ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Proposed Changes to Technical Specifications (Appendix A)

Existing pages 79 through 82 will be replaced with the attached revised pages. These pages have been retyped in their entirely with marginal markings to indicate changes to the text.

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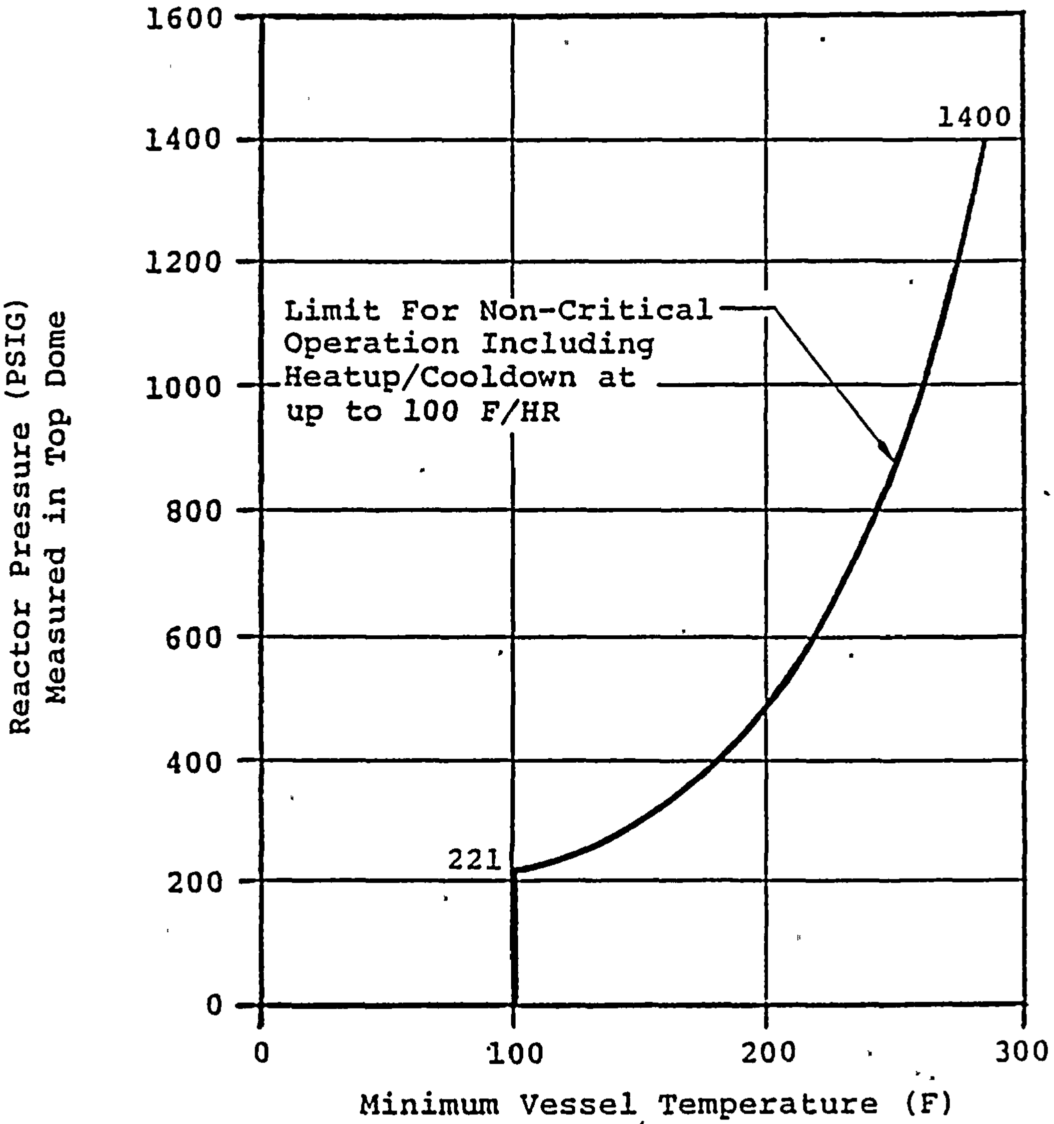

FIGURE 3.2.2.a

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING HEATUP OR COOLDOWN (REACTOR NOT CRITICAL) (HEATING OR COOLING RATE $\leq$ 100 F/HR) FOR UP TO THIRTEEN EFFECTIVE FULL POWER YEARS OF CORE OPERATION

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

PRESSURE (psig)	TEMPERATURE (F)
221	100
300	148
350	167
400	182
450	194
500	204
550	213
600	221
650	228
700	235
750	241
800	247
850	252
900	256
950	261
1000	265
1050	269
1100	272
1150	276
1200	279
1300	285
1400	291

TABLE 3.2.2.a

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING
HEAT-UP OR COOLDOWN (REACTOR NOT CRITICAL)
(HEATING OR COOLING RATE 100F/HR)
FOR UP TO THIRTEEN EFFECTIVE FULL
POWER YEARS OF CORE OPERATION

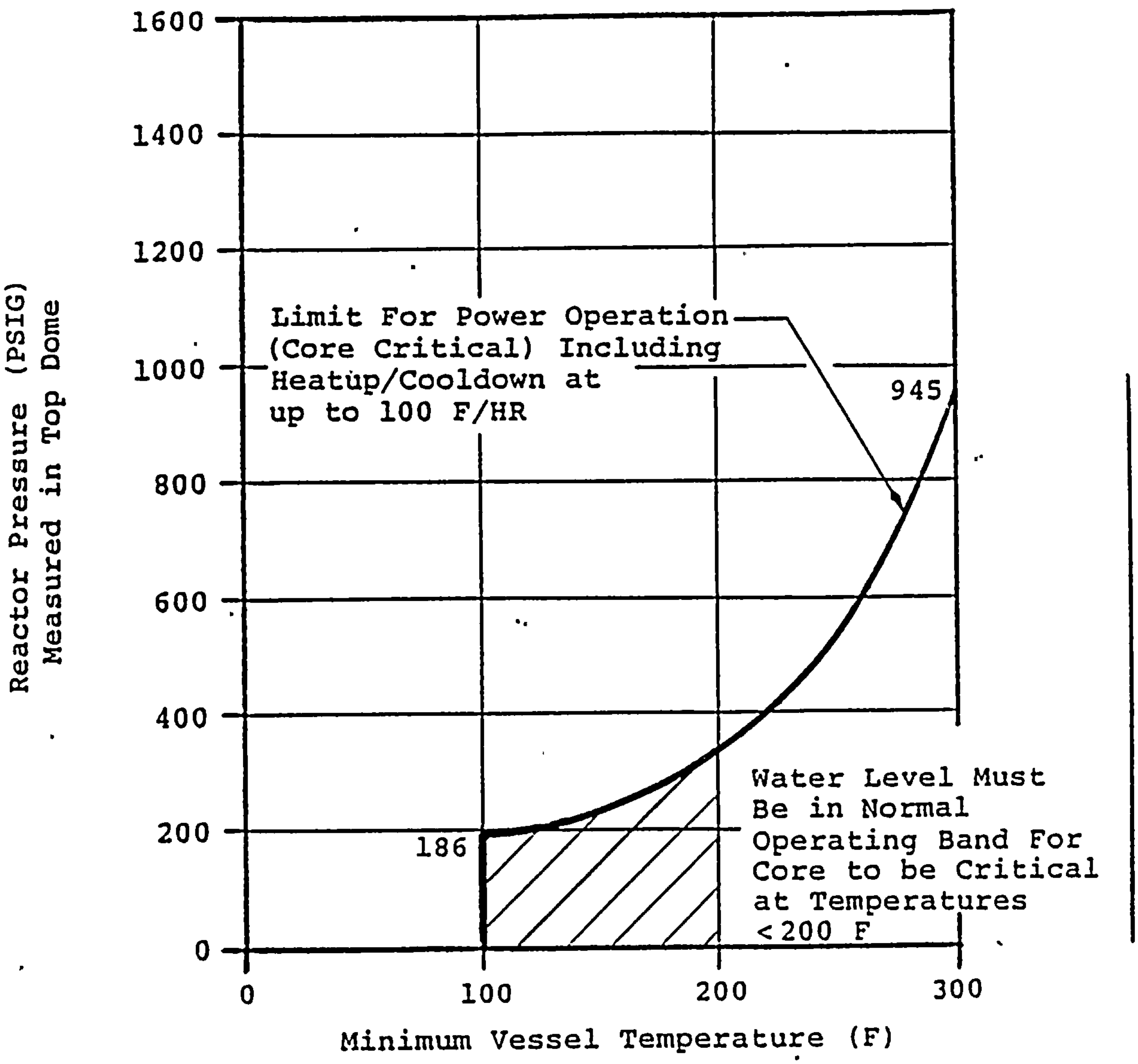

FIGURE 3.2.2.b

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING HEATUP OR COOLDOWN (REACTOR CRITICAL) (HEATING OR COOLING RATE ≤ 100 F/HR) FOR UP TO THIRTEEN EFFECTIVE FULL POWER YEARS OF CORE OPERATION

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

PRESSURE (psig)	TEMPERATURE (F)
186	100
250	162
300	188
350	207
400	222
450	234
500	244
550	253
600	261
650	269
700	275
750	281
800	287
850	292
900	296
950	301
1000	305
1050	308
1100	312
1150	316
1200	319
1300	325
1400	331

TABLE 3.2.2.b.

MINIMUM TEMPERATURE FOR PRESSURIZATION DURING
HEAT-UP OR COOLDOWN (REACTOR CRITICAL)
(HEATING OR COOLING RATE 100F/HR)
FOR UP TO THIRTEEN EFFECTIVE FULL
POWER YEARS OF CORE OPERATION

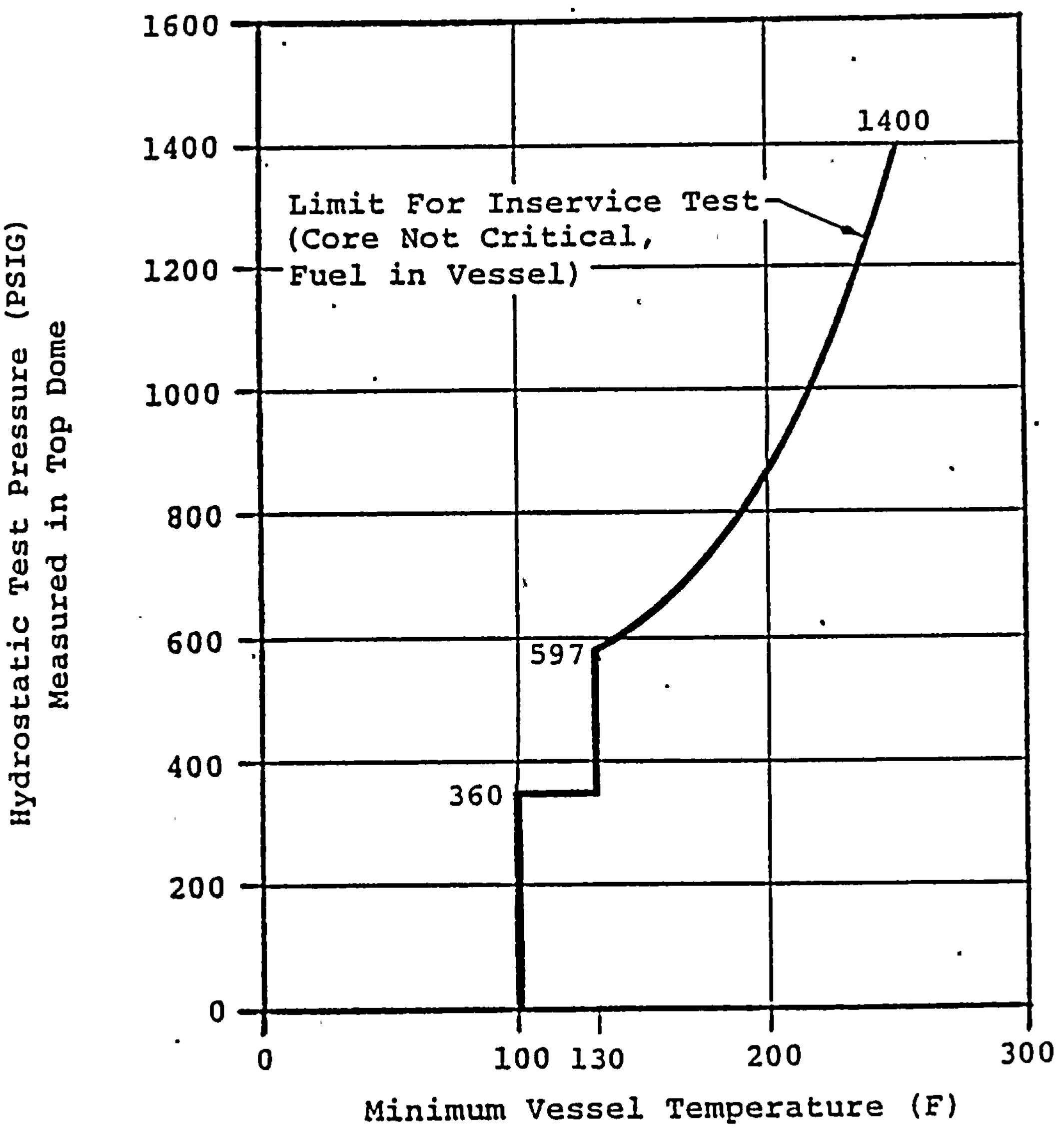

FIGURE 3.2.2.c

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

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

PRESSURE (psig)	TEMPERATURE (F)
360	100-130
597	130
700	164
800	186
900	203
1000	216
1050	222
1100	228
1150	233
1200	237
1300	245
1400	253

TABLE 3.2.2.c

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

Figures 3.2.2.a and 3.2.2.b are plots of pressure versus temperature for a heat-up and cool down rate of 100F/hr. maximum. (Specification 3.2.1). Figure 3.2.2.c is a plot of pressure versus temperature for hydrostatic testing. 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 integrated neutron flux at thirteen effective full power years of operation were incorporated into the figures. These shifts were calculated from the formula presented in Regulatory Guide 1.99, proposed Revision 2. These curves are applicable to the beltline region at low and elevated temperatures and the vessel flange at intermediate temperatures. 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 reflect the calculated elevation head difference between the pressure sensing instrument locations and the pressure sensitive area of the core beltline region.

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 flange. Both the head and vessel and flange have a NDT temperature of 40F and they are not subject to any appreciable neutron radiation exposure. Therefore, the minimum vessel head and head flange temperature for bolting the head flange and vessel flange is established as 40 + 60F or 100F.

Figures 3.2.2.a., 3.2.2.b. and 3.2.2.c. have incorporated a temperature shift due to the calculated integrated neutron flux. The integrated neutron flux at the vessel wall is calculated from core physics data and has been measured using flux monitors installed inside the vessel. The curves are applicable for up to thirteen effective full power years of operation.

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

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION

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Supporting Information and No Significant Hazards Considerations Analysis

The proposed revision to Figures 3.2.2a, b and c reflect a change in the limits for minimum reactor vessel temperature for pressurization. These limits are based on measured nil-ductility temperature shifts of irradiated vessel material samples.

Three surveillance capsules were installed in the Nine Mile Point Unit 1 (NMP-1) reactor vessel prior to startup in 1969. Two surveillance capsules have been removed from the NMP-1 reactor vessel to date. The A Capsule was removed in 1979 after a vessel exposure of 5.8 effective full power years (efpy) and the C Capsule was removed in 1982 after a vessel exposure of 8.0 efpy. Due to the radial, circumferential, and axial position of the capsules, the exposure of the capsule specimens is calculated to be 87 percent of the maximum exposure of the reactor vessel shell at the 1/4t location. Thus, the exposures of the three original capsules lag the maximum reactor vessel exposure.

The full contents from the C Capsule were tested to determine tensile properties and reactor vessel base metal, weld metal, and heat affected zone (HAZ) Charpy impact nil-ductility temperatures (NDT). Six Charpy base metal specimens from the A Capsule were also tested to confirm the NDT shift of the base metal observed in the C Capsule specimens.

Based on these tests, pressure/temperature operating limits (curves) appropriate for up to thirteen effective full power years were established. These pressure/temperature curves were developed at the same time as the pressure/temperature curves for eleven effective full power years which were approved by the NRC staff in Amendment No. 85 to License No. DPR-63 on June 10, 1986. Since that time, no new data has been obtained which would affect the pressure/temperature limits for thirteen effective full power years. The Regulatory Guide 1.99 (proposed Revision 2) method for extrapolation was used except for the recommended addition of one standard deviation to the nil-ductility temperature shift.

In addition, the Bases for 3.2.2 and 4.2.2 have been revised to update the reference to the curves applicable for up to thirteen effective full power years of operation.

10CFR50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis, using the standards in 10CFR50.92, about the issue of no significant hazards consideration. Therefore, in accordance with 10CFR50.91, the following analysis has been performed:

The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment incorporates the results of testing of Nine Mile Point Unit 1 reactor vessel material surveillance specimens which have been irradiated during station operation. Testing of the material surveillance specimens was performed in accordance with 10CFR50 Appendix H.

Components of the reactor primary coolant system are operated so that no substantial pressure is imposed unless the reactor vessel materials are above nil-ductility transition temperature. The nil-ductility transition temperature increases as a function of the integrated neutron dose. The proposed amendment incorporates (1) the results of testing of irradiated Nine Mile Point Unit 1 reactor vessel material, (2) calculation of stress intensity factors according to Appendix G of Section III of the ASME Boiler and Pressure Code 1980 Edition with Winter 1982 Addenda and (3) the Regulatory Guide 1.99 (Proposed Revision 2) method for extrapolation with the exception of the recommended addition of one standard deviation to the nil-ductility shifts.

Operation of Nine Mile Point Unit 1 in accordance with the proposed pressure/temperature operating limits will preclude brittle failure of the reactor vessel material. Safety margins for brittle failure will be in accordance with those specified in 10CFR50 Appendix G and Appendix G of the ASME Code.

Therefore, the proposed amendment will not involve a significant increase in the probability of consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment incorporates pressure/temperature operating limits based on analysis of irradiated samples. No modification to the plant is required in order to implement the proposed amendment. Therefore, the proposed limits will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

Implementation of the proposed pressure/temperature operating limits will ensure station operations are conducted with the reactor vessel materials above nil-ductility transition temperature. Operation in accordance with the proposed pressure/temperature operating limits and proposed surveillance program will preclude brittle failure of the reactor vessel material, since safety margins specified in 10CFR50 Appendix G and the ASME Code Appendix G will be maintained.

As determined by the analysis above, this proposed amendment involves no significant hazards consideration.