REPORT 6-20a

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System Component body, pipe wall or vessel wall.

INSERT 3 →

SEE FOLLOWING PAGE

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71; State regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

INSERT 3

<u>PRESSURE TEMPERATURE LIMITS REPORT (PTLR)</u>

1.21a The PRESSURE TEMPERATURE LIMITS REPORT (PTLR) is the unit specific document that provides the reactor vessel pressure and temperature (P/T) limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These P/T limits shall be determined for each fluence period or effective full-power years (EFPYs) in accordance with Specification 6.9.1.15. Plant operation within these operating limits is addressed in LCO 3.4.10.1, RCS Pressure/Temperature Limits.

REACTOR COOLANT SYSTEM

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITS SPECIFIED IN THE PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

LIMITING CONDITION FOR OPERATION

3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the ~~limit lines shown on Figures 3.4-2 and 3.4-3~~ during heatup, cooldown, criticality, and inservice leak and hydrostatic testing. ~~with:~~

~~a. A maximum heatup of $100^\circ F$ in any one hour period.~~

~~b. A maximum cooldown of $100^\circ F$ in any one hour period.~~

~~c. A maximum temperature change of less than or equal to $10^\circ F$ in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.~~

APPLICABILITY: At all times.

ACTION:

SPECIFIED IN THE PTLR

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation or inspection to determine the effects of the out-of-limit condition on the fracture toughness of the Reactor Pressure Vessel; determine that the Reactor Pressure Vessel remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than $200^\circ F$ and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SPECIFIED IN THE PTLR

4.4.10.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.10.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10CFR50, Appendix H.

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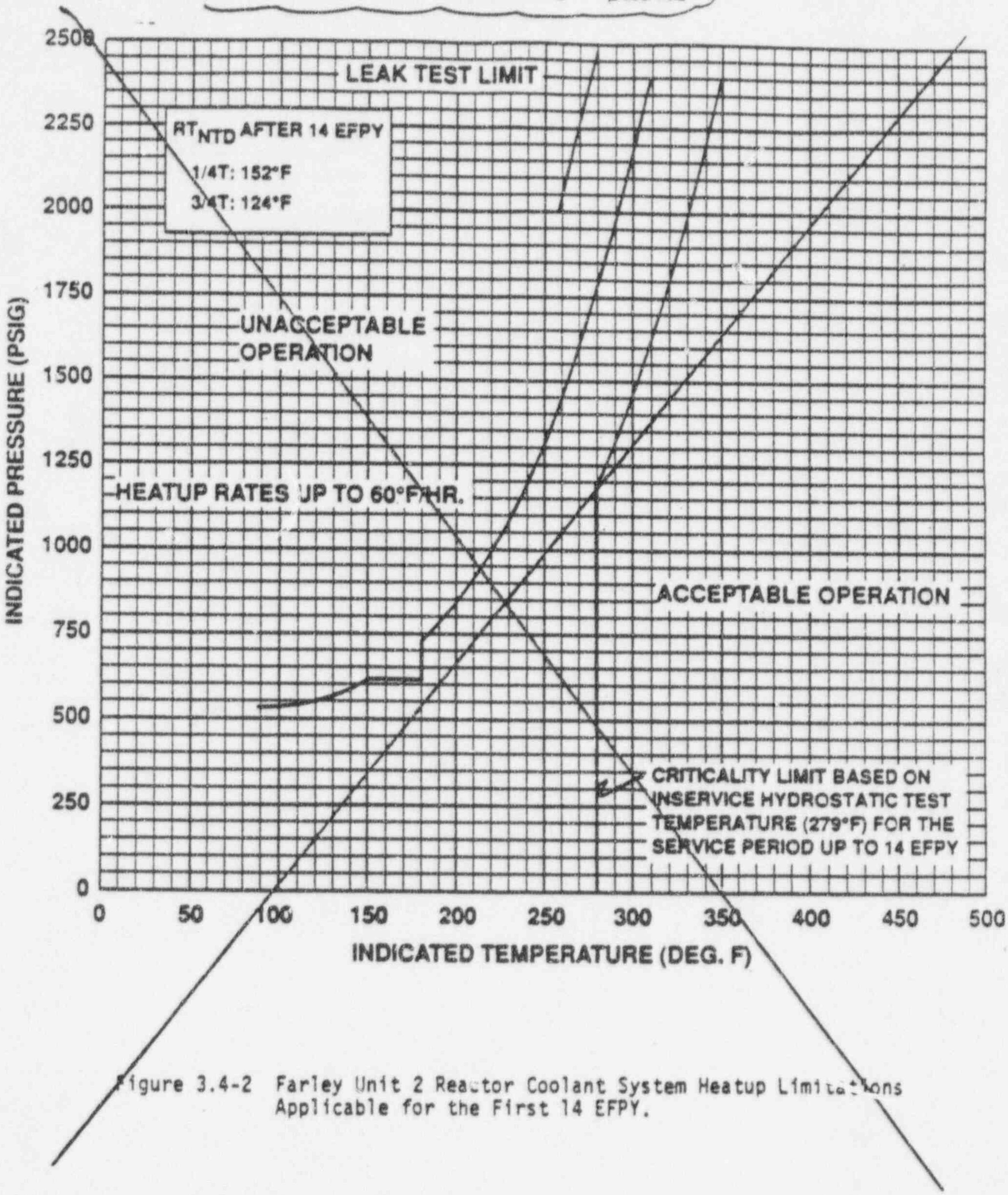

Figure 3.4-2 Farley Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 14 EFPY.

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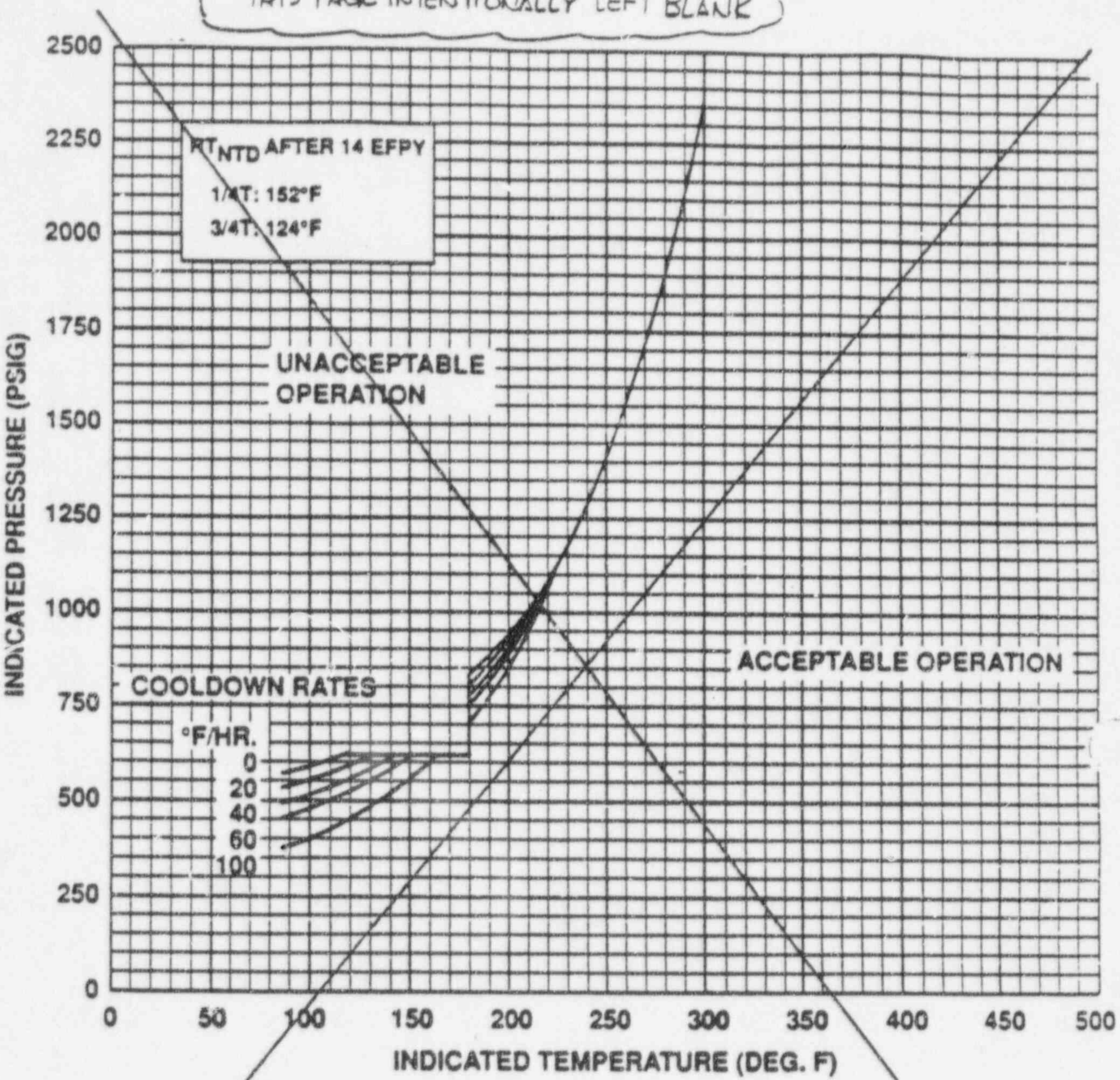

Figure 3.4-3 Farley Unit 2 Reactor Cooling System Cooldown Limitations Applicable for the First 14 EFPY.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section ~~III~~ XI, Appendix G as required per 10CFR Part 50 Appendix G. 1

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 3.4-2 and 3.4-3 for the first full-power service period.~~ THE PRESSURE TEMPERATURE LIMITS REPORT (PTLR).

   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. IN THE PTLR

   b) ~~Figures 3.4-2 and 3.4-3~~ THE PTLR define(s) limits to assure prevention of nonductile 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) These limit lines shall be calculated periodically using methods provided ~~below.~~ IN WCAP-14040-NP-A, REVISION 2, METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES.

3) 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.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°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.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

~~The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."~~

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{ndt}, at the end of ~~14~~ 36 effective full power years (EFPY) of service life. The ~~14~~ 36 EFPY service life period is chosen such that the limiting RT_{ndt} at the 1/4T location in the core region is greater than the RT_{ndt} of the limiting unirradiated material. The selection of such a limiting RT_{ndt} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements. 1

~~The reactor vessel materials have been tested to determine their initial RT_{ndt}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{ndt}. Therefore, an adjusted reference temperature, based upon the fluence and the nickel and copper content of the material in question, can be predicted using WCAP-12471 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{ndt} at the end of 14 EFPY.~~

ACCORDANCE WITH WCAP-14040-NP-A, REVISION 2,

Values of ΔRT_{ndt} determined in ~~this manner~~ may be used until the next results from the material surveillance program, evaluated according to ASTM E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in ~~FSAR Section 5.4~~ THE PTLR. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the next surveillance capsule exceeds the calculated ΔRT_{ndt} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section ~~III~~ XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR 50 and these methods are discussed in detail in ~~WCAP-7924-A.~~ WCAP-14040-NP-A, REVISION 2.

~~The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{ndt}, is used and this includes the radiation induced shift, ΔRT_{ndt}, corresponding to the end of the period for which heatup and cooldown curves are generated.~~

NOTE: THE TWO REMAINING PARAGRAPHS OF BASES SECTION 3/4.4.10, AS MODIFIED, ARE MOVED TO THIS PAGE FROM PAGE B 3/4 4-14 FOR CONTINUITY.

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~~TABLE B3/4.4-1~~

~~REACTOR VESSEL TOUGHNESS DATA~~

Component	Code No.	Grade	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
CL. HD. Dome	B7215-1	A533,B,CL.1	0.17	0.010	0.49	-30	16(a)	83(a)	128
CL. HD. Flange	B7207-1	A508,CL.2	0.14	0.011	0.65	60(a)	60(a)	>56(a)	>86(c)
VES. Flange	B7206-1	A508,CL.2	0.10	0.012	0.67	60(a)	60(a)	>71(a)	>109
Inlet Noz.	B7218-2	A508,CL.2	-	0.010	0.68	50(a)	50(a)	103(a)	158
Inlet Noz.	B7218-1	A508,CL.2	-	0.010	0.71	32(a)	32(a)	112(a)	172
Inlet Noz.	B7218-3	A508,CL.2	-	0.010	0.72	60(a)	60(a)	98(a)	150
Outlet Noz.	B7217-1	A508,CL.2	-	0.010	0.73	60(a)	60(a)	100(a)	154
Outlet Noz.	B7217-2	A508,CL.2	-	0.010	0.72	6(a)	6(a)	108(a)	167
Outlet Noz.	B7217-3	A508,CL.2	-	0.010	0.72	48(a)	48(a)	103(a)	158
Upper Shell	B7216-1	A508,CL.2	-	0.010	0.73	30	30(a)	97(a)	149
Inter Shell	B7203-1	A533,B,CL.1	0.14	0.010	0.60	-40	15	99	140
Inter Shell	B7212-1	A533,B,CL.1	0.20	0.018	0.60	-30	-10	99	134
Lower Shell	B7210-1	A533,B,CL.1	0.13	0.010	0.56	-40	18	103	128
Lower Shell	B7210-2	A533,B,CL.1	0.14	0.015	0.57	-30	0	99	145
Trans. Ring	B7208-1	A508,CL.2	-	0.010	0.73	40	40(a)	89(a)	137
Bot. HD. Dome	B7214-1	A533,B,CL.1	0.11	0.007	0.48	-30	-2(a)	87(a)	134
Inter. Shell	A1.46	SMAW	0.02	0.009	0.96	0(a)	0(a)	>131	-
Long Seams	A1.40	SMAW	0.02	0.010	0.93	-60	-60	>106	-
Inter Shell to Lower Shell	G1.50	SAW	0.13	0.016	<.20(b)	-40	-40	>102	-
Lower Shell Long Seams	G1.39	SAW	0.05	0.006	<.20(b)	-70	-70	>126	-

(a) Estimate per NUREG 0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
(b) Estimated.
(c) Upper shelf not available, value represents minimum energy at the highest test temperature.

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The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

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.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heautp or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are

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BASES

calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed 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 at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

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HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

~~The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.~~

~~Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 2). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. Based upon such a fracture analysis for Farley Unit 2, the 14 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.~~

SEE NOTE BELOW

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of either RHR relief valve or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F, or (2) the start of ~~3~~ charging pumps ARE OPERABLE and their injection into a water solid RCS.

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3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

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IN THE CASE OF THE INJECTION BY THE CHARGING PUMPS, THE ANALYSIS IS BASED ON THE START OF THE MAXIMUM NUMBER OF OPERABLE CHARGING PUMPS ALLOWED BY THE TECHNICAL SPECIFICATIONS.

NOTE: THE PARAGRAPHS INCLUDED IN THE BUBBLE, AS MODIFIED, ARE MOVED TO PAGE B 3/4 4-8 FOR CONTINUITY OF BASES 3/4.4.10.

ANNUAL REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY REPORT

6.9.1.13 This annual report is only required when the results of specific activity analyses of the primary coolant have exceeded the limits of Specification 3.4.9 during the year. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (micro Ci/gm) and one other radioiodine isotope concentration (micro Ci/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL SEALED SOURCE LEAKAGE REPORT

6.9.1.14 A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

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SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Commission in accordance with the requirements of 10CFR50.4 within the time period specified for each report. Reports should be submitted to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
c. ALL REPORTABLE EVENTS submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to the procedures required by Specification 6.8.1.
f. Records of radioactive shipments.
g. Records of sealed source and fission detector leak tests and results.

NOTE: SECTIONS 6.9.1.15, 6.9.2, AND 6.10 MOVED TO PG. 6-20a.

INSERT 5

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.15 The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.10.1.

The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," approved by NRC SER dated October 16, 1995.

The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.