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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. NPF-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated March 21, 1990, as amended November 13, 1990, 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 1;

   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. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 26 are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: January 11, 1991

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 26 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 for hydrostatic or leak testing; Figure 3.4.6.1-2 for heatup by non-nuclear means, Figure 3.4.6.1-3 for cooldown following a nuclear shutdown and low-power PHYSICS TESTS; and Figures 3.4.6.1-4 and 3.4.6.1-5 for operations with a critical core other than low-power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown, and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3, 3.4.6.1-4, and 3.4.6.1-5 as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figures 3.4.6.1-4 and 3.4.6.1-5 within 15 minutes before the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

Provided the water level is in the range for power operation, the core may be critical when left of the criticality limit line if the pressure is maintained below 312 psig (see cross-hatched region in Figures 3.4.6.1-4 and 3.4.6.1-5). In this case, the reactor coolant temperature and pressure shall be determined to be within the cross-hatched region of figures 3.4.6.1-4 and 3.4.6.1-5 within 15 minutes before withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined to determine changes in reactor pressure vessel material properties as required by 10CFR50, Appendix H, in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to bring up to date the curves of Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3, 3.4.6.1-4 and 3.4.6.1-5.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F when reactor vessel head bolting studs are under full tension:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

    1. $\leq$90°F, at least once per 12 hours.
    2. $\leq$80°F, at least once per 30 minutes.

b. Within 30 minutes before, and at least once per 30 minutes during, tensioning of the reactor vessel head bolting studs.

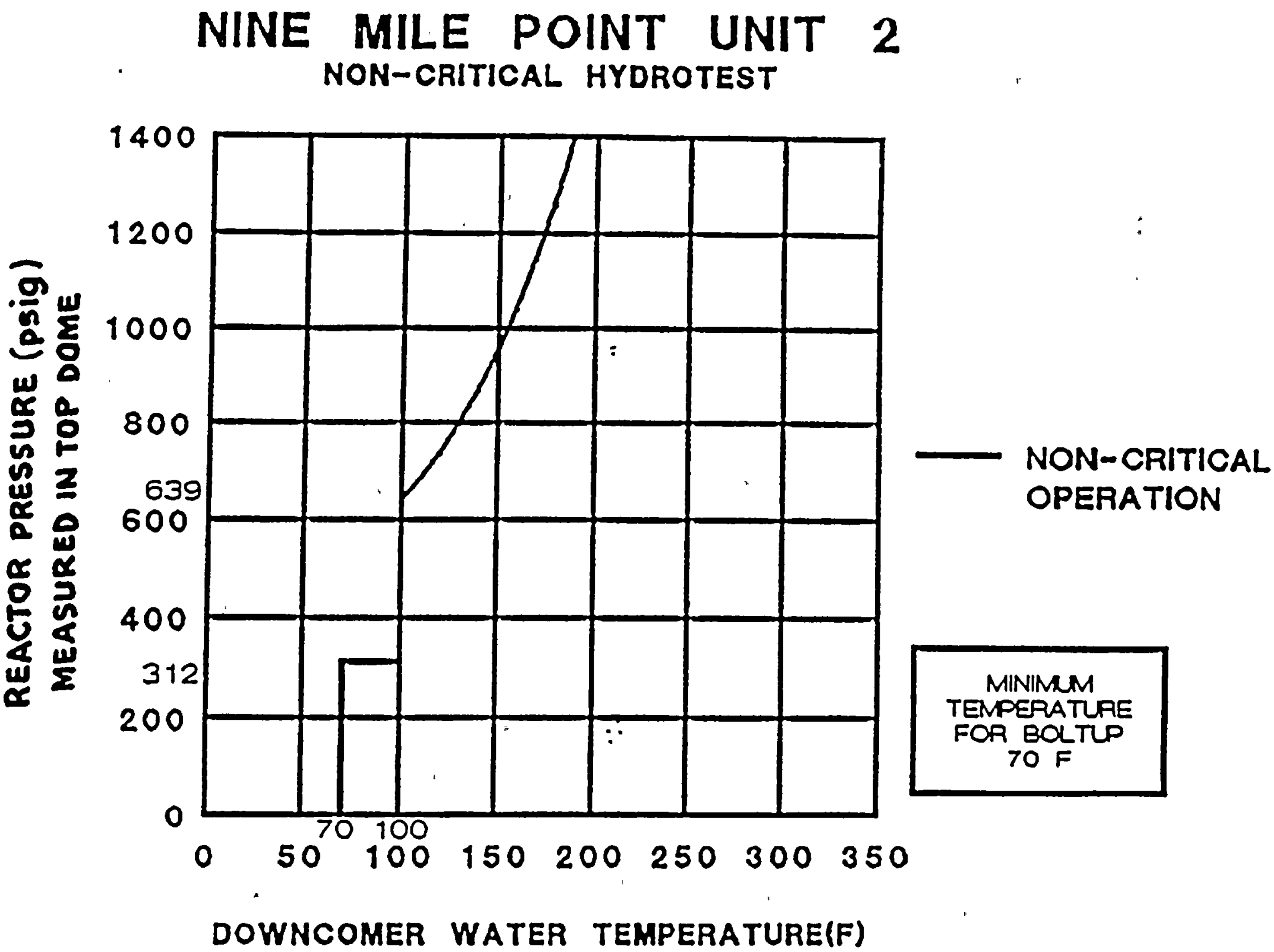

FIGURE 3.4.6.1-1 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING IN-SERVICE HYDROSTATIC TESTING AND LEAK TESTING (REACTOR NOT CRITICAL) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

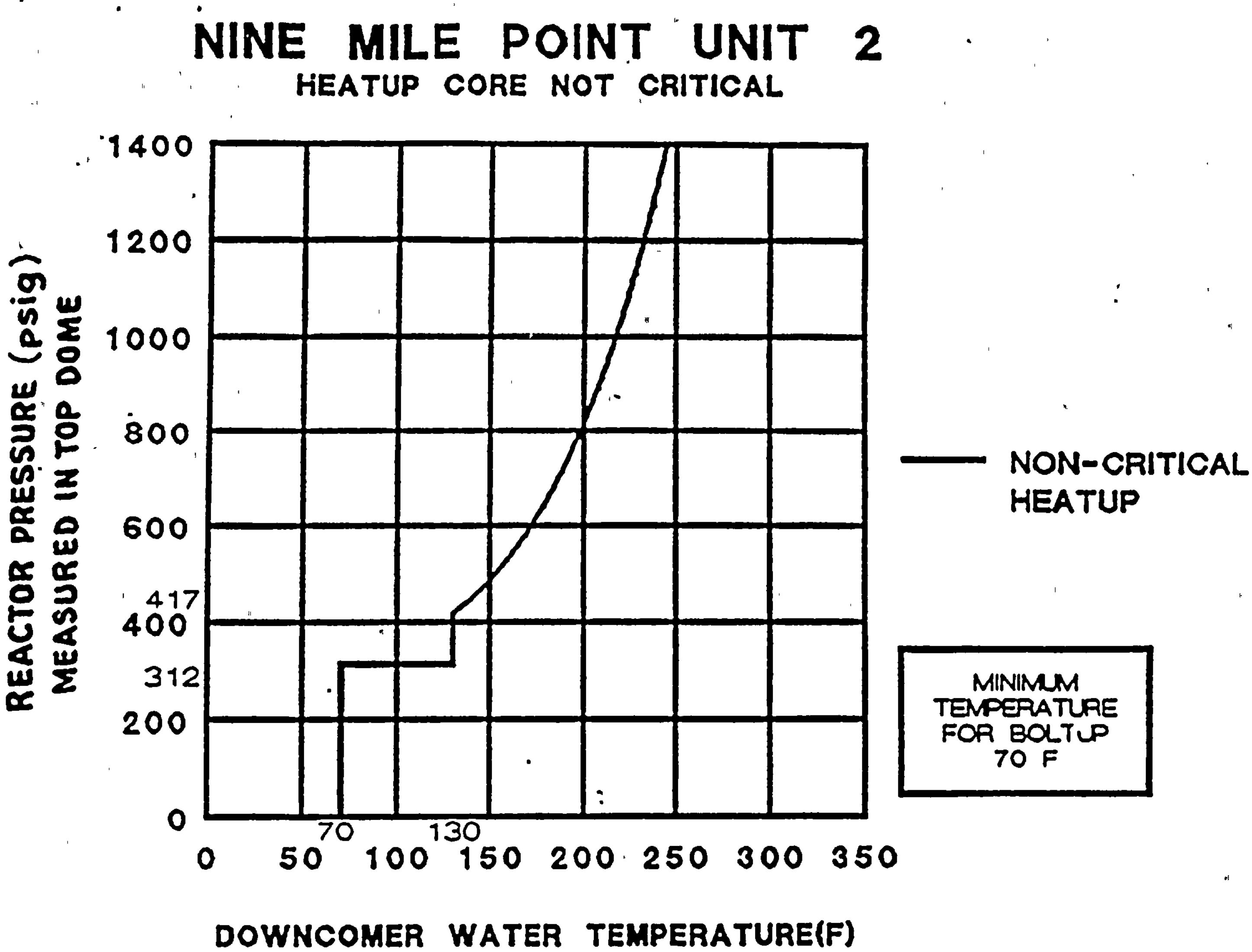

FIGURE 3.4.6.1-2 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING HEATUP AND LOW-POWER PHYSICS TESTS (REACTOR NOT CRITICAL) (HEATING RATE ≤ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

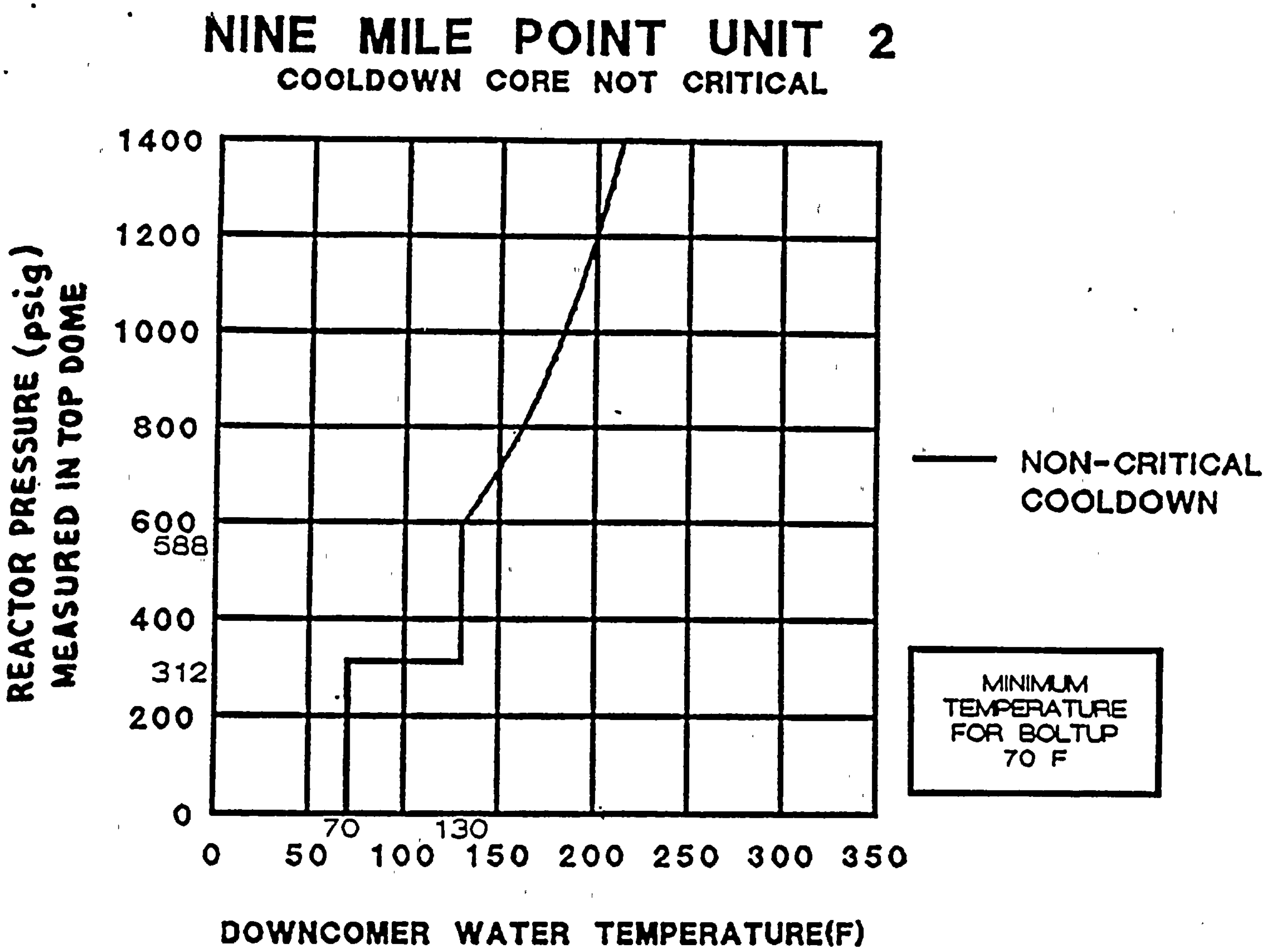

FIGURE 3.4.6.1-3 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING COOLDOWN AND LOW-POWER PHYSICS TESTS (REACTOR NOT CRITICAL) (COOLING RATE ≤ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

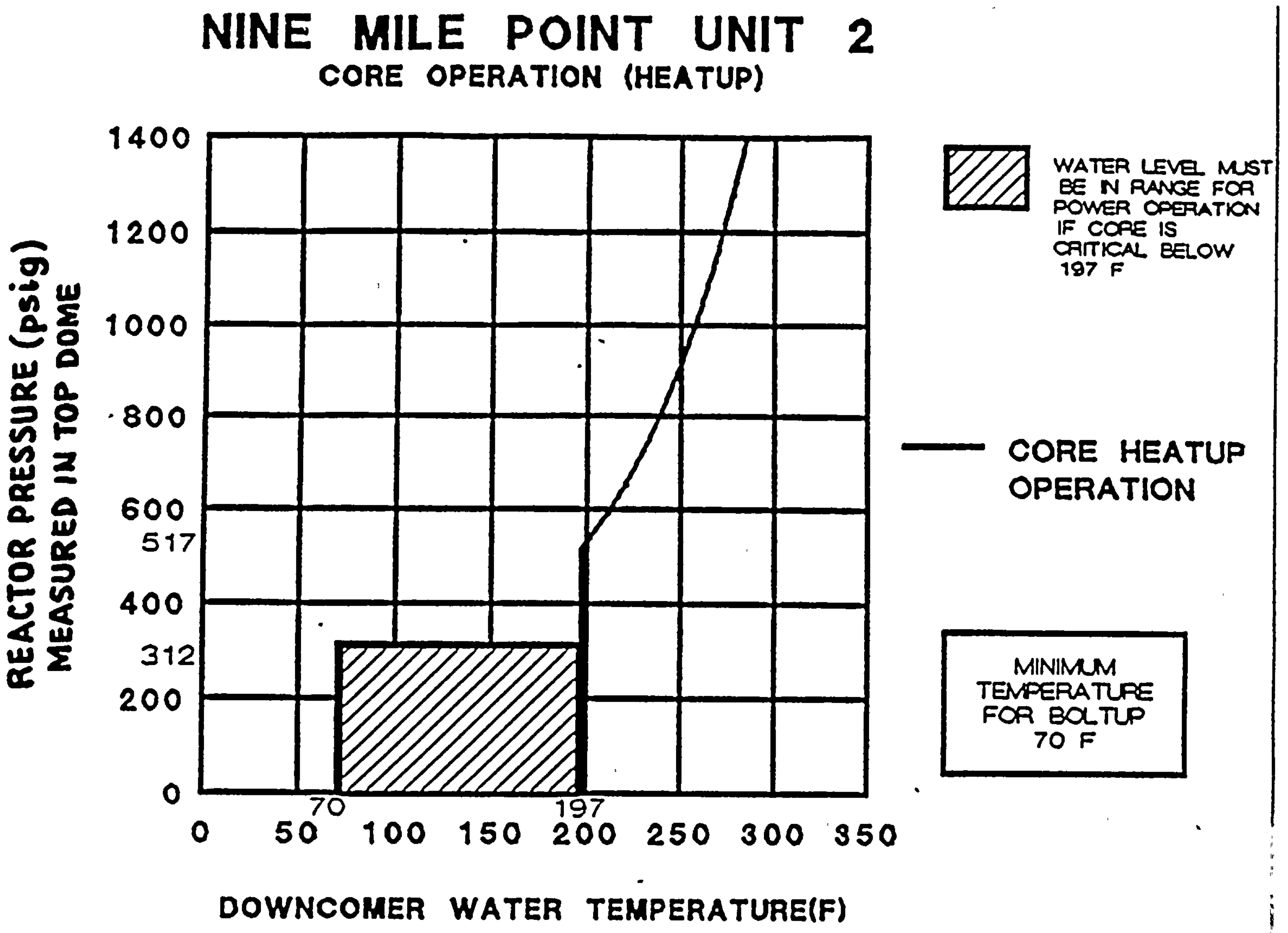

FIGURE 3.4.6.1-4 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING CORE OPERATION (CORE CRITICAL) (HEATUP AT A HEATING RATE ≤ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

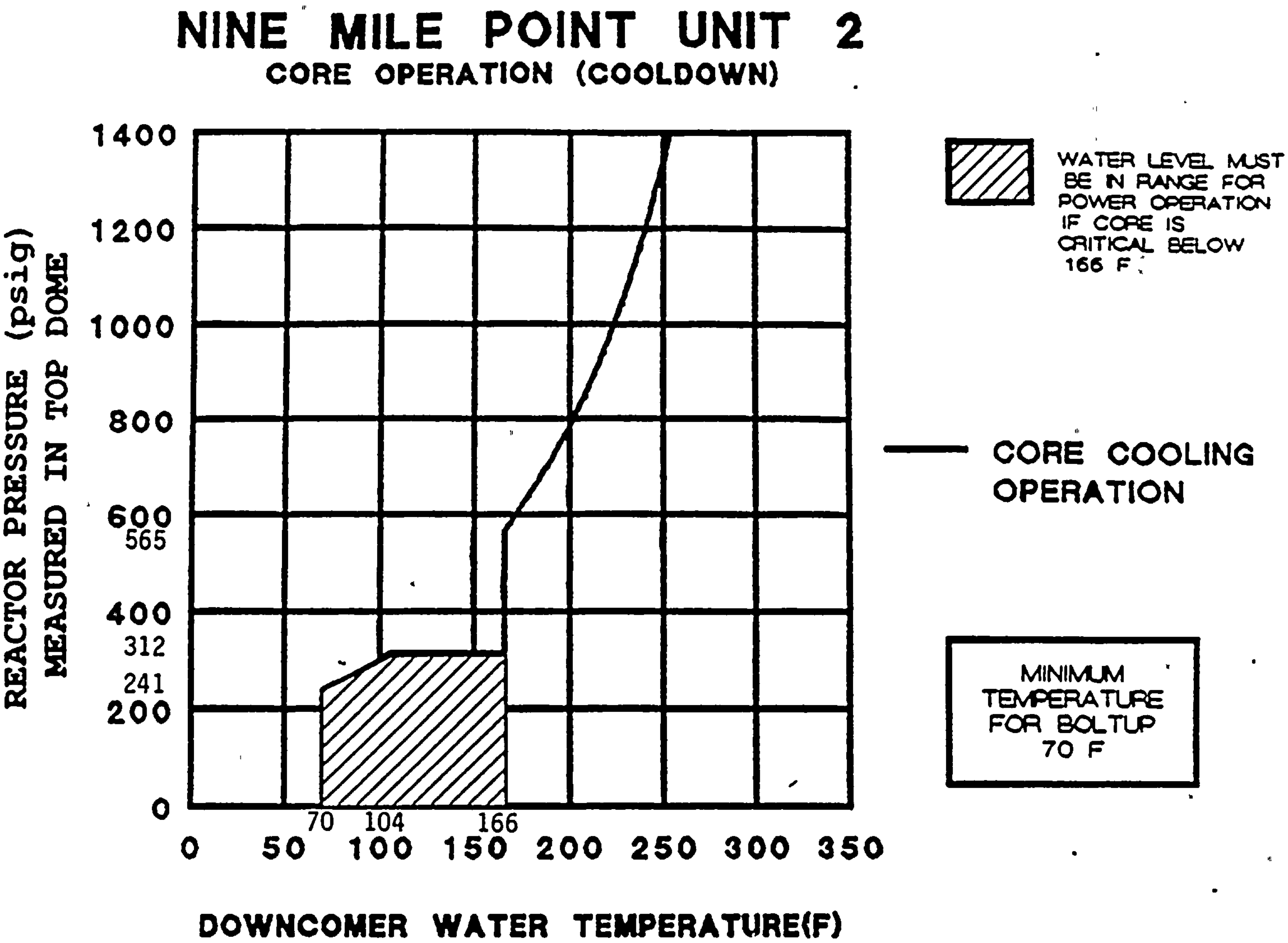

FIGURE 3.4.6.1-5 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING CORE OPERATION (CORE CRITICAL) (COOLDOWN AT A COOLING RATE $\leq$ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR @ 1/4 T	WITHDRAWAL TIME (EFPY)
1	3°	0.41	10
2	177°	0.41	20
3	183°	0.41	Spare

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

REACTOR STEAM DOME

LIMITING CONDITIONS FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1020 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1020 psig at least once per 12 hours.

* Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITIONS FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 4 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or
      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.4.8 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.8.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) before increasing the reactor coolant system temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) before increasing the reactor coolant system temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than Specification 4.0.5.

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.4.9.1 Two* shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation**,† with each loop consisting of at least:

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.††

b. With no RHR shutdown cooling mode loop in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within 1 hour, establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one shutdown cooling mode loop of the residual heat removal system or alternative method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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* One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

** The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period provided the other loop is OPERABLE.

† The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

†† Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat-removal methods.

REACTOR COOLANT SYSTEM

RESIDUAL HEAT REMOVAL

COLD SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.4.9.2 Two* shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation** † with each loop consisting of at least:

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system or alternative method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

* One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

** The shutdown cooling pump may be removed from operation for up to 2 hours every 8-hour period provided the other loop is OPERABLE.

† The shutdown cooling mode loop may be removed from operation during hydrostatic testing.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads from temperature and pressure changes in the system. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The operating limit curves of Figures 3.4.6.1-1 through 3.4.6.1-5 are derived from the fracture toughness requirements of 10CFR50, Appendix G, and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Subsection 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Bases Table B3/4.4.6-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material can be predicted using Bases Figure B3/4.4.6-1 and the recommendations of RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating irradiated specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1 through 3.4.6.1-5 shall be adjusted, as required, on the basis of the specimen data and recommendations of RG 1.99, Revision 2. Data obtained after removal of the first surveillance capsule will be used to adjust the fluence of Bases Figure B3/4.4.6-1.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1 through 3.4.6.1-5 for inservice hydrostatic testing and leak testing for critical operations have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10CFR50.

BASES TABLE B3/4.4.6-1

LIMITING REACTOR VESSEL TOUGHNESS

BELTLINE

COMPONENT	WELD SEAM ID OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	Cu(%)	Ni(%)	STARTING RT_{NDT} (°F)	12.8 EFPY ΔRT_{NDT} (°F)	UNIRRADIATED UPPER SHELF (FT-LB)	12.8 EFPY ART_{NDT} (°F)
Plate	SA-533, Gr. B, Cl. 1	C3147-2	0.11	0.63	0	26	86	65
Weld	Seam AB	4P7216/0751	0.06	0.85	-50	28	89	13

NON-BELTLINE

COMPONENT	WELD SEAM ID OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	MAX RT_{NDT} (°F)
Shell Ring	SA-533, Gr. B, Cl. 1	All Plates	+10
Bottom Head Dome	SA-533, Gr. B, Cl. 1	C3073/2	+10
Bottom Head Torus	SA-533, Gr. B, Cl. 1	C3073/2	+10
Top Head Dome	SA-533, Gr. B, Cl. 1	A0678/1	-20
Top Head Torus	SA-533, Gr. B, Cl. 1	C2325/2	-1
Top Head Flange	SA-508, Cl. 2	49D161, 49B168	-30
Vessel Flange	SA-508, Cl. 2	48D1072, 48B1121	-20
LPCI Nozzle*	SA-508, Cl. 2	Q2QL3W	-20
Feedwater Nozzle	SA-508, Cl. 2	Q2QL2W	-20
Weld	INMM/LINDE 124	All Heats	-20
Closure Studs	SA-540, Gr. B24	All Heats	+10**

* The design location of the low-pressure core injection (LPCI) nozzles results in these components and their related vessel welds to experience and end-of-life (EOL) fluence of 1.7×10^{17} n/cm^2 (E>1 MeV). As a result, the nozzles are predicted to have an EOL RT_{NDT} of -13°F and the limiting weld material will have an EOL RT_{NDT} of -12°F.

** Meet 45 ft-lb and 25 mils lateral expansion requirement at 10°F.