January 11, 1991

Docket No. 50-410

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Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 2 (TAC NOS. 71519 AND 76399)

The Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station Unit No. 2 (NMP-2). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 21, 1990, as amended November 13, 1990.

This amendment revises Technical Specification Section 3/4.4.6, Reactor Coolant System - Pressure/Temperature Limits, and the associated Bases. These changes are in accordance with Generic Letter 88-11.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 26 to NPF-69
2. Safety Evaluation

cc: w/enclosures
See next page

PDI-1:LA	PDI-1:PE	PDI-1PM	OGC	PDI-1:D
CVogan	DOudinot	DBrinkman:rsc		RACapra
12-16-90	12/17/90	12/17/90	12/20/90	01/11/91

DOCUMENT NAME: NMP2 AMEND 76399

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P PDR

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

January 11, 1991

Docket No. 50-410

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 2 (TAC NOS. 71519 and 76399)

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Sincerely,

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 26 to NPF-69
2. Safety Evaluation

cc: w/enclosures
See next page

Mr. B. Ralph Sylvia
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit 2

cc:

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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:

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(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

Revise Appendix A as follows:

Remove Page	Insert Page
x	x
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
---	3/4 4-35
---	3/4 4-36
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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.

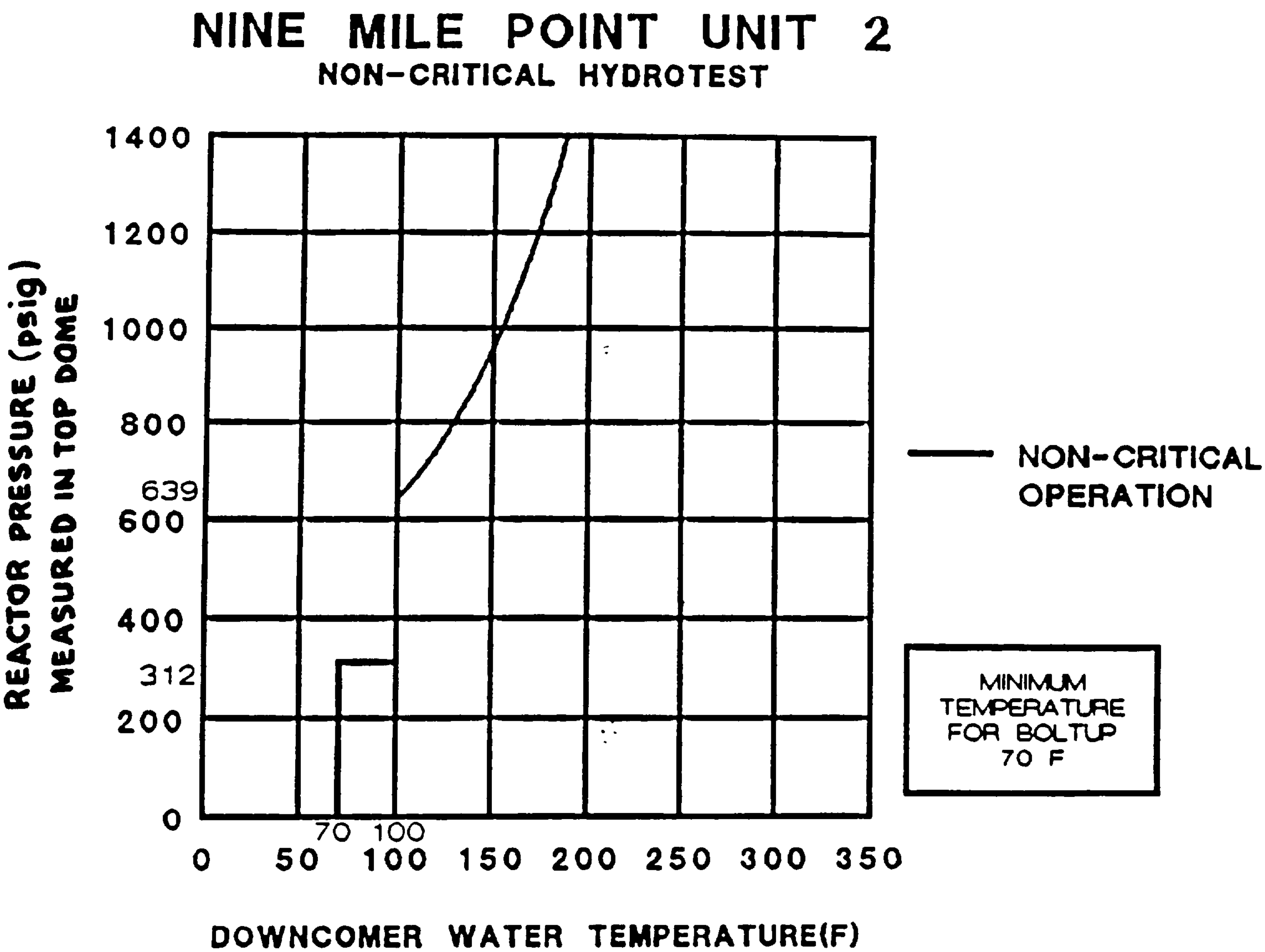

FIGURE 3.4.6.1-1 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING IN-SERVICE HYDROSTATIC TESTING AND LEAK TESTING (REACTOR NOT CRITICAL) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

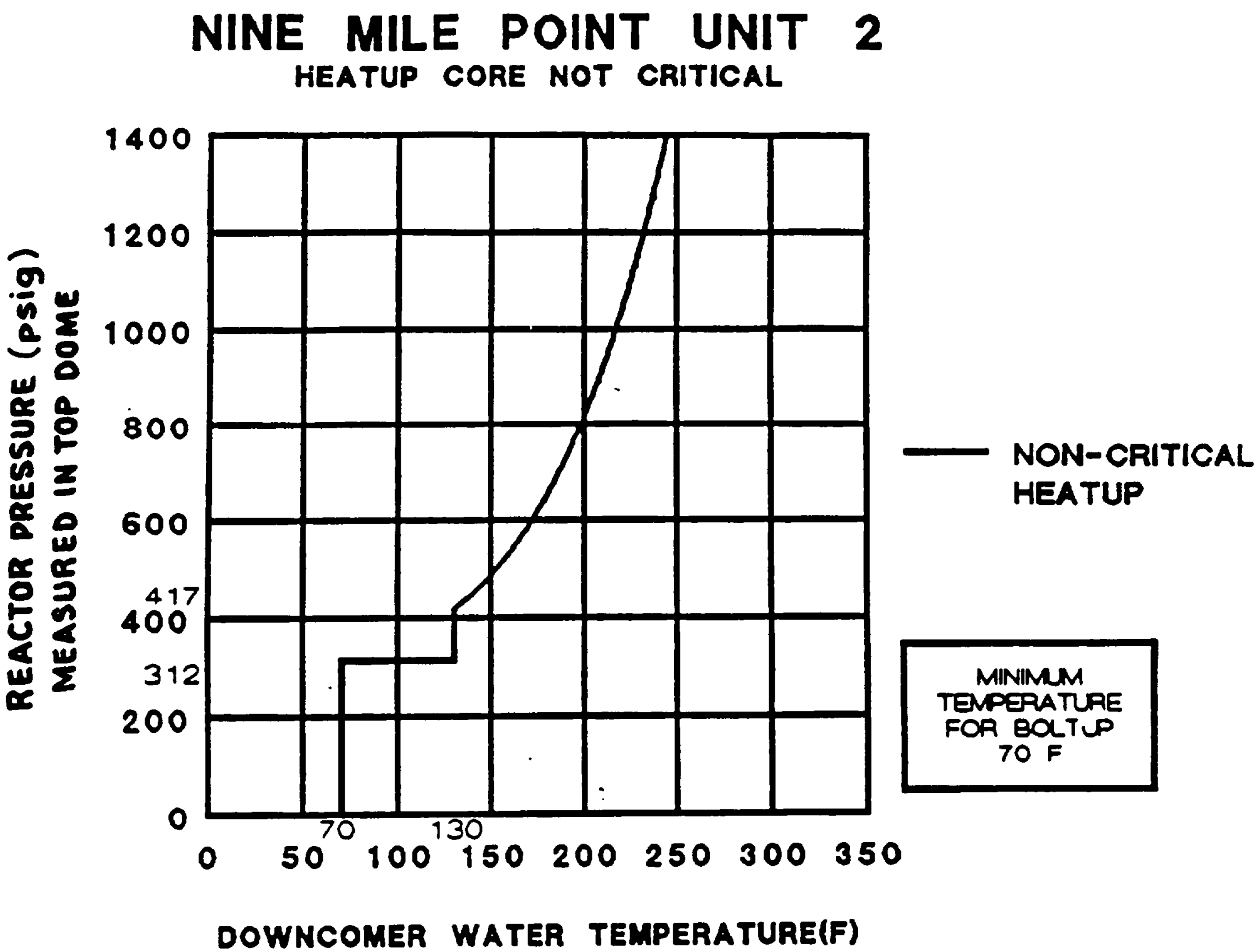

FIGURE 3.4.6.1-2 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING HEATUP AND LOW-POWER PHYSICS TESTS (REACTOR NOT CRITICAL) (HEATING RATE ≤ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

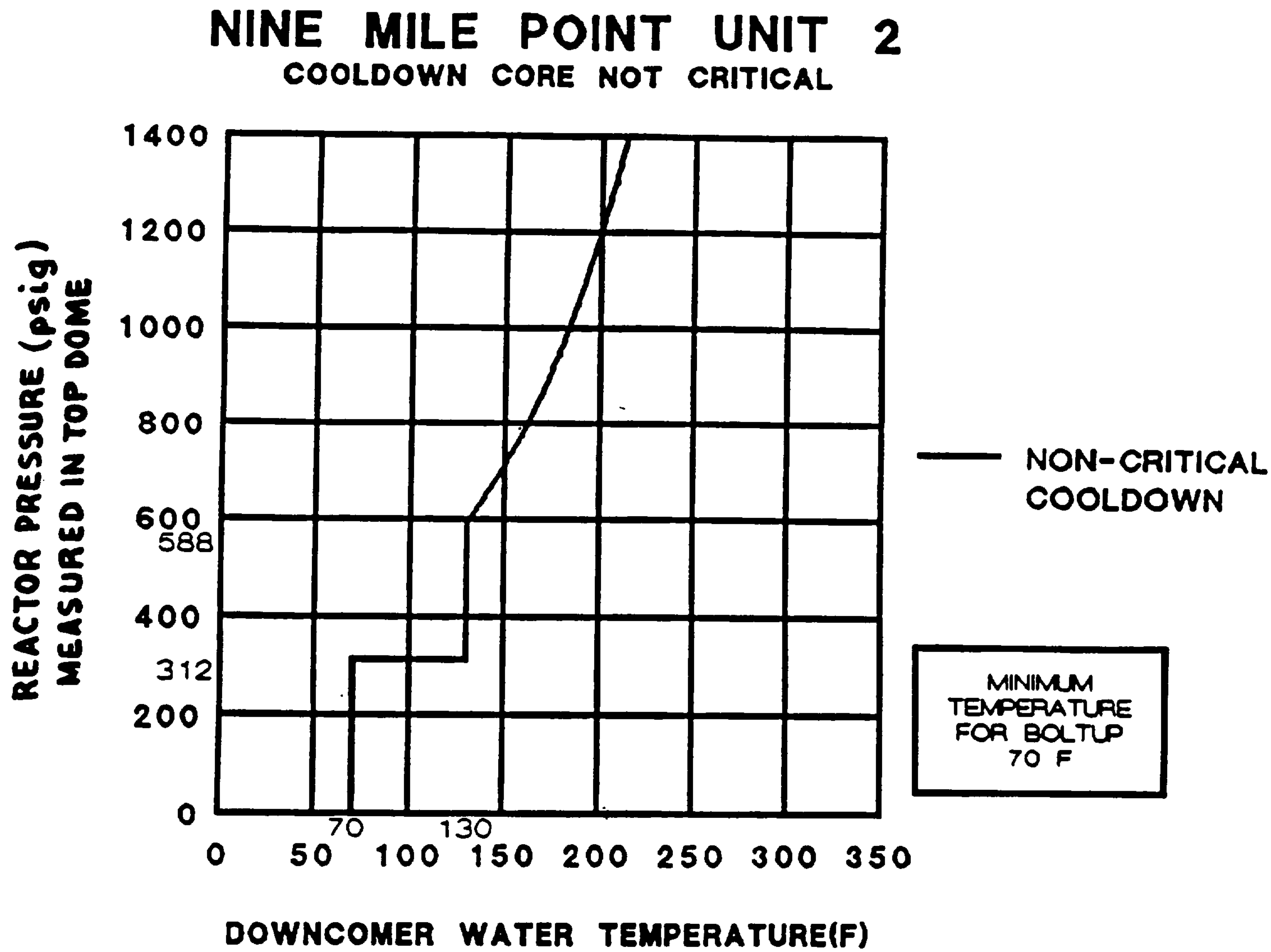

FIGURE 3.4.6.1-3 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING COOLDOWN AND LOW-POWER PHYSICS TESTS (REACTOR NOT CRITICAL) (COOLING RATE ≤ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

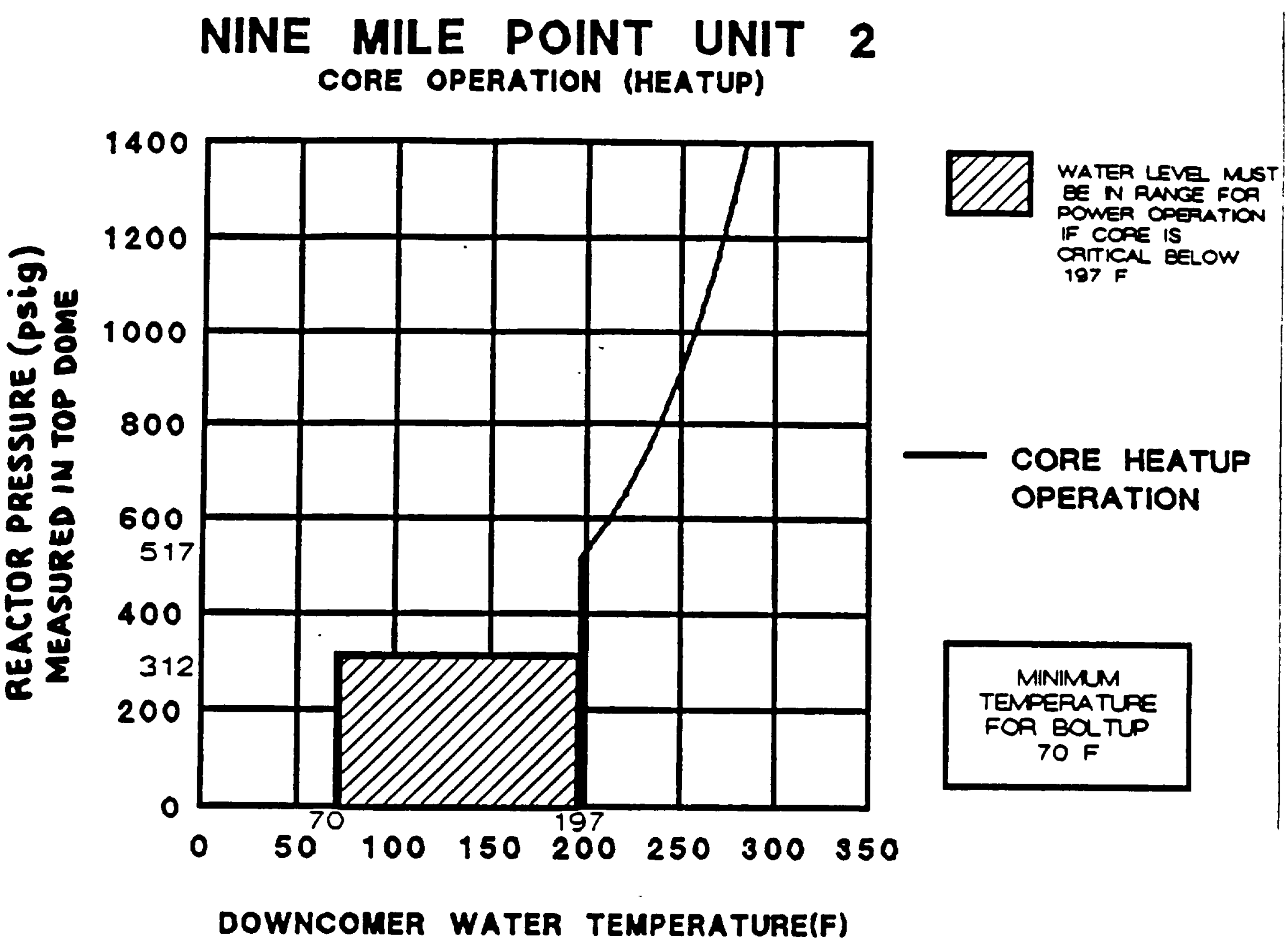

FIGURE 3.4.6.1-4 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING CORE OPERATION (CORE CRITICAL) (HEATUP AT A HEATING RATE ≤ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

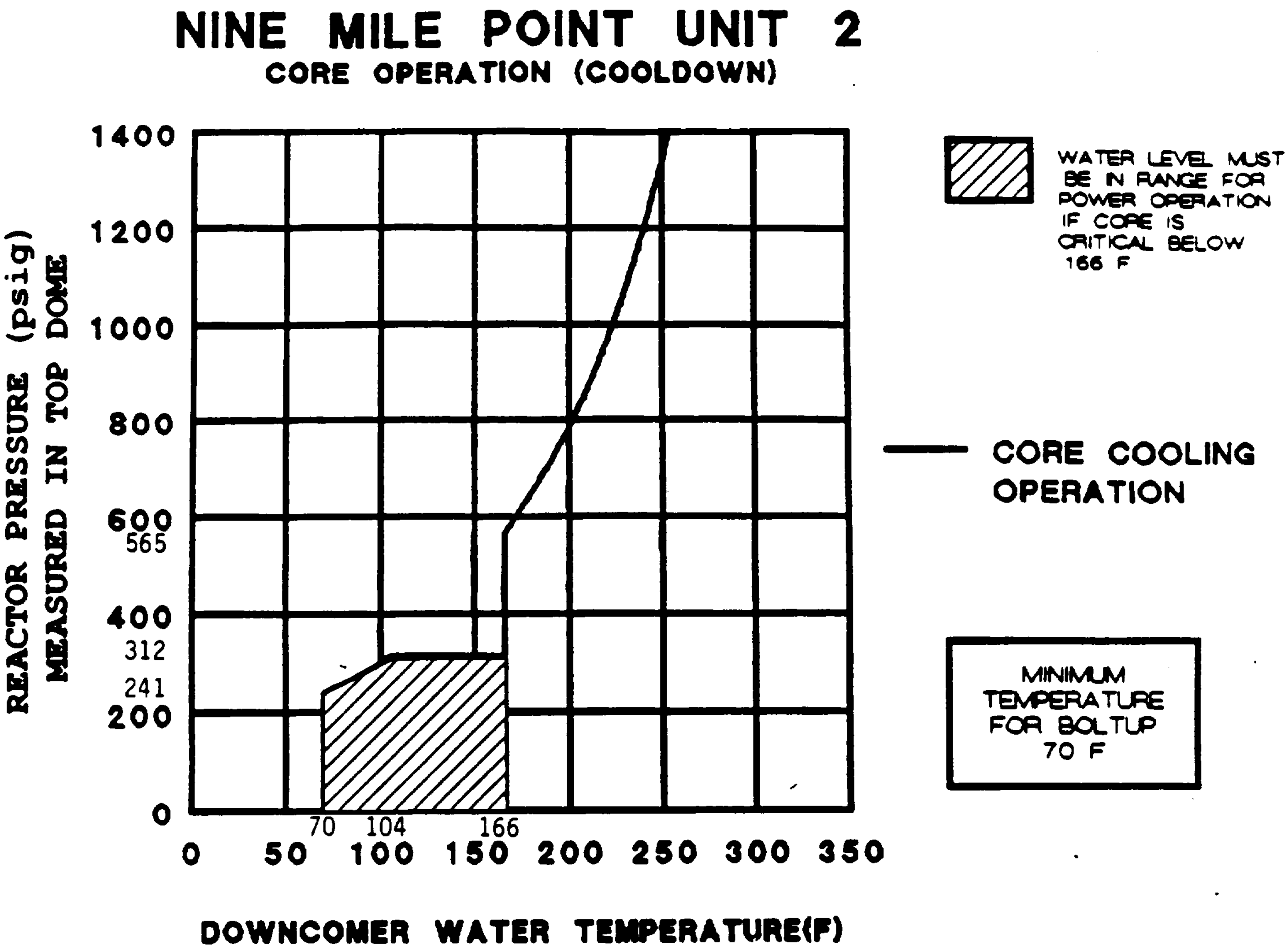

FIGURE 3.4.6.1-5 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING CORE OPERATION (CORE CRITICAL) (COOLDOWN AT A COOLING RATE ≤ 100 F/HR) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

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.

REACTOR COOLANT SYSTEM

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.

* 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-410

INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operation," Niagara Mohawk Power Corporation (the licensee) proposed to revise the pressure/temperature (P/T) limits in the Nine Mile Point Nuclear Station, Unit No. 2 (NMP-2) Technical Specifications, Section 3/4.4.6 and the associated Bases. The proposed revision was documented in letters from the licensee dated November 28, 1988, March 21, 1990, and November 13, 1990. The proposed revision would also make the proposed P/T limits applicable for 12.8 effective full power years (EFPY). The proposed P/T limits were developed based on Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

BACKGROUND

The P/T limits are among the limiting conditions of operations in the Technical Specifications. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted nil-ductility transition reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of

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unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the NMP-2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the materials with the highest ART at 12.8 EFPY were plates C3147-1 and C3147-2 with 0.11% copper (Cu), 0.63% nickel (Ni), and an initial RT_{ndt} (nil-ductility transition reference temperature) of 0°F.

The licensee has not removed any surveillance capsules from the NMP-2 reactor vessel. All surveillance capsules contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline materials, plates C3147-1 and C3147-2, the staff calculated the ART to be 56.93°F at 1/4T (T - reactor vessel beltline thickness) and 45.54°F at 3/4T for 12.8 EFPY. The staff used a neutron fluence of 4.6E17 n/cm^2 at 1/4T and 2.0E17 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Revision 2, because no surveillance capsules have been removed from the NMP-2 reactor vessel.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 56.93°F at 1/4T and 45.54°F at 3/4T at 12.8 EFPY for the same limiting plates. Substituting the ART of 56.93°F into equations in Standard Review Plan Section 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, criticality, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest initial Charpy USE is plate C3147-1 with 74 ft-lb. Using Figure 2 of RG 1.,99, Revision 2, the staff calculated that the end of life USE will be 64.4 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

SUMMARY

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 12.8 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the NMP-2 Technical Specifications.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of the facility components located within the restricted areas as defined in 10 CFR Part 20. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 11, 1991

PRINCIPAL CONTRIBUTOR:

J. Tsao