November , 1998

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
P.O. Box 63
Lycoming, NY 13093

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 (TAC NO. MA1413)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No. 164 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated June 19, 1998, as supplemented by letter dated November 6, 1998.

This amendment changes TS 3.2.2, "Minimum Reactor Vessel Temperature for Pressurization," and the associated TS Bases to update pressure-temperature operating curves and tables for continued plant operation up to 28 effective full-power years.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

J Cushing for D Hood

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

Docket No. 50-220

Enclosures: 1. Amendment No. 164 to DRP-63
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 25, 1998

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
P.O. Box 63
Lycoming, NY 13093

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Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No. 164 to DRP-63
2. Safety Evaluation

cc w/encls: See next page

cc:

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DATED: November 25, 1998

AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-63 NINE MILE POINT NUCLEAR POWER STATION UNIT NO. 1

Docket File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. DPR-63

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated June 19, 1998, as supplemented by letter dated November 6, 1998, 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. 164 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 before core operation exceeds 18 effective full-power years.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, 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: November 25, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 164

TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Replace the following page of the Appendix A Technical Specifications with the attached page.

Remove	Insert
84	84
85	85
86	86
87	87
88	88
89	89
90	90
91	91
92	92
93	93
94	94
---	94a (new)
---	94b (new)
---	94c (new)
---	94d (new)
95	95

LIMITING CONDITION FOR OPERATION

c. During leakage and hydrostatic testing, the reactor vessel temperature and pressure shall satisfy the requirements of Figures 3.2.2.e, 3.2.2.f, or 3.2.2.g, as appropriate, if the core is not critical. During reactor vessel heatup and cooldown for the purpose of leakage and hydrostatic testing, the reactor vessel temperature and pressure shall satisfy the requirements of Figures 3.2.2.a and 3.2.2.b for non-critical heatup and cooldown, respectively.

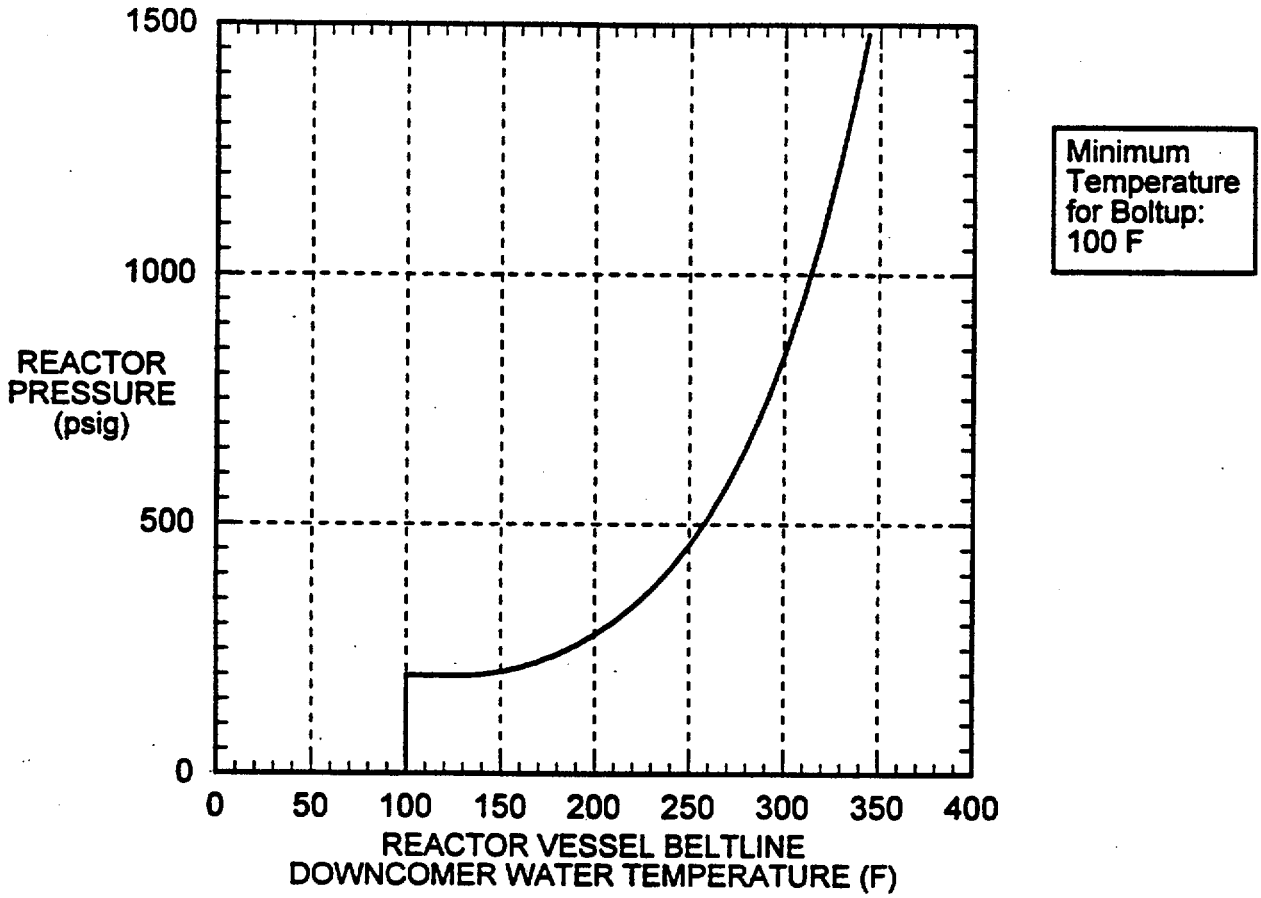
d. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head are equal to or greater than 100°F.

SURVEILLANCE REQUIREMENT

In order to generate additional plant-specific data, a capsule containing irradiated and unirradiated material will be re-inserted at the B capsule location. Re-insertion capsules have already been installed at the A and C locations. A prime (') is used to indicate a re-insertion capsule. The withdrawal schedule for the re-insertion capsules is as follows:

Fourth capsule (A') - 24 EFPY
Fifth capsule (C') - 32 EFPY
Sixth capsule (B') - 40 EFPY

HEATUP - CORE NOT CRITICAL



(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this figure)

FIGURE 3.2.2.a

MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING HEATUP AND LOW-POWER PHYSICS TESTS (CORE NOT CRITICAL) (HEATING RATE ≤ 100°F/HR) FOR UP TO 28 EFFECTIVE FULL POWER YEARS OF OPERATION

LIMIT FOR NON-CRITICAL OPERATION
HEATUP AT UP TO 100°F/HR

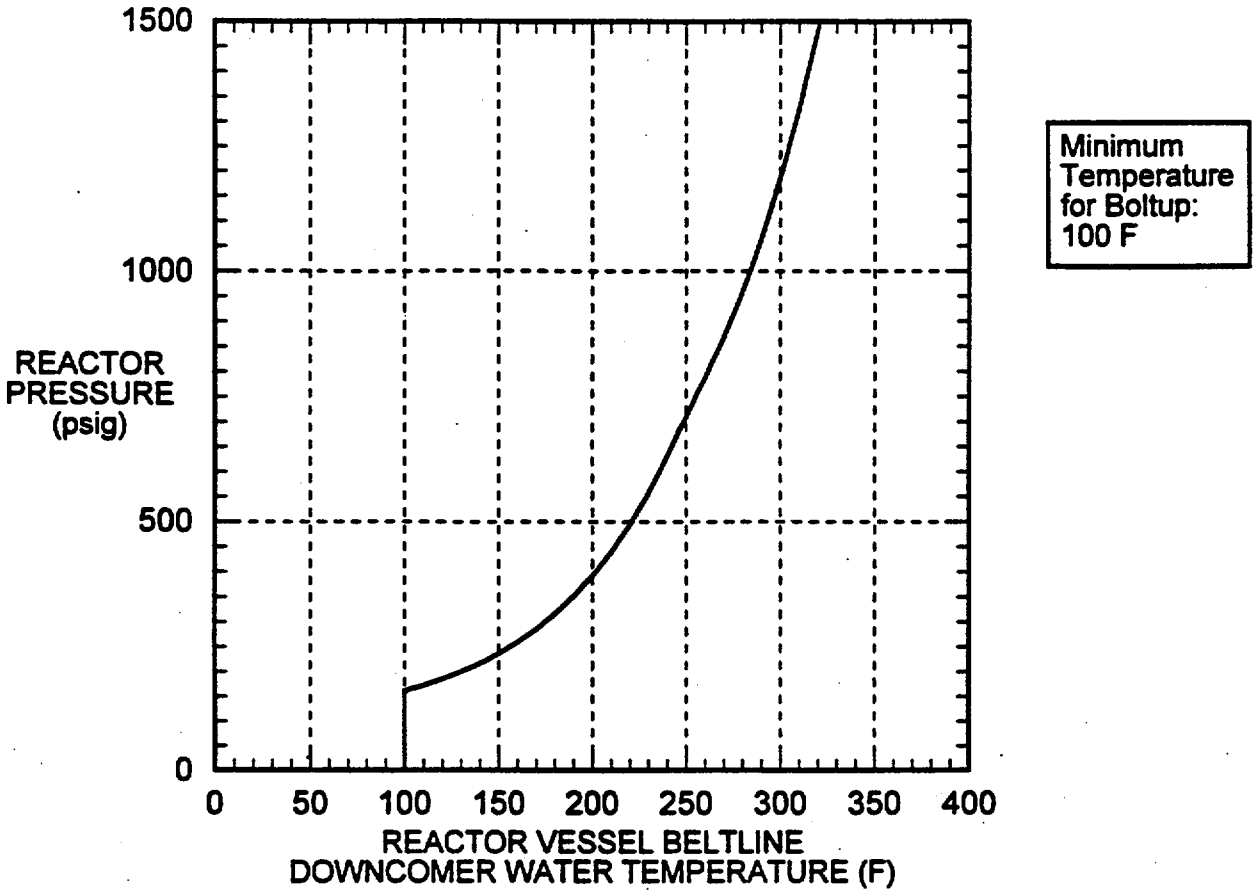
REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)
197	100
197	110
197	120
197	130
199	140
205	150
213	160
225	170
239	180
257	190
279	200
304	210
334	220
369	230
410	240
458	250
513	260
577	270
651	280
737	290
835	300
949	310
1079	320

(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this table)

TABLE 3.2.2.a

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING HEAT-UP (CORE NOT CRITICAL) (HEATING RATE ≤ 100°F/HR) FOR UP TO TWENTY EIGHT EFFECTIVE FULL POWER YEARS OF CORE OPERATION

COOLDOWN - CORE NOT CRITICAL



(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this figure)

FIGURE 3.2.2.b

MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING COOLDOWN AND LOW-POWER PHYSICS TESTS (CORE NOT CRITICAL) (COOLING RATE ≤ 100°F/HR) FOR UP TO 28 EFFECTIVE FULL POWER YEARS OF OPERATION

LIMIT FOR NON-CRITICAL OPERATION COOLDOWN AT UP TO 100°F/HR

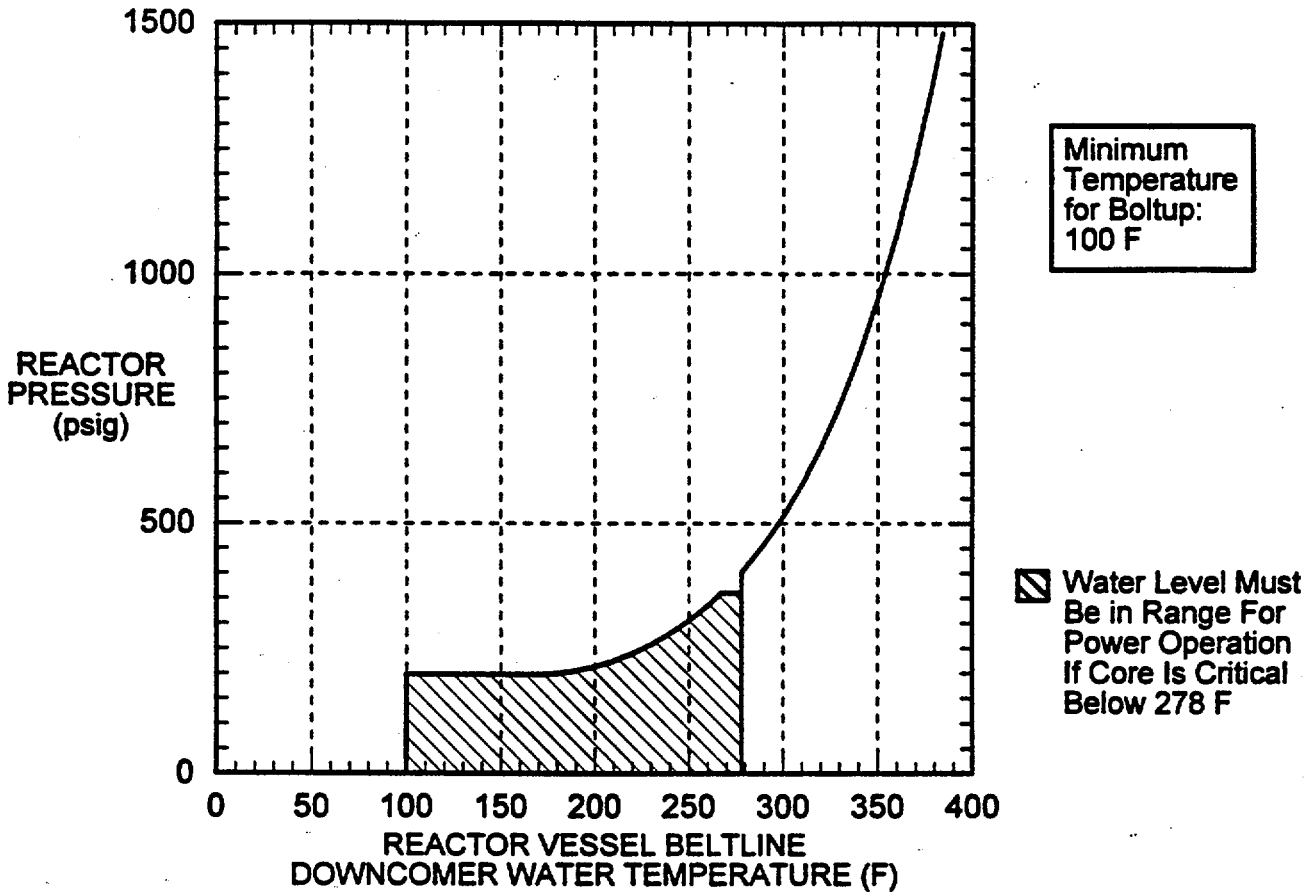
REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)
160	100
171	110
184	120
199	130
216	140
235	150
258	160
284	170
315	180
350	190
391	200
438	210
493	220
556	230
630	240
708	250
786	260
866	270
957	280
1062	290

(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this table)

TABLE 3.2.2.b

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING COOLDOWN (CORE NOT CRITICAL) (COOLING RATE ≤ 100°F/HR) FOR UP TO TWENTY EIGHT EFFECTIVE FULL POWER YEARS OF CORE OPERATION

HEATUP - CORE CRITICAL



(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this figure)

FIGURE 3.2.2.c

MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING CORE OPERATION (CORE CRITICAL) (HEATING RATE ≤ 100°F/HR) FOR UP TO 28 EFFECTIVE FULL POWER YEARS OF OPERATION

LIMIT FOR POWER OPERATION (CORE CRITICAL)
HEATUP AT UP TO 100°F/HR

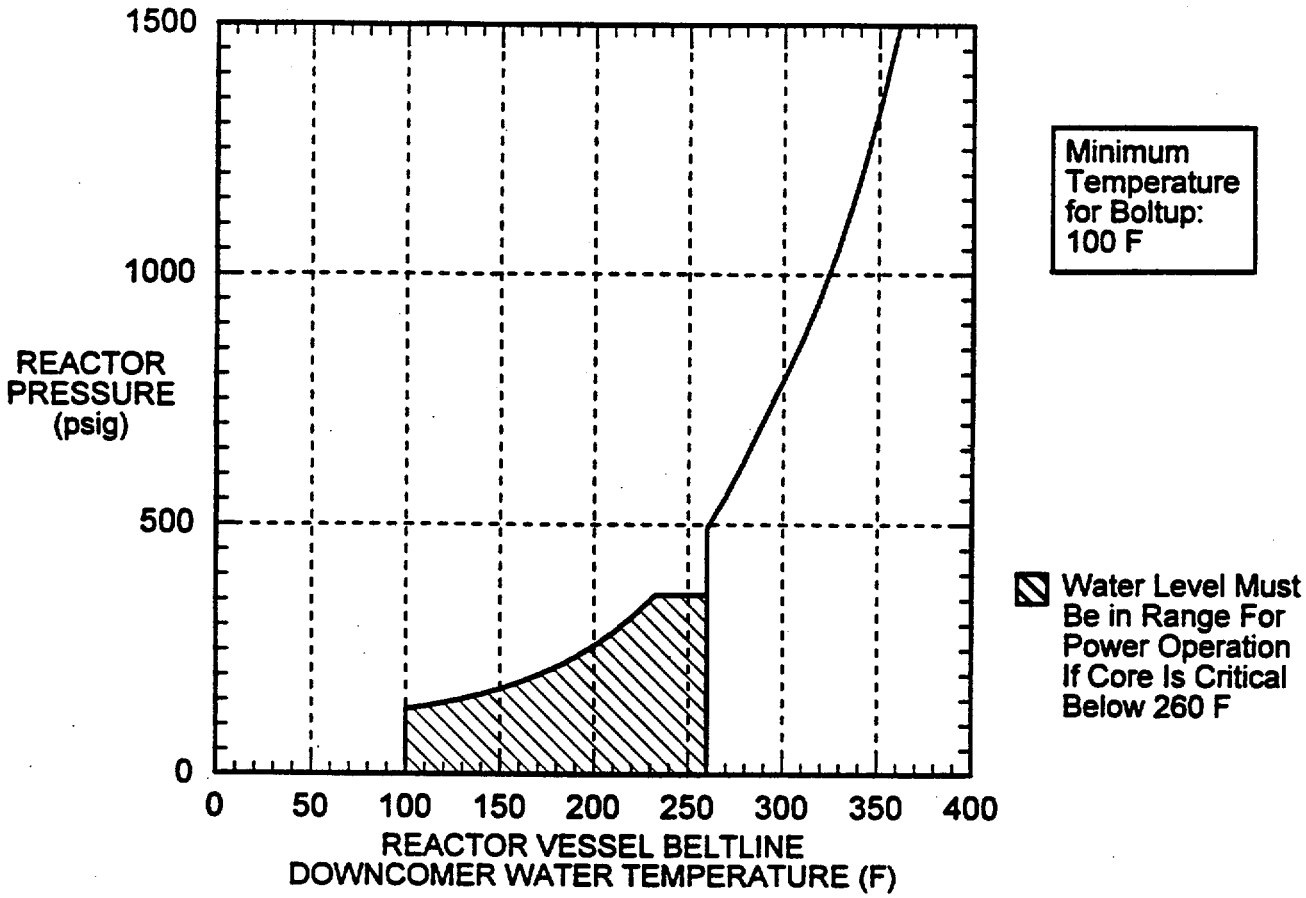
REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)
197	100
197	110
197	120
197	130
197	140
197	150
197	160
197	170
199	180
205	190
213	200
225	210
239	220
257	230
279	240
304	250
334	260
360	268
360	270
360	277
402	278[a]
410	280
458	290
513	300
577	310
651	320
737	330
835	340
949	350
1079	360

(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
([a]water level must be in range for power operation if core is critical below 278 F)
(instrument uncertainties have been included in this table)

TABLE 3.2.2.c

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING HEATUP (CORE CRITICAL) (HEATING RATE ≤ 100°F/HR) FOR UP TO TWENTY EIGHT EFFECTIVE FULL POWER YEARS OF CORE OPERATION

COOLDOWN - CORE CRITICAL



(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this figure)

FIGURE 3.2.2.d

MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING CORE OPERATION (CORE CRITICAL) (COOLING RATE ≤100°F/HR) FOR UP TO 28 EFFECTIVE FULL POWER YEARS OF OPERATION

LIMIT FOR POWER OPERATION (CORE CRITICAL) COOLDOWN AT UP TO 100°F/HR

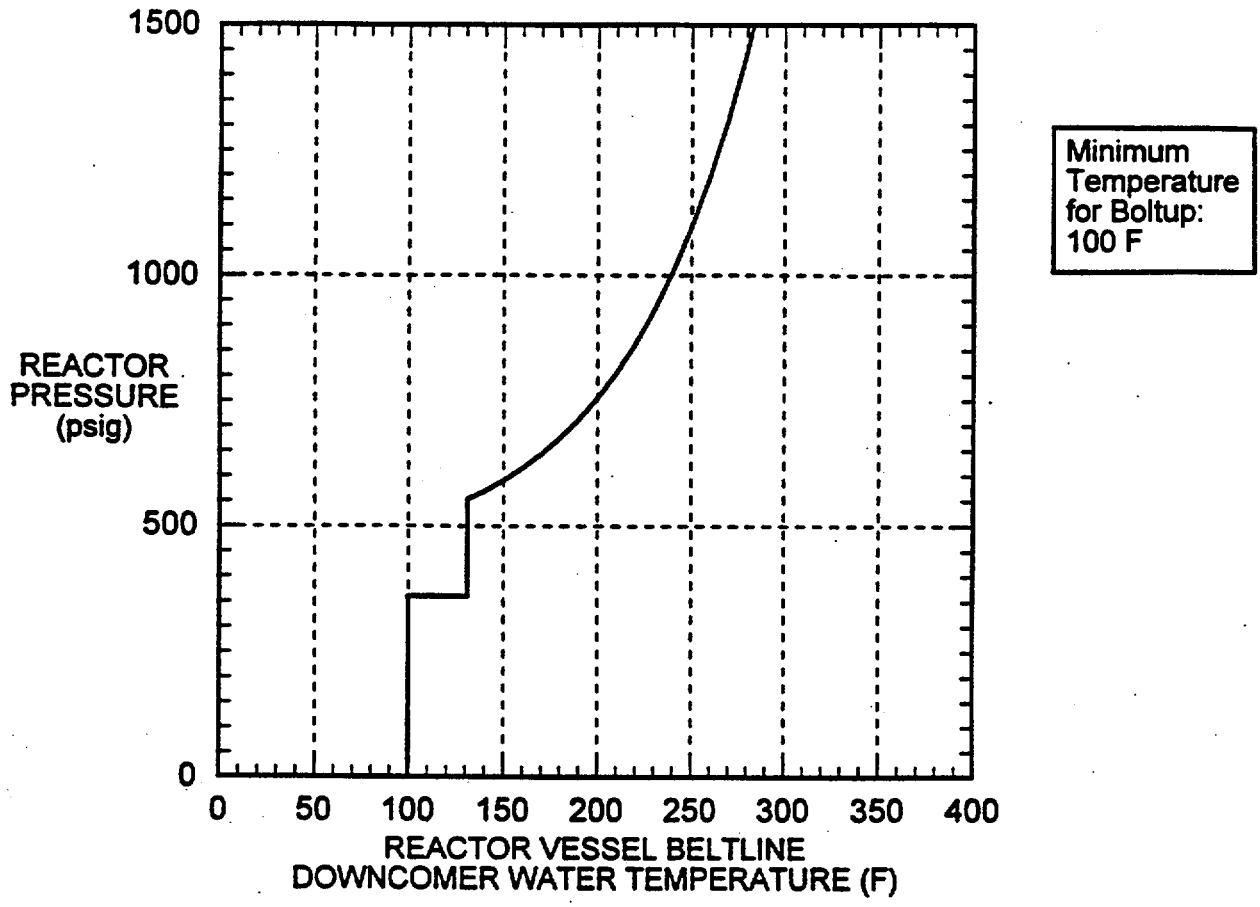
REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)
130	100
136	110
143	120
151	130
160	140
171	150
184	160
199	170
216	180
235	190
258	200
284	210
315	220
350	230
360	233
360	240
360	250
360	259
493	260*
556	270
630	280
708	290
786	300
866	310
957	320
1062	330

(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(*water level must be in range for power operation if core is critical below 260 F)
(instrument uncertainties have been included in this table)

TABLE 3.2.2.d

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING COOLDOWN (CORE CRITICAL) (COOLING RATE ≤ 100°F/HR) FOR UP TO TWENTY EIGHT EFFECTIVE FULL POWER YEARS OF CORE OPERATION

LEAK/HYDRO TEST - CORE NOT CRITICAL



(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this figure)

FIGURE 3.2.2.e

MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING IN-SERVICE HYDROSTATIC TESTING AND LEAK TESTING (CORE NOT CRITICAL) FOR UP TO 28 EFFECTIVE FULL POWER YEARS OF OPERATION

LIMIT FOR IN-SERVICE TEST
(CORE NOT CRITICAL, FUEL IN VESSEL)

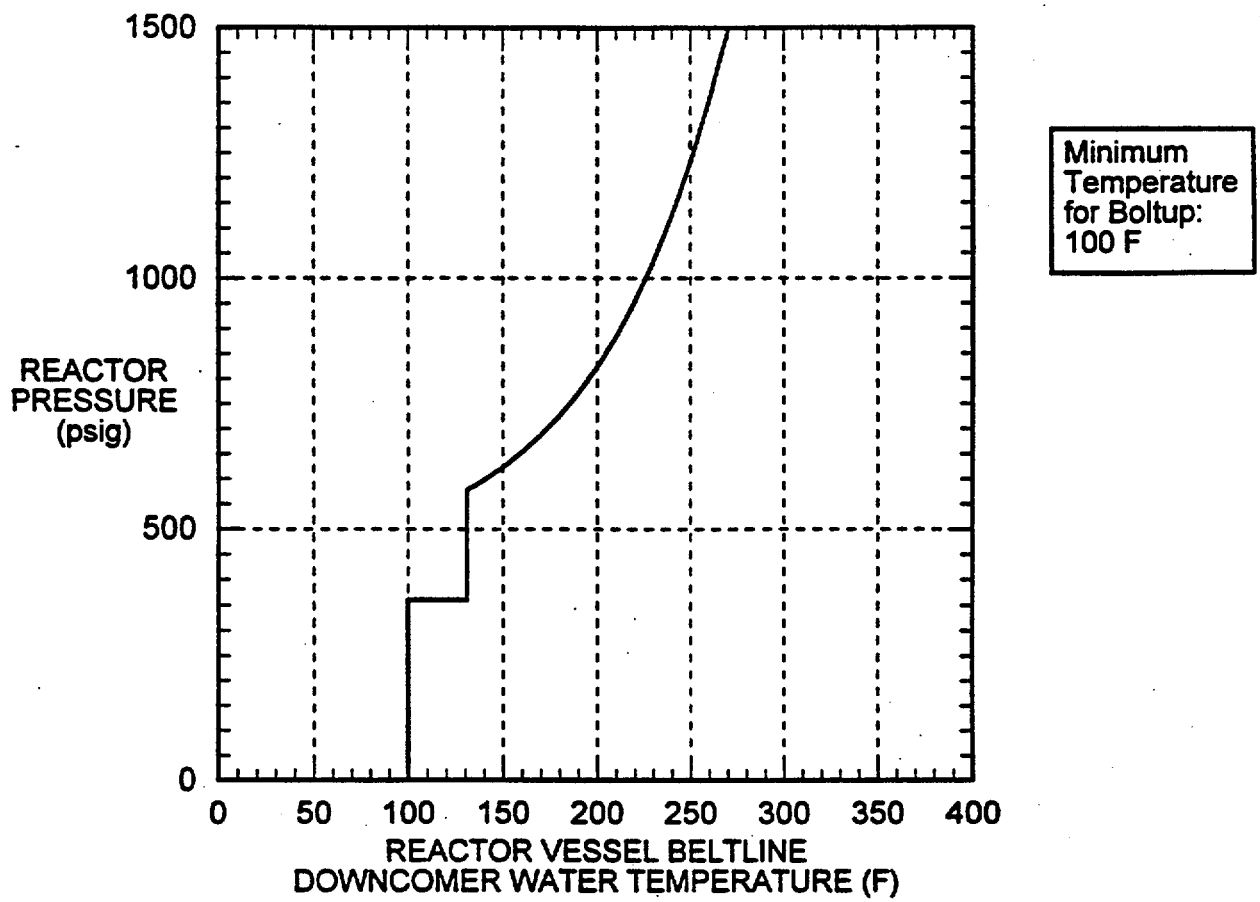
REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)
360	100
360	110
360	120
360	130
569	140
590	150
614	160
642	170
675	180
712	190
755	200
805	210
862	220
929	230
1005	240
1032	243
1093	250
1195	260

(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this table)

TABLE 3.2.2.e

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING LEAK/HYDROSTATIC TESTING (CORE NOT CRITICAL) FOR UP TO TWENTY EIGHT EFFECTIVE FULL POWER YEARS OF CORE OPERATION

LEAK/HYDRO TEST - CORE NOT CRITICAL



(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this figure)

FIGURE 3.2.2.f

MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING IN-SERVICE HYDROSTATIC TESTING AND LEAK TESTING (CORE NOT CRITICAL) FOR UP TO 20 EFFECTIVE FULL POWER YEARS OF OPERATION

LIMIT FOR IN-SERVICE TEST
(CORE NOT CRITICAL, FUEL IN VESSEL)

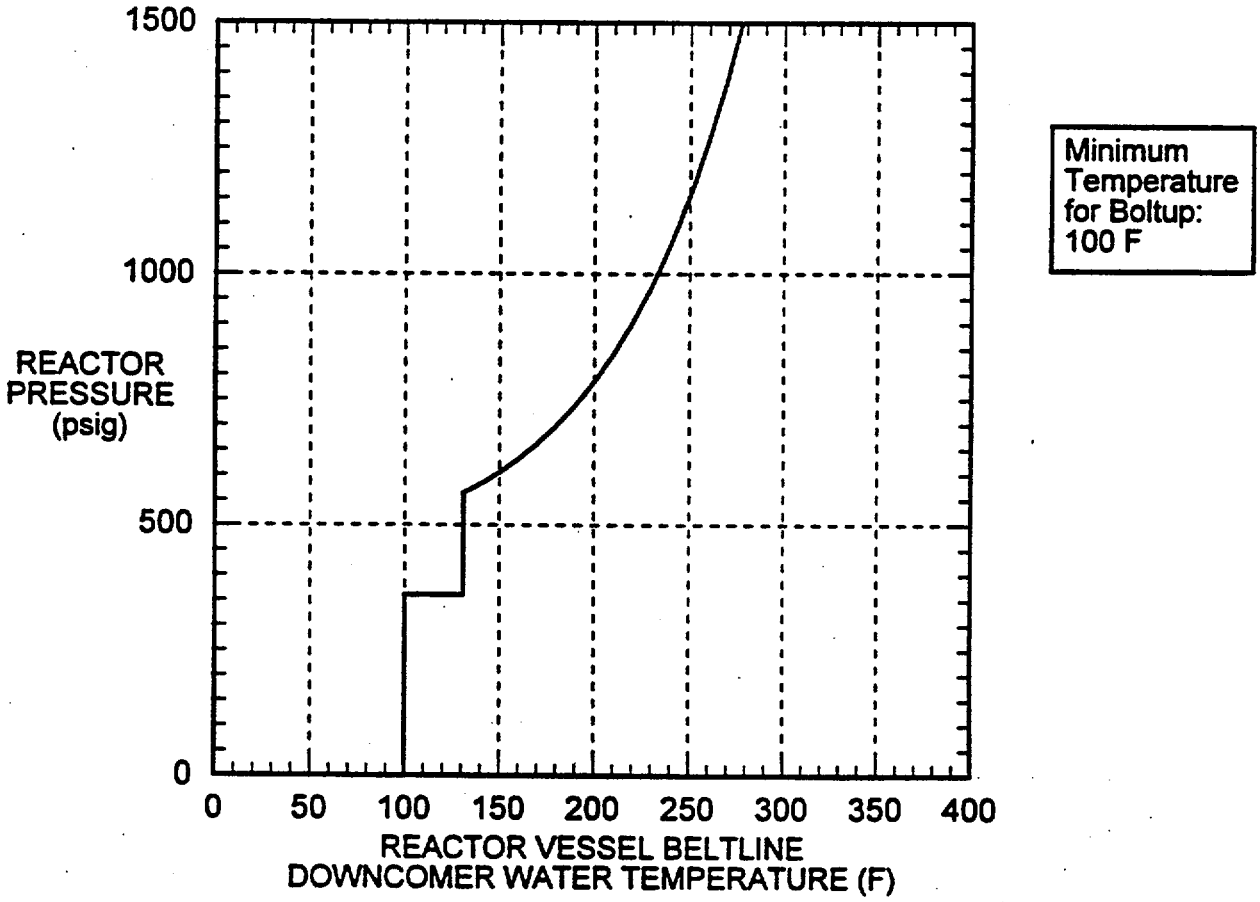
REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)
360	100
360	110
360	120
360	130
597	140
622	150
652	160
685	170
724	180
769	190
821	200
881	210
951	220
1031	230
1123	240
1229	250

(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this table)

TABLE 3.2.2.f

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING LEAK/HYDROSTATIC TESTING (CORE NOT CRITICAL) FOR UP TO TWENTY EFFECTIVE FULL POWER YEARS OF CORE OPERATION

LEAK/HYDRO TEST - CORE NOT CRITICAL



(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this figure)

FIGURE 3.2.2.g

MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING IN-SERVICE HYDROSTATIC TESTING AND LEAK TESTING (CORE NOT CRITICAL) FOR UP TO 24 EFFECTIVE FULL POWER YEARS OF OPERATION

LIMIT FOR IN-SERVICE TEST
(CORE NOT CRITICAL, FUEL IN VESSEL)

REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)
360	100
360	110
360	120
360	130
582	140
604	150
631	160
661	170
697	180
737	190
785	200
839	210
902	220
974	230
1033	237
1058	240
1154	250

(reactor vessel beltline downcomer water temperature is measured at recirculation loop suction)
(instrument uncertainties have been included in this table)

TABLE 3.2.2.g

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING LEAK/HYDROSTATIC TESTING (CORE NOT CRITICAL) FOR UP TO TWENTY FOUR EFFECTIVE FULL POWER YEARS OF CORE OPERATION

BASES FOR 3.2.2 AND 4.2.2 MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION

Figures 3.2.2.a, 3.2.2.b, 3.2.2.c, and 3.2.2.d are plots of pressure versus temperature for heatup and cooldown rates of up to 100°F/hr. maximum (Specification 3.2.1). Figures 3.2.2.e, 3.2.2.f, and 3.2.2.g are plots of pressure versus temperature for leakage and hydrostatic testing. When the minimum temperature for leakage and hydrostatic testing is reached, a thermal soak shall be performed to ensure that the thermal gradient across the vessel wall is negligible. These curves are based on calculations of stress intensity factors according to Appendix G of Section III of the ASME Boiler and Pressure Vessel Code 1980 Edition with Winter 1982 Addenda. In addition, temperature shifts due to fast neutron fluence at twenty-eight effective full power years of operation were incorporated into the figures. These shifts were calculated using the procedure presented in Regulatory Guide 1.99, Revision 2. Reactor vessel flange/reactor head flange boltup is governed by other criteria as stated in Specification 3.2.2.d. The pressure readings on the figures have been adjusted to account for instrument uncertainties and to reflect the calculated elevation head difference between the pressure sensing instrument locations and the pressure sensitive area of the core beltline region. The temperature readings on the figures have been adjusted to account for instrument uncertainties.

The reactor vessel head flange and vessel flange in combination with the double "O" ring type seal are designed to provide a leak-tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flanges. Both the head and vessel flanges have an NDT temperature of 40°F and they are not subject to any appreciable neutron radiation exposure. Therefore, the minimum vessel flange and head flange temperature for bolting is established at 40°F + 60°F or 100°F.

Figures 3.2.2.a, 3.2.2.b, 3.2.2.c, 3.2.2.d, 3.2.2.e, 3.2.2.f and 3.2.2.g have incorporated a temperature shift due to the calculated fast neutron fluence. The neutron flux at the vessel wall is calculated from core physics data and has been determined using flux monitors installed inside the vessel. The curves, except for 3.2.2.f and 3.2.2.g, are applicable for up to twenty-eight effective full power years of operation. Curves 3.2.2.f and 3.2.2.g are applicable for up to twenty and twenty-four effective full power years, respectively.

Vessel material surveillance samples are located within the core region to permit periodic monitoring of exposure and changes in material properties. The material sample program conforms with ASTM E185-66 except for the material withdrawal schedule which is specified in Specification 4.2.2.b.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated June 19, 1998, as supplemented November 6, 1998, Niagara Mohawk Power Corporation (NMPC and the licensee) proposed a license amendment to change the Technical Specifications (TSs) for Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The proposed amendment would change TS 3.2.2, "Minimum Reactor Vessel Temperature for Pressurization," and associated TS Bases 3/4.2.2 to update TS 3.2.2 tables and figures ("pressure-temperature (P-T) curves") that limit the minimum reactor vessel temperature for a given pressure. Because the limits are based upon the number of effective full-power years (EFPYs) of core operation and the existing tables and figures are valid for up to 18 EFPYs, the proposed amendment would substitute new tables and figures that are valid for 20, 24, and 28 EFPYs. The word "leakage" would be added to clarify that this TS applies to both leakage and hydrostatic tests.

By letter dated November 6, 1998, NMPC provided additional information in support of the application for amendment. The additional information does not affect the Commission's finding of no significant hazards consideration that was issued in a Federal Register notice (63 FR 40557, July 29, 1998).

2.0 EVALUATION

The NRC staff evaluated the P-T limits based upon the following NRC regulations and guidance: Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"; Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated July 12, 1998; GL 92-01, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," Revision 1, dated March 6, 1992; Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, dated May 1988; and Standard Review Plan (SRP or NUREG-0800) Section 5.3.2, "Pressure-Temperature Limits." Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel shall be at least as conservative as those obtained by Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). GL 88-11 specifies that licensees should use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation upon the adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor and a fluence factor.

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The chemistry factor may be calculated using credible surveillance data, obtained by a licensee's surveillance program, as specified by Position 2 of RG 1.99, Revision 2. If credible surveillance data are not available, the calculated chemistry factor depends upon the amount of copper and nickel in the vessel material as specified in Table 1 of RG 1.99, Revision 2. GL 92-01 provides for the submittal of reactor vessel materials data by licensees, which the NRC staff uses in the review of the P-T limits submittals.

SRP 5.3.2 provides guidance on calculation of P-T limits using the linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress. The postulated defects have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 times the beltline thickness. The critical locations in the vessel for this methodology are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

For the NMP-1 reactor vessel, NMPC determined that the most limiting material at the 1/4T and 3/4T locations is plate G-307-4. This plate was fabricated using plate heat P2076. NMPC calculated an ART of 167.7 °F at the 1/4T location and 136.9 °F at the 3/4T location at 28 EFPY. The neutron fluence used in the ART calculation was 1.76×10^{18} n/cm^2 at the 1/4T location and 7.48×10^{17} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting plate was 40 °F. The margin term used in calculating the ART for the limiting plate was 34 °F.

The NRC staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the NRC staff verified that NMPC's limiting material for the NMP-1 reactor vessel is plate G-307-4. The NRC staff's calculated ART value for the limiting material agreed with NMPC's calculated ART value.

Substituting the ART values for the NMP-1 vessel in SRP 5.3.2 equations, the NRC staff verified that the proposed P-T limits satisfy the requirements in paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based upon the reference temperature for the flange material. Table 1 of Section IV.A.2 of Appendix G shows 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 shall 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. Based on the flange RT_{NDT} of 40 °F for NMP-1, the NRC staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation and inservice leak and hydrostatic testing.

Accordingly, on the basis of its review of NMPC's submittal and its own independent analysis which verified NMPC's proposed P-T limits, the NRC staff concludes that the proposed P-T limits for heatup and cooldown for non-critical operations; heatup and cooldown for critical operations;

and non-critical leakage/hydrotest, are acceptable for use up to 28 EFPY because the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11. Therefore, the proposed changes to the NMP1 TS based upon these P-T limits are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 40557). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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