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

March 24, 1992

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 2 (TAC NO. M82635)

The Commission has issued the enclosed Amendment No. 37 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station Unit 2 (NMP-2). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 29, 1992.

The amendment revises Technical Specifications 1.31, "Primary Containment Integrity;" 3/4.6.1, "Primary Containment;" 3/4.6.3, "Primary Containment Isolation Valves;" 3/4.8.4, Electrical Equipment Protective Devices;" and deletes associated Tables 3.6.3-1, 3.8.4.1-1, and 3.8.4.3-1. The removal of the equipment lists contained in the tables allows for administrative control of any future changes to the lists without processing a license amendment. This is in accordance with Generic Letter 91-08.

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

Sincerely,

Donald S. Brinkman

for Richard A. Laura, Acting Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 37 to NPF-69
2. Safety Evaluation

cc w/enclosures:
See next page

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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. 37
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 January 29, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-69 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

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. 37 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: March 24, 1992

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
xi	xi
xii	xii
xiii	xiii
xix	xix
1-6	1-6
3/4 3-15	3/4 3-15
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-21	3/4 6-21
3/4 6-22	3/4 6-22
3/4 6-23	3/4 6-23
3/4 8-24	3/4 8-24
3/4 8-25	3/4 8-25
3/4 8-30	3/4 8-30
3/4 8-31	3/4 8-31
B 3/4 6-5	B 3/4 6-5
B 3/4 6-6	B 3/4 6-6
B 3/4 8-3	B 3/4 8-3
3/4 6-24 through 6-35	Pages deleted
3/4 8-26	Page deleted
3/4 8-27	Page deleted

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

PRIMARY CONTAINMENT INTEGRITY

1.31 (Continued)

1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression pool is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

PROCESS CONTROL PROGRAM

1.32 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formula sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of radioactive wastes, based on demonstrated processing of actual or simulated wet or liquid wastes, will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 61, 10 CFR 71, and Federal and State regulations and other requirements governing the transport and disposal of radioactive waste.

PURGE - PURGING

1.33 PURGE and PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATIONS

* During CORE ALTERATIONS and operations with a potential for draining the reactor vessel. This applies to functions described in notes (c) and (d) that isolate secondary containment and automatically start the SGTS.

** When any turbine stop valve is greater than 90% open and/or when the key-locked condenser low vacuum bypass switch is open (in Normal position).

† Valves 2WCS*MOV102 and 2WCS*MOV112 are also required to be OPERABLE or closed in OPERATIONAL CONDITION 5 with any control rod withdrawn but not with control rods removed per Specifications 3.9.10.1 and 3.9.10.2.

†† When handling irradiated fuel in the reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) Refer to Table 3.3.2-4 for valve groups, associated isolation signals and key to isolation signals.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the Trip System in the tripped condition provided at least one other OPERABLE channel in the same Trip System is monitoring that parameter.

(c) Also actuates the standby gas treatment system.

(d) Also actuates reactor building ventilation isolation dampers per Table 3.6.5.2-1.

(e) Also trips and isolates the air removal pumps.

(f) Initiation of SLCS pump 2SLS*P1B closes 2WCS*MOV102 and manual initiation of SLCS pump 2SLS*P1A closes 2WCS*MOV112.

(g) For this signal one Trip System has 2 channels which close valves 2ICS*MOV 128 and 2ICS*MOV 170, while the other Trip System has 2 channels which close 2ICS*MOV 121.

(h) Manual initiation only isolates 2ICS*MOV121 and only following manual or automatic initiation of the RCIC system.

(i) Only used in conjunction with low RCIC steam supply pressure and high drywell pressure to isolate 2ICS*MOV148 and 2ICS*MOV164.

(j) Signal from LPCS/RHR initiation circuitry.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 39.75 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

* See Special Test Exception 3.10.1

** Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once every 92 days.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITIONS FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to:

   1. La, 1.1% by weight of the containment air every 24 hours at Pa, 39.75 psig, or

   2. Lt, 0.72% by weight of the containment air every 24 hours at a reduced pressure of Pt, 20.0 psig.

b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and all Primary Containment Isolation Valves, except for main steam line isolation valves* (and Primary Containment Isolation Valves which are hydrostatically leak tested), subject to Type B and C tests when pressurized to Pa, 39.75 psig.

c. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 Pa, 43.73 psig.

d. Less than or equal to that specified in Table 3.6.1.2-1 through valves in lines that are potential bypass leakage pathways when tested at 40.0 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 La or 0.75 Lt, as applicable, or

* Exemption to Appendix J of 10 CFR 50.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITIONS FOR OPERATION

3.6.1.2 (Continued)

ACTION:

b. The measured combined leakage rate for all penetrations and all Primary Containment Isolation Valves, except for main steam line isolation valves* and valves which are hydrostatically leak tested, subject to Type B and C tests exceeding 0.60 La, or

c. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves, or

d. The measured leakage rate through any valve that is part of a potential bypass leakage pathway exceeding the limit specified in Table 3.6.1.2-1

Restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 La or 0.75 Lt, as applicable, and

b. The combined leakage rate for all penetrations and all Primary Containment Isolation Valves, except for main steamline isolation valves* and valves which are hydrostatically leak tested, subject to Type B and C tests to less than or equal to 0.60 La, and

c. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which pentrate the primary containment to less than or equal to 1 gpm times the total number of such valves, and

d. The leakage rate to less than or equal to that specified in Table 3.6.1.2-1 for any valve that is part of a potential bypass leakage path.

prior to increasing reactor coolant system temperature above 200°F.

* Exemption to Appendix J to 10 CFR 50.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A overall integrated containment leakage rate tests shall be conducted at 40 ± 10-month intervals during shutdown at Pa, 39.75 psig or at Pt, 20.0 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 La or 0.75 Lt, as applicable, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 La or 0.75 Lt, as applicable, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La or 0.75 Lt, as applicable, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 La or 0.25 Lt, as applicable.

   2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at Pa, 39.75 psig, or Pt, 20.0 psig, as applicable.

d. Type B and C tests shall be conducted with gas at Pa, 39.75 psig, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Main steam line isolation valves and the remainder of the valves specified in Table 3.6.1.2-1.

   3. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

   4. Purge supply and exhaust isolation valves with resilient seals.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITIONS FOR OPERATION

3.6.3 Each primary containment isolation valve and reactor instrumentation line excess flow check valve shall be OPERABLE**.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

a. With one or more of the primary containment isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

   4. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with ACTION a.2 or a.3 above, and provided that the associated system is declared inoperable, if applicable, and the appropriate ACTION statements for that system are performed.

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

b. With one or more of the reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either;

   1. The inoperable valve is returned to OPERABLE status, or
   2. The instrument line is isolated and the associated instrument is declared inoperable.

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

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* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

**Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

PRIMARY CONTAINMENT ISOLATION VALVES

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE before returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control, or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from at least one explosive valve , such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and operating life, as applicable.

Pages 3/4 6-23 through 3/4 6-35 not used.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

AC CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITIONS FOR OPERATION

3.8.4.1 The AC circuits inside primary containment that are not provided with primary and backup containment penetration conductor overcurrent protective devices shall be deenergized:*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required AC circuits shall be determined to be deenergized at least once every 24 hours** by verifying that the associated circuit breakers are in the tripped condition.

* Required before power ascension and following final drywell inspection. Excluded from this specification are those penetration assemblies that are capable of withstanding the maximum current available because of an electrical fault inside containment.

**Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

Pages 3/4 8-25 through 3/4 8-27 not used.

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

EMERGENCY LIGHTING SYSTEM - OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.3 The emergency lighting system overcurrent protection devices shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With one or more of the overcurrent protective devices inoperable, within 72 hours remove the inoperable circuit breaker(s) from service by opening the breaker. Return the breaker(s) to OPERABLE status within 7 days, otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The overcurrent protective devices shall be demonstrated OPERABLE at least once per 18 months by selecting and testing one-half of each type of circuit breaker on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the time delay element. The measured response time shall be compared with the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of the nominal instantaneous pickup setting and verifying that circuit breaker trips instantaneously with no intentional time delay.

This page not used.

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT ISOLATION VALVES

3/4.6.3 (Continued)

GDC 54 through 57 of Appendix A to 10 CFR 50. Measurement of the closure time of automatic containment isolation valves is performed for the purpose of demonstrating PRIMARY CONTAINMENT INTEGRITY and system OPERABILITY (Specification 3/4.6.1).

The list of primary containment isolation valves is contained in procedure AP-8.8 and revisions will be processed in accordance with Section 6.0, Administrative Controls.

The maximum isolation times for primary containment automatic isolation valves listed in this specification are either the analytical times used in the accident analysis as described in the FSAR; or times derived by applying margins to the vendor test data obtained in accordance with industry codes and standards. For non-analytical automatic primary containment isolation valves, the maximum isolation time is derived as follows:

1) Valves with full stroke times less than or equal to 10 seconds, maximum isolation time approximately equals the vendor tested closure time multiplied by 2.0.

2) Valves with full stroke time greater than 10 seconds, maximum isolation time approximately equals the vendor tested closure time multiplied by 1.5. Valve closing times do not include isolation instrumentation response times.

Valve closing times do not include isolation instrumentation response times. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

3/4.6.4 SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are four pairs of valves to provide redundancy so that operation may continue for up to 72 hours with no more than one pair of vacuum breakers inoperable in the closed position.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times, the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a subatmospheric condition in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The drawdown time limit has been established considering the same fan performance and building inleakage assumptions as in the post-LOCA analysis except that, since the surveillance test is performed when the plant is shut down, (1) post-LOCA heat-loads are not present; (2) the initial secondary containment pressure is atmospheric; and (3) loss of offsite power is not assumed. Meeting this drawdown time verified that secondary containment leakage and fan performance are consistent with the assumptions of the LOCA analysis.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters operating for 10 hours during each 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and high-efficiency particulate air (HEPA) filters.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. The drywell and suppression chamber hydrogen recombiner system is capable of controlling the expected hydrogen and oxygen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

BASES

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Primary containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance. The list of primary containment AC circuits required to be deenergized is contained in administrative procedure AP-8.8 and revisions will be processed in accordance with Section 6.0, Administrative Controls.

The Surveillance Requirements applicable to lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all-breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The emergency lighting system overcurrent protective devices ensure that a failure of the non-Class 1E portion of the circuit will not affect the operation of the remaining portions of the Class 1E circuits that are necessary for safe shutdown. The list of these overcurrent protective devices is contained in administrative procedure AP-8.8 and revisions will be processed in accordance with Section 6.0, Administrative Controls.

The EPAs provide Class 1E isolation capabilities for the RPS power supplies and the scram power supplies. This is required because the power supplies are not Class 1E power supplies.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. NPF-69

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated January 29, 1992, which superseded the November 6, 1991, submittal, Niagara Mohawk Power Corporation (the licensee) submitted a request for changes to the Nine Mile Point Nuclear Station, Unit 2, Technical Specifications (TS). The proposed amendment would delete the TS tables that include lists of components referenced in individual specifications. In addition, the TS requirements would be modified such that all references to these tables would be removed. Finally, the TS would be modified to state requirements in general terms that include the components listed in the tables. Guidance on the proposed TS changes was provided by Generic Letter 91-08, dated May 6, 1991.

2.0 EVALUATION

The licensee has proposed the removal of Table 3.6.3-1, "Primary Containment Isolation Valves," that is referenced in TS 3/4.6.3. With the removal of this table, the licensee has proposed to include the following statement of the LCO under TS 3.6.3:

> Each primary containment isolation valve and reactor instrumentation line excess flow check valve shall be OPERABLE.

In addition, the licensee proposes to remove all references to Table 3.6.3-1 by revising the following: the definition of Containment Integrity (TS 1.31); TS 4.6.1.1; the action requirements under TS 3.6.3; TS 4.6.3.1 through 4.6.3.4; TS 3.6.12; and TS 4.6.12. The definition of Containment Integrity and TS 4.6.1.1 refer to TS 3.6.3 for an exception that is now covered by a footnote to the LCO rather than by the table removed from the TS. With the removal of the reference to Table 3.6.3-1, the licensee has proposed to state this exception as:

> ..., except as provided in Specification 3.6.3.

The surveillance requirements of TS 4.6.3.1 through 4.6.3.4 would be revised to state: "Each primary containment isolation valve shall...," "Each primary containment automatic isolation valve shall...," "... each primary containment power operated or automatic valve shall...," and "each reactor instrumentation line excess flow check valve shall..." rather than stating the requirements in relation to the valves listed in Table 3.6.3-1.

With the deletion of Table 3.6.3-1, the operability requirements would be stated in general terms that apply to all containment isolation valves including those that are locked or sealed closed. These valves are locked or sealed closed consistent with the regulatory requirements for manually-operated valves that are used as containment isolation valves. Because opening these valves would be contrary to the operability requirements of these valves, the following footnote to the LCO has been proposed.

> Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

The licensee has proposed the removal of Table 3.8.4.1-1, "Primary Containment AC Circuits Deenergized" that is referenced in TS 3.8.4.1. With the removal of this table, the licensee has proposed to include the following statement of the LCO under TS 3.8.4.1:

> The AC circuits inside primary containment that are not provided with primary and backup containment penetration conductor overcurrent protective devices shall be deenergized:

In addition, the licensee has proposed to revise the action requirements under TS 3.8.4.1 to remove the reference to Table 3.8.4.1-1 as follows:

> With any of the above required circuits energized, trip the associated circuit breaker(s) within 1 hour.

The licensee has proposed to add the following statement to the footnote for LCO 3.8.4.1 to further define which circuits shall be deenergized:

> Excluded from this specification are those penetration assemblies that are capable of withstanding the maximum current available because of an electrical fault inside containment.

The licensee has proposed the removal of Table 3.8.4.3-1, "Overcurrent Protective Devices For Non-Class 1E Lighting Fixtures On 1E Emergency System" that is referenced in TS 3.8.4.3. With removal of this table, the licensee has proposed to include the following statement of the LCO under TS 3.8.4.3:

> The emergency lighting system overcurrent protection devices shall be OPERABLE.

In addition, the licensee has proposed to revise the ACTION statement for LCO 3.8.4.3 to remove the reference to the table as follows:

> With one or more of the overcurrent protective devices inoperable,....

The licensee has proposed changes to the above TS that are consistent with the guidance provided in Generic Letter 91-08. In addition, the licensee has provided an updated copy of Bases Section of TS 3/4.6.3 that addresses appropriate considerations for opening locked or sealed closed valves on an intermittent basis. Also, changes were made to Sections 3/4.6.3 and 3/4.8.4 in the Bases section to reference the respective administrative procedures where the removed equipment lists are now located. Finally, the licensee has confirmed that component lists removed from the TS have been updated to identify all components for which the TS requirements apply and are located in controlled plant procedures.

The staff has reviewed the proposed changes and we find them to be acceptable because the removal of the equipment lists and references to such was done in accordance with Generic Letter 91-08.

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 (57 FR 6038). 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 Contributors:
Thomas G. Dunning
Richard A. Laura

Date: March 24, 1992

March 24, 1992

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 2 (TAC NO. M82635)

The Commission has issued the enclosed Amendment No. 37 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station Unit 2 (NMP-2). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 29, 1992.

The amendment revises Technical Specifications 1.31, "Primary Containment Integrity;" 3/4.6.1, "Primary Containment;" 3/4.6.3, "Primary Containment Isolation Valves;" 3/4.8.4, Electrical Equipment Protective Devices;" and deletes associated Tables 3.6.3-1, 3.8.4.1-1, and 3.8.4.3-1. The removal of the equipment lists contained in the tables allows for administrative control of any future changes to the lists without processing a license amendment. This is in accordance with Generic Letter 91-08.

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

Sincerely,

Original Signed By Donald Brinkman for Richard Laura
Richard A. Laura, Acting Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 37 to NPF-69
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PDI-1	PM:PDI-1 DJB	OGC*	D:PDI-1 ROC	
NAME	CVogan CV	RLaura:avl for	JHull	RACapra	
DATE	3/11/92	3/12/92	2/26/92	3/24/92	/ /

*See previous concurrence
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FILENAME: A:\NM282635.AMD

DATED: March 24, 1992

AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. NPF-69-NINE MILE POINT UNIT 2

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