**ATTACHMENT A**

**NIAGARA MOHAWK POWER CORPORATION**

**LICENSE NO. NPF-69**

**DOCKET NO. 50-410**

Proposed Changes to the Technical Specifications

Replace existing pages xi, xii, xiii, 1-6, 3/4 3-15, 3/4 6-1, 3/4 6-2, 3/4 6-3, 3/4 6-4, 3/4 6-21, 3/4 6-22, 3/4 6-23, 3/4 8-24, 3/4 8-25, 3/4 8-30, 3/4 8-31, B3/4 6-5, B3/4 6-6, and B3/4 8-3 with the attached revised pages. These pages have been retyped in their entirety with marginal markings to indicate changes to the text. Pages 3/4 6-24 through 3/4 6-35, 3/4 8-26 and 3/4 8-27 have been removed and notation added indicating that these pages are "Not Used".

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 ROOM

1.32 The PROCESS CONTROL ROOM (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 keylocked 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 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.

## 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

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 penetrate 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.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT LEAKAGE

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.

CONTAINMENT SYSTEMS

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITIONS FOR OPERATIONS

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

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

CONTAINMENT SYSTEMS

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

ELECTRICAL POWER SYSTEMS

## 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.

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* 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 CONDITIONS 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.

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 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.

PRIMARY CONTAINMENT

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 of 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 absorbers 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.

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.

**ATTACHMENT B**

**NIAGARA MOHAWK POWER CORPORATION**

**LICENSE NO. NPF-69**

**DOCKET NO. 50-410**

Supporting Information

Niagara Mohawk proposes to change Nine Mile Point Unit 2 Technical Specifications to remove Tables 3.6.3-1, 3.7.2-1, 3.8.4.1-1, and 3.8.4.3-1 and any references citing those tables in the associated Index, Definitions, Limiting Conditions For Operation, Surveillance Requirements and Bases. These tables constitute component lists which will be relocated to a plant procedure under the change control provisions in the Administrative Controls section of the Technical Specifications. The Technical Specifications are being proposed for revision such that an appropriate description of the scope of the components to which the Technical Specification requirements apply will be incorporated in lieu of the tables or reference to the tables. Reference to the plant procedure containing the component lists is being added to the associated Technical Specification Bases. The removal of these component lists from the Technical Specifications will permit administrative control of any future changes to the lists without processing a license amendment while maintaining an appropriate regulatory process for change control. These Technical Specification improvements are proposed in accordance with the guidance provided in Generic Letter 91-08 "Removal of Component Lists from Technical Specifications".

Additionally, typographical errors have been corrected. The "Plant Service Water System - Operating" page designation has been corrected from 3/5 7-1 to 3/4 7-1 on page xii. Page B3/4 6-6 has the spelling for building corrected.

No Significant Hazards Analysis

10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis, using the standards in Section 50.92, about the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92, the following analysis has been performed:

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Removal of the component lists and correction of the typographical errors does not alter existing Technical Specification Operability or Surveillance Requirements or those components to which they apply. Therefore, the proposed changes do not increase the probability or consequences of any accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

An appropriate description of the removed components has been incorporated in the associated Technical Specification requirements in lieu of the component lists or references thereto. The lists of components to which the Technical Specification requirements apply will be incorporated in plant procedures under the change control provisions in the Administrative Controls Section of the Technical Specifications. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The Technical Specification Limiting Conditions For Operation or Surveillance Requirements for the removed components are not being altered. The component lists will be incorporated into plant procedures which are controlled by administrative procedures which require that all changes be evaluated in accordance with 10 CFR 50.59. The plant procedures will be under the change control provisions in the Administrative Controls Section of the Technical Specifications. Therefore, the proposed changes do not adversely affect a limiting Safety System Setting or involve a reduction in a margin of safety.

# NRC CORRESPONDENCE APPROVAL FORM

PLANT #: NMP 2 APPLIES TO OTHER UNIT? ___ YES X NO

SUBJECT: Tech Spec Amendment DUE: ASAP

PREPARED BY: RAY WITTMAN APPLICABLE NCTS NO.: ______

REFERENCE: GENERIC LETTER 9108

**ENGINEERING AND LICENSING REVIEW** Signature

- ___ Design Engineer
- ___ NRC Project Manager
- ✓ Supervisor Licensing Support
- ___ Manager Technology Services
- ✓ Manager Nuclear Engineering
- ___ Other (Specify) ______

**SITE REVIEW**

- ___ Plant Manager
- ___ Manager Maintenance
- ___ Manager Technical Support
- ___ Manager Operations
- ✓ SORC Review 92-11 01/28/92
- ✓ Tech Review — WAIVED PER CONWAY
- ✓ SRAB Review — PER Phone Con/Poll 1/28/92
- ___ Other (Specify) ______

**FINAL REVIEW** Signature

- ✓ System Attorney — in lieu of Wilson MARK WITERHUND per fone con 1/9/92
- ✓ Manager Licensing
- ✓ Proofreader — 1/28/92

Comments This is resubmittal that removes revetment ditch survey points from original submittal and give provisions of GL91-08 to CIV

**DISPOSITION**

- ___ NCTS Forms Attached GENERATED BUT AWAITING NRC Approval
- ___ N/A No new commitments made or followup actions required.
- ___ FSAR Change Required. LDCN # ______
- ___ Mod Work Request Generated.

**KEYWORDS** (For Records Mgmt.) TECH SPEC TABLE REMOVAL GL91-08

Ray Wittman
Responsible Licensing Engineer/Date

FORM # NEL-502-1-020891

Package Totals 25 pages for Submittal