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

December 3, 1998

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT NO. 2 (TAC NO. MA0869)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No. 84 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated February 5, 1998.

This amendment changes TSs to update the terminology and references to 10 CFR 50.55a(f) and (g) consistent with the 1989 edition of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, and consistent with the second 10-year interval of the Inservice Inspections and Inservice Testing Program Plans for NMP2.

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

Sincerely,

ORIGINAL SIGNED BY:

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

Docket No. 50-410

Enclosures: 1. Amendment No. 84 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

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

December 3, 1998

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cc w/encls: See next page

DATED: December 3, 1998

AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-69 NINE MILE POINT NUCLEAR POWER STATION UNIT NO. 2

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
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 February 5, 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.84 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

SBajwa

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: December 3, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 84

TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by Amendment number and contains vertical lines indicating the areas of change.

Remove	Insert
x	x
xiv	xiv
xx	xx
3/4 0-2	3/4 0-2
3/4 3-87	3/4 3-87
3/4 4-14	3/4 4-14
3/4 4-24	3/4 4-24
3/4 4-26	3/4 4-26
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 8-11	3/4 8-11
3/4 10-7	3/4 10-7
B3/4 4-5	B3/4 4-5
B3/4 10-1	B3/4 10-1
5-9	5-9

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50.55a(f), except where specific written relief has been granted by the Commission pursuant to 10CFR50.55a(f)(6)(i). Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable addenda shall be applicable as follows in these Technical Specifications:

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* Excludes sensors; sensor comparison shall be done in lieu of sensor calibration.

** Using sample gas containing:

a. One volume percent hydrogen, balance nitrogen.
b. Four volume percent hydrogen, balance nitrogen.

*** The CHANNEL CALIBRATION shall consist of position indication verification using the criteria specified for the Inservice Testing Program.

† The CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

†† Red, Green or other indication shall be verified as indicating valve position.

LIMITING CONDITIONS FOR OPERATION

e. With one or more of the required interlocks shown in Table 3.4.3.2-3 inoperable, restore the inoperable interlock to OPERABLE status within 7 days or isolate the affected heat exchanger(s) from the RCIC steam supply by closing and deenergizing heat exchanger valves 2RHS*MOV22A and 2RHS*MOV80A or 2RHS*MOV22B and 2RHS*MOV80B, as appropriate.

f. With any reactor coolant system leakage greater than the limit in 3.4.3.2.e above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The RCS leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment airborne particulate radioactivity at least once per 12 hours,

b. Monitoring the drywell floor drain tank and equipment drain tank fill rate at least once per 8 hours,

c. Monitoring the primary containment airborne gaseous radioactivity at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each RCS pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5, using the method and acceptance criteria specified in the Inservice Testing Program, and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and
b. Before returning the valve to service following maintenance, repair, or replacement work on the valve.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

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 system leakage 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 hydrostatic and system leakage 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 system leakage 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.

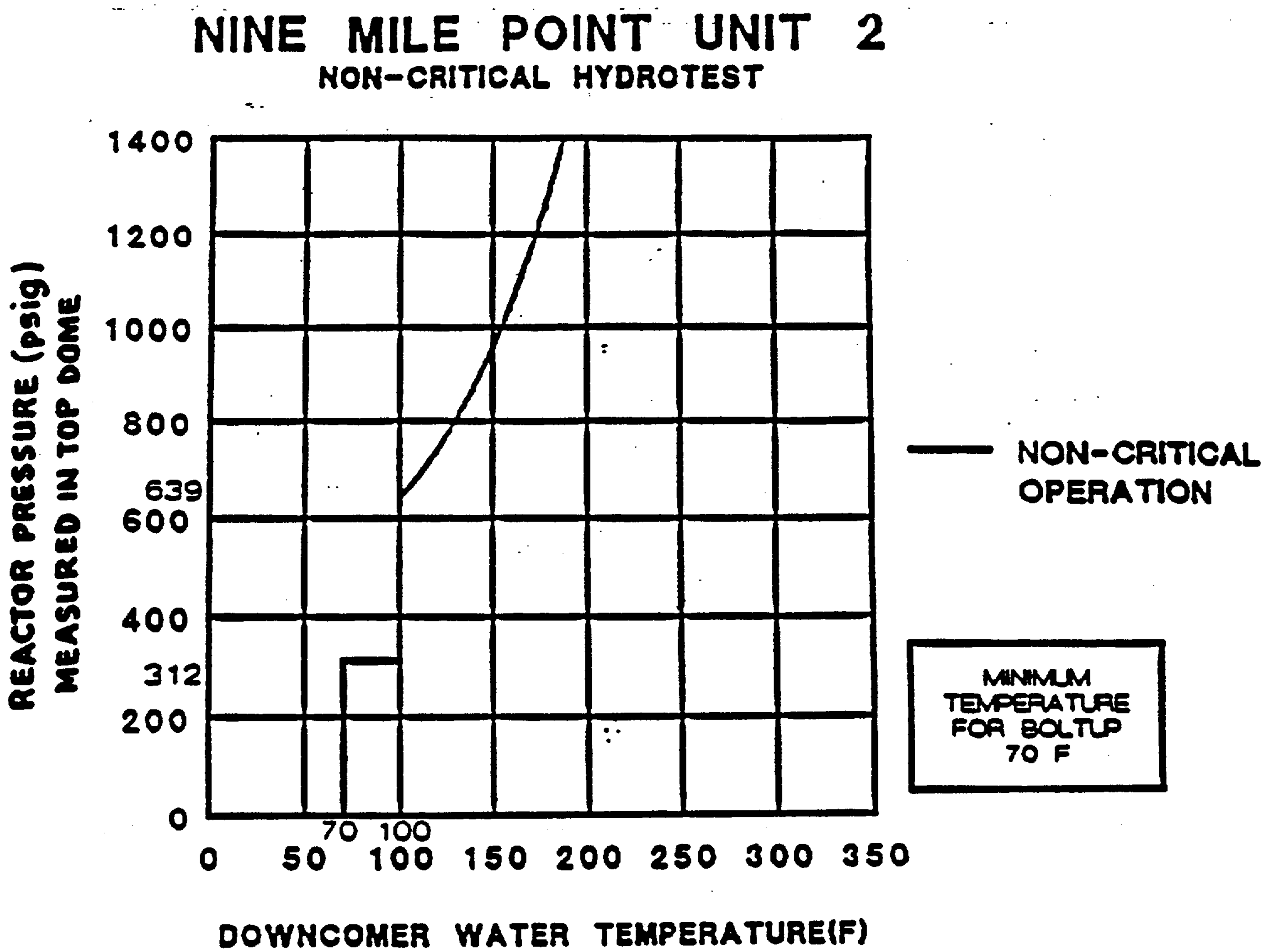

FIGURE 3.4.6.1-1 MINIMUM BELTLINE DOWNCOMER WATER TEMPERATURE FOR PRESSURIZATION DURING HYDROSTATIC TESTING AND SYSTEM LEAKAGE TESTING (REACTOR NOT CRITICAL) FOR UP TO 12.8 EFFECTIVE FULL POWER YEARS OF OPERATION

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 and system leakage 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.

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 and system leakage testing.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e (Continued)

12. Verifying that the automatic load timer relays are OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators EDG*1 and EDG*3.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) For Divisions I and II, turning gear engaged and emergency stop.

    b) For Division III, engine in the maintenance mode and diesel generator lockout.

f. At least once per 18 months verify each diesel generator starts and accelerates to at least 600 RPM within 10 seconds for EDG*1 and EDG*3, and 870 RPM within 10 seconds for EDG*2. The generator voltage and frequency for EDG*1 and EDG*3 shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 seconds after the start signal. The generator voltage and frequency for EDG*2 shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the start signal. This test shall be performed within 5 minutes of shutting down the diesel generator after the diesel generator has operated for at least 2 hours at 4400 kW or more for EDG*1 and EDG*3 and 2600 kW or more for EDG*2. For any start of a diesel, the diesel must be loaded in accordance with manufacturer's recommendations. Momentary transients due to changing bus loads shall not invalidate this test.

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that all diesel generators EDG*1 and EDG*3 accelerate to at least 600 rpm and EDG*2 accelerates to at least 870 rpm in less than or equal to 10 seconds.

h. At least once per 10 years by:

   1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

   2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

4.8.1.1.3 All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2, within 30 days. Reports of diesel generator failures shall include the information recommended in Position C.3.b of RG 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Position C.3.b of RG 1.108, Revision 1, August 1977.

3/4.10.7 SYSTEM LEAKAGE AND HYDROSTATIC TESTING

LIMITING CONDITION FOR OPERATION

3.10.7 When conducting system leakage or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased above 200°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of a system leakage or hydrostatic test provided the maximum reactor coolant temperature does not exceed 212°F and the following OPERATIONAL CONDITION 3 LCO's are met:

a. 3.3.2, "Isolation Actuation Instrumentation", Functions 1.a.2, 1.b, and 3.a and b of Table 3.3.2-1;

b. 3.6.5.1, "Secondary Containment Integrity";

c. 3.6.5.2, "Secondary Containment Automatic Isolation Dampers"; and

d. 3.6.5.3, "Standby Gas Treatment System."

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature >200°F.

ACTION:

With the requirements of the above specification not satisfied, immediately enter the applicable condition of the affected specification or immediately suspend activities that could increase the average reactor coolant temperature or pressure and reduce the average reactor coolant temperature to ≤200°F within 24 hours.

SURVEILLANCE REQUIREMENTS

4.10.7 Verify applicable OPERATIONAL CONDITION 3 surveillances for specifications listed in 3.10.7 are met.

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 hydrostatic testing and system leakage 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

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low-power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the Technical Specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirements ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this Limiting Condition for Operation.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no-flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access, the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize the discharge of contaminated water to the radioactive waste disposal system.

3/4.10.7 SYSTEM LEAKAGE AND HYDROSTATIC TESTING

This special test exception allows reactor vessel system leakage and hydrostatic testing to be performed in OPERATIONAL CONDITION 4 with the maximum reactor coolant temperature not exceeding 212°F. The additionally imposed OPERATIONAL CONDITION 3 requirement for secondary containment operability provides conservatism in the response of the unit to an operational event. This allows flexibility since temperatures approach 190°F during the testing and can drift higher because of decay and mechanical heat. Additionally, because reactor vessel fluence increases over time, this testing will require coolant temperatures >200°F.

TABLE 5.7.1-1

REACTOR CYCLIC OR TRANSIENT LIMITS AND DESIGN CYCLE OR TRANSIENT

CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
120 heatup and cooldown cycles	70°F to 565°F to 70°F
80 step change cycles	Loss of feedwater heaters
198 reactor trip cycles	100% to 0% of RATED THERMAL POWER
130 hydrostatic and system leakage tests	Pressurized to ≥930 psig and ≤1250 psig

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-69

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated February 5, 1998, Niagara Mohawk Power Corporation (the licensee), proposed a license amendment to change the Technical Specifications (TS) for Nine Mile Point Nuclear Station, Unit 2 (NMP2). The proposed changes would update the terminology and references to 10 CFR 50.55a(f) and (g) consistent with the 1989 edition of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

Specifically, TS 4.0.5 would be changed to reference 10 CFR 50.55a(f) for the inservice testing (IST) Program and 10 CFR 50.55a(g) for the inservice inspection (ISI) Program. Changes to TS Table 4.3.7.5-1 and TS 4.4.3.2.2 would replace the references to ASME Section XI with references to criteria in the IST Program. Changes to TS 3.4.9.1 and 3.4.9.2 would add the phrase "system leakage" to notes that identify testing conditions when the shutdown cooling mode loop may be removed from service. Changes to TS 4.8.1.1.2.h.2 would correct a typographical error for which a reference to ASME Code Section II should refer to Section XI. Appropriate changes would be made to the TS index. Editorial changes to several other TS (i.e., TS 3/4.4.6.1, TS Figure 3.4.6.1-1, TS 3/4.10.7, TS Bases 3/4.4.6, TS Bases 3/4.10.7, and TS Table 5.7.1-1) would make references to "hydrostatic testing" and "leak testing" conform to the terminology to be used in the second 10-year ISI/IST Programs.

2.0 EVALUATION

The licensee has proposed changes to the TS to provide for consistency between (1) the NMP2 TS, (2) the second 10-year interval of the ISI and IST Program Plans for NMP2, and (3) the requirement of 10 CFR 50.55a that the ISI/IST activities conducted during successive 10-year intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code that was in effect 12 months before the start of the 10-year interval. The second 10-year ISI/IST interval for NMP2 began on April 5, 1998. The effective edition of the ASME Code for this second interval is the 1989 edition. The 1989 edition uses some terminology that differs from the terminology used by the 1983 edition to the ASME Code--the edition upon which the existing TS is based. Therefore, changes are proposed to make the TS terminology consistent with the terminology of the 1989 edition, rather than the 1983 edition of the ASME Code.

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The NMP2 programs for the second 10-year interval are written to address the requirements of 10 CFR 50.55a(f), "Inservice testing requirements," and 10 CFR 50.55a (g), "Inservice inspection requirements." Changes to TS 4.0.5 are proposed to reflect these changes in the applicable regulations. The NRC staff agrees that these regulations are the appropriate bases for the NMP2 programs and, therefore, finds these changes to TS 4.0.5 acceptable.

The proposed changes to TS Table 4.3.7.5-1 and TS 4.4.3.2.2 replace references to ASME Code Section XI with references to criteria in the IST Program. As previously noted, the IST Program for the second 10-year interval is based on Section XI of the 1989 edition of the ASME Code. Therefore, the NRC staff finds these changes to be appropriate and acceptable.

The proposed changes to TS 3.4.9.1 and 3.4.9.2 add the phrase "system leakage" to notes that identify testing conditions when the shutdown cooling mode loop may be removed from service. The proposed changes do not change the test conditions or frequency. The proposed changes are consistent with Section XI Code Case N-416-1, dated February 15, 1994, entitled "Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Items by Welding Class 1, 2, and 3, Section XI, Division 1" and Section XI Code Case N-498-1, dated May 11, 1994, entitled "Alternative rules for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems." These two code cases allow a licensee to perform a system leakage test at nominal operating pressure in lieu of a hydrostatic test. The NRC staff has authorized use of Code Cases N-416-1 and N-498-1 at NMP2 by letters dated October 18, 1994, and January 13, 1995, respecitvely. Therefore, the NRC staff finds these changes to be acceptable.

The licensee proposes a change to TS 4.8.1.1.2.h.2 to correct an erroneous reference to "ASME Code Section II" which should have referred to "ASME Code Section XI." The NRC staff finds that Section XI is the appropriate reference and, therefore, finds this correction to be acceptable.

The licensee proposes several changes to terminology to make the references to "hydrostatic testing" and "leak testing" conform with the terminology used in the second 10-year ISI/IST program. The NRC staff finds these changes to be of an editorial nature which does not change the substance of the TS requirement. Similarly, the proposed changes to the TS index are of an editorial nature. These changes are, therefore, acceptable.

The NRC staff finds that each of the proposed changes are of a clarifying nature which do not alter the substance of the TS requirement and are consistent with regulatory requirements in 10 CFR 50.55a. Rather, the changes substitute terminology and references, or make corrections, that are appropriate and consistent with the ISI/IST Program Plans for the second 10-year interval.

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 and changes surveillance requirements. 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 11920). 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.

Principal Contributor: D. Hood

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