NUREG-1047
Supplement No. 6

Safety Evaluation Report related to the operation of Nine Mile Point Nuclear Station, Unit No. 2

Docket No. 50-410

Niagara Mohawk Power Corporation
Rochester Gas and Electric Corporation
Central Hudson Gas and Electric Corporation
New York State Electric and Gas Corporation
Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

July 1987

UNITED STATES NUCLEAR REGULATORY COMMISSION

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ABSTRACT

This report supplements the Safety Evaluation Report (NUREG-1047, February 1985) for the application filed by Niagara Mohawk Power Corporation, as applicant and co-owner, for a license to operate Nine Mile Point Nuclear Station, Unit 2 (Docket No. 50-410). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near Oswego, New York.

Supplement 1 to the Safety Evaluation Report was published in June 1985 and contained the report from the Advisory Committee on Reactor Safeguards as well as the resolution of a number of outstanding issues from the Safety Evaluation Report. Supplement 2 was published in November 1985 and reported the resolution of a number of outstanding and confirmatory issues. Supplement 3 was published in July 1986 and reported the resolution of a number of outstanding and confirmatory items, one new confirmatory item, the evaluation of the Engineering Assurance Program, and the evaluation of a number of exemption requests. Supplement 4 was published in September 1986 and reported the resolution of a number of outstanding and confirmatory issues and the evaluation of a number of exemption requests. Supplement 5 was published in October 1986 and reported the resolution of a number of issues and the evaluation of a number of exemption requests. Supplement 5 also supported the issuance of an operating license limited to 5% of rated power for Nine Mile Point Nuclear Station, Unit 2 on October 31, 1986.

This sixth supplement reports the resolution of a number of issues that have been reviewed since Supplement 5 was issued. This report also supports the issuance of a license allowing operation beyond 5% of rated power.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In February 1985, the Nuclear Regulatory Commission staff (NRC or staff) issued a Safety Evaluation Report (SER), NUREG-1047, on the application of the Niagara Mohawk Power Corporation (hereinafter referred to as the licensee) for a license to operate Nine Mile Point Nuclear Station, Unit No. 2 (NMP-2). Supplement 1 to the SER was issued in June 1985 and contained the report of the Advisory Committee on Reactor Safeguards, as well as the staff evaluation of a number of outstanding issues. Supplement 2 was issued in November 1985 and reported the resolution of a number of outstanding and confirmatory issues. Supplement 3 was issued in July 1986 and contained the resolution of a number of open and confirmatory issues, one new confirmatory issue, the evaluation of the Engineering Assurance Program, and the evaluation of a number of exemption requests. Supplement 4, issued in September 1986, resolved a number of outstanding and confirmatory issues and evaluated a number of exemption requests. Supplement 5 was issued in October 1986 and provided the staff evaluation of a number of outstanding and confirmatory issues as well as the evaluation of a number of exemption requests. Supplement 5 supported the issuance of the license limited to 5% of power operation for NMP-2. This sixth supplement contains the resolution of a number of issues that have been resolved since Supplement 5 was issued, and also supports the issuance of a license allowing operation beyond 5% of rated power.

Each of the sections and appendices of this supplement is designated the same as the related portion of the SER. Appendix A, a continuation of the chronology of this safety review, lists in chronological order the correspondence and meetings between the licensee and staff. Appendix B lists reference materials cited in this document. Appendix D lists abbreviations used in this supplement, and Appendix E lists the principal staff contributors. Appendix Q contains errata to the SER and its supplements. Appendices C, and F through P have not been changed by this supplement.

The contents of this document are supplementary to the initial SER, and not in lieu of the SER unless otherwise noted. The NRC Project Manager for the NMP-2 operating license is Joseph D. Neighbors. He may be reached by telephone at (301) 492-8140 or by mail at the following address:

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Copies of this SER supplement are available for inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Local Public Document Room at the Penfield Library, State University College, Oswego, N.Y. 13126.

1.8 Outstanding Issues

The SER identified certain outstanding issues in the staff review that had not been resolved with the licensee at the time the SER was issued. The list of those issues is reproduced in Table 1.3 and the current status of each issue is given. All of these issues are closed.

1.9 Confirmatory Issues

The SER listed certain issues that have essentially been resolved to the staff's satisfaction, but for which certain confirmatory information had not yet been provided by the licensee. In these instances, the licensee had committed to provide the confirmatory information in the near future. If staff review of the information provided for an issue does not confirm preliminary conclusions, that issue will be treated as outstanding and the NRC staff will report on its resolution in another supplement to the SER. Table 1.4 contains a list of confirmatory issues and their current status. All of these issues are closed.

Table 1.3 Outstanding issues

Issue	SER section	Status
(1) Snow loads	2.3.2	Closed, SSER 2
(2) Break analysis of reactor water cleanup line	3.6.2	Closed, SSER 1
(3) Preservice and inservice inspection plan	3.9.6, 5.2.4, 6.6	Closed, SSER 5
(4) Equipment qualification	3.10, 3.11	Closed, SSER 4
(5) Steam bypass of the suppression pool	6.2.1.8	Closed, SSER 1
(6) Secondary containment bypass leakage	6.2.3.1, 15.6	Closed, SSER 5
(7) Containment isolation	6.2.4	Closed, SSER 3
(8) Containment leak testing	6.2.6	Closed, SSER 3
(9) Containment fracture toughness (GDC 51)	6.2.7	Closed, SSER 2
(10) Postaccident monitoring instrumentation	7.5.2.2	Closed, SSER 4
(11) Separation criteria	8.4.5	Closed, SSER 4
(12) Safe and alternate shutdown	9.5.1.4	Closed, SSER 2
(13) Essential lighting	9.5.3	Closed, SSER 2
(14) Air start system	9.5.4, 9.5.6	Closed, SSER 2
(15) Operations management	13.1, 13.4, 13.5	Closed, SSER 1
(16) Procedures generation package	13.5.2	Closed, SSER 5
(17) Preoperational and startup test abstracts	14	Closed, SSER 5
(18) DCRDR and SPDS		
(a) DCRDR	18.1	Closed, SSER 5
(b) SPDS	18.2	Closed, SSER 3

Table 1.4 Confirmatory issues

Issue	SER section	Status
(1) Design of parapet scuppers on roofs of safety-related buildings	2.4.2.2	Closed, SSER 3
(2) Construction quality control tests on revetment ditch	2.4.10	Closed, SSER 4
(3) Feedwater check valves	3.6.2	Closed, SSER 3
(4) Pipe break criteria	3.6.2	Closed, SSER 3
(5) Vertical floor flexibility	3.7.2, 3.7.3	Closed, SSER 2
(6) SRV/pool dynamic loads on containment interior structure	3.8.3	Closed, SSER 4
(7) Analytical results for the reactor internals for LOCA and SSE	3.9.2.4	Closed, SSER 4
(8) Results of Mark II hydrodynamic loads for NSSS piping, components, and equipment	3.9.3.1	Closed, SSER 4
(9) Leak rate test program	3.9.6	Closed, SSER 3
(10) Confirmation of number of ADS SRVs needed to achieve a rapid depressurization during a small-break LOCA based on a plant-specific ECCS analysis.	5.2.2	Closed, SSER 2
(11) Lead factors	5.3.1.2	Closed, SSER 3
(12) Verificaton of CONTEMPT LT/028 computer code	6.2.1.3	Closed, SSER 3
(13) Pool dynamics	6.2.1.7.3	Closed, SSER 3
(a) Pool swell loads		
(b) Loads on submerged boundaries		
(c) Multi-event, lateral load		
(d) Condensation oscillation loads inside the pedestal		
(e) Steam condensation submerged drag loads		

Table 1.4 (Continued)

Issue	SER section	Status
(13) Pool dynamics (continued)		
(f) Bulk-to-local temperature differences		
(g) Single-failure analysis		
(h) Quencher air clearing load		
(i) SRV submerged structure load		
(j) SRV inplant test		
(k) Wetwell-drywell vacuum breakers		
(l) Mark III containment concerns		
(14) Reverse flow testing	6.2.6	Closed, SSER 3
(15) Plant-specific LOCA analysis	6.3, 15.9.3	Closed, SSER 2
(16) Maximum hydrogen generation from the chemical reaction of the cladding with water or steam	6.3.5	Closed, SSER 2
(17) Instrument setpoints	7.2.2.3	Closed, SSER 3
(18) Anticipated transients without scram - mitigation system	7.2.2.4	Closed, SSER 3
(19) Minimum number of channels required to initiate protection actions	7.2.2.6	Closed, SSER 3
(20) Isolation of circuits	7.2.2.8	Closed, SSER 4
(21) Separation of Class 1E equipment and circuits	7.2.2.10	Closed, SSER 6
(22) Testing of protection systems instrumentation	7.3.2.5	Closed, SSER 3
(23) Manual initiation of RCIC	7.4.2.2	Closed, SSER 2

Table 1.4 (Continued)

Issue	SER section	Status
(24) Capability for safe shutdown following loss of electrical power to instrumentation and controls	7.4.2.4	Closed, SSER 3
(25) LPCI and LPCS injection valves interlocks	7.6.2.1	Closed, SSER 3
(26) Multiple control system failures	7.7.2.1	Closed, SSER 3
(27) High-energy-line breaks and consequential control systems failures	7.7.2.2	Closed, SSER 3
(28) Adequacy of station electric distribution system voltages	8.4.1	Closed, SSER 5
(29) Supporting analysis required to confirm adequacy of LFMG motor circuit breaker as backup overcurrent protection for recirculation pump motor electrical penetration	8.4.2	Closed, SSER 4
(30) Site visit confirmation that the 15-ft color-marking interval for cables is sufficient to verify their correct separation	8.4.5	Closed, SSER 4
(31) Verification of the implementation of the electrical separation design criteria during site visit	8.4.5	Closed, SSER 4
(32) Review of analysis or design changes related to qualification of electrical equipment for flooding	8.4.7	Closed, SSER 4
(33) Portable radio communications demonstration	9.5.2	Closed, SSER 3
(34) Emergency lighting	9.5.3	Closed, SSER 2
(35) Procedures for filling fuel oil storage tanks	9.5.4.1	Closed, SSER 4

Table 1.4 (Continued)

Issue	SER section	Status
(36) Details of 1-in. vent line	9.5.4.1	Closed, SSER 2
(37) Division III diesel generator operation - severe conditions	9.5.4.1	Closed, SSER 2
(38) Fuel oil storage and transfer system - P&ID	9.5.4.2	Closed, SSER 2
(39) Procedures for maintaining diesel generator jacket water temperature	9.5.5	Closed, SSER 2
(40) Diesel generator interface on P&ID	9.5.5, 9.5.6	Closed, SSER 3
(41) Procedures for minimum loading of diesel generators	9.5.5	Closed, SSER 4
(42) Divisions I, II, and III diesel generator air-start systems	9.5.6	Closed, SSER 2
(43) Division III air dryer - installation and performance monitoring	9.5.6	Closed, SSER 2
(44) Fire damper control of combustion products	9.5.8	Closed, SSER 2
(45) Concrete dust control	9.5.8	Closed, SSER 2
(46) Solid radioactive waste process control program and a compliance program to meet the requirements of 10 CFR 61 for land disposal of radioactive waste	11.4.2	Closed, SSER 3
(47) Alert and notification of the public within 15 minutes	13.3.2.5	Closed, SSER 3
(48) EOF staffing	13.3.2.8	Closed, SSER 3
(49) Basis for recommendations for protective measures	13.3.2.10	Closed, SSER 3
(50) Compliance with ATWS rule (10 CFR 50.62)	15.8	Closed, SSER 2

Table 1.4 (Continued)

Issue	SER section	Status
(51) IE Bulletin 79-08 Item 6 (NUREG-0737 Item II.K.1.5, Review ESF Valves) and Item 8 (NUREG-0737 Item II.K.1.10, Operability Status)	15.9.2	Closed, SSER 5
(52) Installation of equipment for the automatic restart of RCIC on low water level	15.9.3	Closed, SSER 4
(53) NUREG-0737 Item II.K.3.18, Modification of ADS Logic	15.9.3	Closed, SSER 2
(54) NUREG-0737 Item II.K.3.15, Installation of Modification to RCIC Pipe Break Detection Circuitry	15.9.3	Closed, SSER 4
(55) NUREG-0737 Item III.D.1.1, Integrity of Systems Outside Containment Likely To Contain Radioactive Material	15.9.4	Closed, SSER 3
(56) Site drainage	2.4.2	Closed, SSER 4

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.3 Turbine Missiles

3.5.1.3.10 Summary

License Condition 2.C.(4) of Facility Operating License No. NPF-54 states: "In addition, Niagara Mohawk Power Corporation shall conduct turbine-steam-valve maintenance (following initiation of power output) in accordance with NRC recommendations."

This license condition will not appear in the full-power license since it presently exists in Technical Specification 4.3.8.2 (Surveillance Requirements for Turbine Overspeed Protection System).

5 REACTOR COOLANT SYSTEMS

5.4 Component and Subsystem Design

5.4.8 Reactor Water Cleanup System

5.4.8.1 Introduction

In Section 5.4.8.1 of SER Supplement No. 5, page 5-4, the staff stated:

> The suction line of the RCPB portion of the reactor water cleanup system contains two motor-operated isolation valves which automatically close in response to signals from the reactor pressure vessel low water level and the leak detection system. Actuation of the standby liquid control system and nonregenerative heat exchanger high outlet temperature close the outside isolation valve only.

In a letter dated November 26, 1986 the licensee stated that for clarification and consistency with actual plant performance, the paragraph should read:

> The suction line of the RCPB portion of the reactor water cleanup system contains two motor-operated isolation valves which automatically close in response to signals from the reactor pressure vessel low water level and the leak detection system, and actuation of the standby liquid control system. A nonregenerative heat exchanger high outlet temperature signal closes the outside isolation valve only.

The licensee has committed to revise FSAR page 5.4-46 to reflect this change. The staff finds this change acceptable, subject to verification by inspection.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.7 Pool Dynamic Analysis

6.2.1.7.3 Plant-Unique Loads

(9) Pool Temperature Limit

(c) Bulk-to-Local Temperature Differences

In Supplement 3 to the SER (SSER 3), Section 6.2.1.7.3(9)(c), "Bulk-to-Local Temperature Differences"; the staff concluded that the use of a local-to-bulk temperature difference of 10°F was acceptable. In the staff's final evaluation of the LaSalle safety/relief valve (SRV) in-plant test program (letter from E. Adensam, NRC, to D. Farrar, Commonwealth Edison, dated November 12, 1986), the staff concluded that the local-to-bulk temperature difference of 12°F should be used during plant transients involving SRV discharges in Mark II plants that utilize Mark II quenchers equipped with end cap holes similar to the LaSalle Unit 1 design. Since NMP-2 quencher design is similar to that of LaSalle Unit 1, the 12°F local-to-bulk temperature difference should be used in place of the 10°F local-to-bulk temperature difference referenced in SSER 3.

The staff requested that the licensee review the impact of the change from 10°F to 12°F local-to-bulk temperature difference. By letter dated November 21, 1986, the licensee stated that the calculated minimum allowable difference for NMP-2 is 13.4°F (FSAR Table 6A.10-1). SRV discharge tests for LaSalle Unit 1 show that the average measured local-to-bulk temperature difference is 8.1°F. The 12°F value is the corresponding 95% confidence level, non-exceedance temperature differential. Since NMP-2 utilizes similar SRV quenchers as are used in LaSalle, the 12°F local-to-bulk temperature difference should be used for design purposes. The 12°F value is less than the calculated allowable difference of 13.4°F as presented in Table 6A.10-1 of the NMP-2 FSAR. Therefore, the conclusion on acceptability remains the same.

On the basis of the staff's evaluation, the use of a 12°F local-to-bulk pool temperature difference for use in plant SRV discharge transient in NMP-2 does not result in any design values being exceeded. Therefore, the staff finds this design acceptable.

6.2.4 Containment Isolation System

6.2.4.2 NUREG-0737 Item II.E.4.2, Containment Isolation Dependability

In Table 6.4, "Key to Isolation Signals (revised from SSER 3)," of SER Supplement No. 5, page 6-21, signals H, K, and M were listed as follows:

Signal	Parameter sensed
H	Steam supply pressure low
K	Reactor core isolation cooling high pipe routing or equipment area ambient or differential high temperatures, low steam supply pressure. High steam line differential pressure, high turbine exhaust diaphragm pressure
M	High residual heat removal system equipment area differential or ambient temperatures

For clarification and consistency with FSAR Table 6.2-56 and Technical Specifications Table 3.3.2-4, SSER 5, signals H, K, and M of Table 6.4 should read:

H	Low RCIC steam supply pressure
K	Reactor core isolation cooling high pipe routing or equipment area high temperature, high steamline flow, high turbine exhaust diaphragm pressure
M	High residual heat removal system equipment area ambient temperatures

6.2.4.3 Main Steam Isolation Valve Leakage Concerns

The following safety evaluation was issued to the licensee by letter from Robert M. Bernero to C. V. Mangan dated November 20, 1986. The safety evaluation discusses removal of the main steam isolation valve (MSIV) actuators for the ball valve design. Although the licensee indicated on March 11, 1987, that the MSIV ball valves would be replaced by the wye-pattern globe valves, this safety evaluation is included in this supplement for completeness.

In Supplement 5 to the SER, dated October 1986, the staff stated that the four MSIVs, with modified seat spring configurations and recoated tungsten carbide balls, used for secondary containment integrity prior to criticality would have deactivated, unmodified actuators. The licensee, in letters dated November 11 and 17, 1986, proposed to remove the actuators from the MSIVs which were being used to isolate secondary containment, in order to begin modifications to these actuators while secondary containment was still required. Upon completion of the modifications the actuators would be reinstalled on the valve bodies. The MSIVs would remain in the closed position during the entire time that the actuators were being removed, modified, and reinstalled. Also, inadvertent ball movement was not expected while the actuators were not in place because of the high seat spring force on the ball and the large torque required to rotate the ball. The torque needed to rotate the balls was estimated by the licensee to be a minimum of 50,000 in.-lb.

The staff, therefore, had reasonable assurance that the MSIVs ball valves would remain in the closed position, thereby maintaining secondary containment integrity during actuator removal, modifications, and reinstallation. Removal of the actuators in the manner described in the licensee's letters of November 11 and 17, 1986, in order to begin modifications on the MSIV while secondary containment integrity was required, was therefore acceptable.

The ball valves have been replaced by wye-pattern globe valves because experience with the ball valves has shown that they did not function as well as expected. On May 15, 1987, License Amendment No. 2 was issued to License No. NPF-54 which approved the Technical Specifications for the wye-pattern globe valves and provided a safety evaluation supporting the use of wye-pattern globe valves without a leakage control system.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.2 Specific Findings

7.2.2.10 Separation of Class 1E Equipment and Circuits

By letter dated January 28, 1986 (NMP2L-0594), the licensee submitted a failure modes and effects analysis (FMEA) report evaluating the electrical separation of NMP-2 instrumentation and control system circuits within the power generation control complex (PGCC). This FMEA was issued for the purpose of addressing a staff concern identified in Section 7.2.2.10 of the NMP-2 SER related to compliance of the NMP-2 design to the criteria contained in IEEE-279 (1971), "Criteria for Protection Systems for Nuclear Power Generating Stations." As a result of the staff's review of the FMEA, the licensee provided additional information (letter dated June 2, 1986, NMP2L-0730) which provided commitments to upgrade (make design changes) the NMP-2 design to ensure compliance with required regulatory design criteria.

Section 7.2.2.10 of SER Supplement 5 (SSER 5) provides the staff's evaluation related to the PGCC separation issue. The staff concluded in that evaluation that the licensee provided sufficient information to resolve the issue and that the NMP-2 design will comply with regulatory requirements of IEEE-279 (1971) upon completion of required design modifications. It should be noted that the completion of the required PGCC design upgrade changes is associated with Schedular Exemption 2.D.vii in the low-power license, dated October 31, 1986, which was approved by the staff as part of the evaluation provided in SSER 5. By letter dated June 23, 1987 (NMP2L-1055), the licensee confirmed that 95% of the items covered by this schedular exemption have been resolved.

By letter dated May 18, 1987 (NMP2L-1035), the licensee provided another FMEA report describing additional cases in which non-Class 1E components are connected to Class 1E system circuits and power buses. Specifically, it appeared that numerous non-safety-related system circuits were connected to the Class 1E reactor protection system (RPS) power supply buses or RPS Class 1E circuits.

The staff met with the licensee on June 10, 1987 in Bethesda, Md. to specifically discuss the May 18, 1987 FMEA submittal. At the conclusion of the meeting, the staff informed the licensee that the NMP-2 design appeared not to be in full compliance with the requirements of IEEE-279 (1971). The FMEA reflects the utilization of (1) non-Class 1E components within protection circuits and (2) non-Class 1E protective devices for separation related to the connection of non-Class 1E devices (non-safety-related circuits) to the safety-related RPS power buses and Class 1E protection signal circuits. In part, the regulatory criteria of IEEE-279 (1971) require that adequate isolation from control functions (non-safety related) be designed into protection systems and that any equipment used for safety-related functions be classified as part of the protection system.

The licensee provided more information (letter dated June 16, 1987, NMP2L-1053) requesting an additional schedular exemption from full compliance with 10 CFR 50.55a(h) [IEEE-279 (1971)] until design changes could be implemented before startup after the first refueling outage to upgrade the NMP-2 design for full conformance with regulatory requirements.

Subsequently, by letter dated June 23, 1987 (NMP2L-1056), the licensee informed the staff that the General Electric Co. (GE) has confirmed to them (letter from GE to Niagara Mohawk, dated June 19, 1987) that the individual non-Class 1E components identified in the May 18, 1987 FMEA have been qualified as Class 1E as a result of the implementation of the GE Quality Assurance (QA) Program with the exception of the temperature controller and associated temperature probe in the nuclear steam supply shutoff system. The GE QA Program acquires Class 1E-qualified equipment through dedication by tests and analyses of the components to ensure devices installed in safety-related equipment will perform the required safety function. The licensee has committed to resolve the temperature controller issue by installing redundant Class 1E fuses to accomplish isolation from the Class 1E power supply before exceeding 5% of rated power. The staff finds this commitment acceptable. On this basis, the licensee claims that the schedular exemption request and commitment to upgrade certain RPS non-Class 1E components to Class 1E, including various other hardware changes (isolation devices), before the completion of the first refueling outage is no longer required, and, thus, the licensee has withdrawn its May 18 and June 16, 1987 submittals. Also, by letter dated June 23, 1987 (NMP2L-1055), the licensee informed the staff that the findings associated with GE's QA Program are applicable to the scope of the remaining (5%) PGCC upgrade issue as well. Thus, the licensee certifies that all systems, components, and hardware modifications associated with Schedular Exemption 2.D.vii of License NPF-54 have been completed in accordance with the requirements of the regulations for which the exemption was granted. The staff considers this action withdrawing the May 18 and June 16, 1987 submittals to be acceptable and concludes that the licensee's required actions related to Schedular Exemption 2.D.vii of License NPF-54 is complete.

As a result of the May 18, 1987 FMEA review, the staff asked the licensee to review all systems in the GE scope of supply that are required to comply with IEEE-279 (1971). The purpose of this requested effort was to identify all areas in which non-Class 1E components are being utilized within Class 1E circuits and in which adequate isolation is not being provided between non-Class 1E and Class 1E interfaces within the GE scope of supply. In order to perform this review, the licensee assembled a task force and implemented a formatted program to be followed. The staff performed a site audit on June 15 and 16, 1987 to review the methodology utilized by the licensee. The staff concluded that the licensee used a logical approach to accomplish the required tasks.

The staff was informed by the licensee (letter dated June 23, 1987, NMP2L-1057) that as a result of the task force efforts, no additional uses of non-Class 1E components within Class 1E systems were identified, but that an additional Class 1E/non-Class 1E interface situation was discovered associated with the neutron-monitoring system (NMS) communication with non-safety-related plant equipment such as computers, the rod block monitoring system, and the control room annunciator system. The June 23, 1987 letter describes the use of four categories of isolation devices within the NMS. These are

(1) relays (coil-to-contact and contact-to-contact)
(2) fuse/Zener diode combinations
(3) current-limiting resistors
(4) buffer amplifiers (high impedance input/low output impedance).

As a result of the staff's request to evaluate the adequacy of the isolation between the Class 1E/non-Class 1E interfaces, the licensee informed the staff that the isolation devices identified above have not been satisfactorily qualification tested [i.e., maximum credible fault tests have not been performed to confirm full compliance with the requirements of IEEE-279 (1971)]. The staff informed the licensee that the performance of maximum credible fault tests would be required for acceptable types of isolation devices to resolve the issue unless it can be shown that such confirmatory testing already exists.

The licensee was informed of the staff's position that resistive-type devices (e.g., fuse/diode combination, resistor circuits) are not usually acceptable for use as isolation devices in instrumentation and control circuit signal interface applications and, therefore, the licensee will be required to provide justification or make design modifications to ensure that the Class 1E and non-Class 1E interfaces are isolated with acceptable, fully qualified Class 1E isolation devices. It should be noted that the staff will accept relay contact-to-contact isolation (subject to satisfactory maximum credible fault qualification tests) for the NMS since such isolation was specifically discussed in Section 2.2.8.7 of the GE report, NEDO 10139, "Compliance of Protection Systems to Industry Criteria," which has been approved by the staff, as supported by the letter dated December 13, 1971 from the NRC to GE.

The licensee has requested (letter dated June 23, 1987, NMP2L-1057) a schedular exemption from full compliance to 10 CFR 50.55a(h) to permit interim operation of NMP-2 until a permanent solution satisfactory to the staff is achieved. By letter dated June 25, 1987 (NMP2L-1058), the licensee has committed to provide acceptable qualified Class 1E isolation devices before the end of the first refueling outage. Further, the licensee has committed to keep the staff informed of its actions regarding qualification of the neutron-monitoring system isolation devices and will provide documentation on the qualification of the isolation devices to be utilized. The licensee should provide the confirmatory documentation for the NMS Class 1E isolation devices 60 days before the scheduled first refueling outage.

The June 23, 1987 (NMP2L-1057) and June 25, 1987 (NMP2L-1058) letters provide FMEA information to support the planned interim operation. The FMEA demonstrates that the failure of the existing isolation devices would not adversely affect the connected Class 1E circuits and, thus, will not prevent the NMS from performing its required safety function during the short period of interim operation. Further, the FMEA confirms that

(1) The existing NMS processing electronic equipment is located in the mild environment of the control room which is less severe than the maximum rated operating temperature of this equipment.

(2) The equipment is located in control room panels that have been seismically qualified (as part of overall assembly qualification) to the NMP-2 seismic licensing criteria approved by the staff.

(3) The same equipment has been in service at other licensed boiling-water reactors (BWR/4 and 5) for approximately 15 years and the licensee (including GE) is not aware of any incidents attributable to the Class 1E/non-Class 1E interfaces that have caused failures of the NMS.

(4) The NMS processing electronic equipment is associated with low energy circuits.

(5) The NMS panels that are the same as those at NMP-2 have been qualified as Class 1E by General Electric.

On the basis of the above discussion, the staff considers the licensee's commitments to be acceptable and concludes that the NMP-2 design will comply with the regulatory requirements of IEEE-279 (1971) upon satisfactory qualification testing and the installation of acceptable qualified Class 1E isolation devices within the NMS where necessary. The staff finds that this schedular exemption will not present undue risk to the public health and safety and, thus, should be granted. In accordance with 10 CFR 50.12(a)(2), special circumstances exist that would warrant issuance of the requested exemption. The exemption would provide only temporary relief from the applicable regulation and the licensee has made good-faith efforts to comply with the regulation. In addition, the staff finds the granting of the exemption is authorized by law and is consistent with the common defense and security.

7.3 <u>Engineered Safety Features Systems</u>

7.3.1 System Descriptions

7.3.1.2 Primary Containment and Reactor Vessel Isolation Control System

In Section 7.3.1.2 of SSER 5, in the section discussing the reactor water cleanup (RWCU) system, the staff stated,

> When a predetermined increase in RWCU system area ambient temperatures is detected by any one or more of the three ambient temperature sensors assigned to Division 1, the RWCU system inside isolation valve is signaled to close. Similarly, if any one of the three sensors assigned to Division 2 exceeds its setpoint, the RWCU system outside isolation valve will be signaled to close.

In a letter dated November 26, 1986, the licensee stated that to reflect the actual plant performance the statement should read,

> When a predetermined increase in RWCU system area ambient temperatures is detected by any one or more of the three ambient temperature sensors assigned to Division 1, the RWCU system outside isolation valve is signaled to close. Similarly, if any one of the three sensors assigned to Division 2 exceeds its setpoint, the RWCU system inside isolation valve will be signaled to close.

In its November 26, 1986 letter, the licensee committed to correct the FSAR to reflect the actual plant performance as discussed above. The staff finds this change acceptable.

In Section 7.3.1.2 of SSER 5, in the section discussing the residual heat removal (RHR) system, the staff stated,

> When a predetermined increase in RHR system area ambient temperature is detected by any one or more of the two ambient temperature sensors/switches assigned to Division 1, the RHR outside isolation valve is signaled to close. Similarly, any one of the two sensors/switches assigned to Division 2 exceeding its setpoint will signal the RHR inside isolation valve to close.

In its letter dated November 26, 1986, the licensee stated that to reflect the actual plant performance the statement should read,

> When a predetermined increase in RHR system area ambient temperature is detected by any one or more of the two ambient temperature sensors/switches assigned to Division 1, the associated RHR Division 1 isolation valves are signaled to close. Similarly, any one of the two sensors/switches assigned to Division 2 exceeding its setpoint will signal the associated Division 2 isolation valves to close.

In its letter the licensee committed to correct the FSAR to reflect the actual plant performance as discussed above. The staff finds this change acceptable.

7.4 <u>Systems Required for Safe Shutdown</u>

7.4.1 System Description

7.4.1.1 Reactor Core Isolation Cooling System

In Section 7.4.1.1 of SSER 5 the staff, in accordance with the licensee's letter of July 16, 1986, corrected the second paragraph of Section 7.4.1.1 of the SER to read,

> Separate isolation signals are provided for each isolation valve and include the following: RCIC [reactor core isolation cooling] equipment room high ambient temperature, RCIC emergency area cooler high temperature, RCIC steamline high differential pressure or instrument line break, RCIC turbine exhaust diaphragm high pressure, and RCIC steam supply low pressure.

In its letter dated November 26, 1986, the licensee stated the statement should be further corrected to read,

> Separate isolation signals are provided for each isolation valve and include the following: RCIC [reactor core isolation cooling] equipment room high ambient temperature, pipe routing or RCIC steamline high differential pressure or instrument line break, RCIC turbine exhaust diaphragm high pressure, and RCIC steam supply low pressure.

The licensee stated that this change is consistent with system performance and FSAR Table 6.2-56. The staff finds this change acceptable.

9 AUXILIARY SYSTEMS

9.3 Process Auxiliaries

9.3.2 Process and Post-Accident Sampling System

Paragraph 2C.(9) of Facility Operating License No. NPF-54 states:

> Before exceeding five percent of rated power operation, Niagara Mohawk Power Corporation shall have installed and demonstrated the operability of the Post-Accident Sampling System.

By letter dated December 16, 1986, the licensee provided a certification of completion for the Post-Accident Sampling System which states:

> The preoperational test of the Post-Accident Sampling System has been completed, reviewed and approved. This preoperational test was performed in accordance with the approved Nine Mile Point 2 Preoperational Test Procedure (N2-POT-17-4). The test results were reviewed and approved by the Joint Test Group (JTG) and the Site Operations Review Committee (SORC). The Site Operations Review Committee review and operational acceptance sign-off was completed on November 6, 1986. The Station Superintendent's review and operational acceptance sign-off was completed on November 18, 1986.

The staff finds that this certification satisfies the license condition and is acceptable subject to a post-implementation inspection.

11 RADIOACTIVE WASTE MANAGEMENT

11.4 Solid Waste Management System

In a letter dated February 6, 1987, the licensee provided the preoperational test results for the Waste-Chem solidification system installed at NMP-2. Subsequently, in a letter dated February 13, 1987, the licensee stated that the system is ready for operational use, subject to the completion of the staff's review of the Waste-Chem topical report entitled, "10 CFR 61 Waste Form Conformance Program for Solidified Waste Product Produced by a Waste-Chem Corporation Volume Reduction and Solidification System." In Section 11.4.1 of the SER, the staff stated in part: "On receipt of a compliance program to meet 10 CFR 61 from the applicant, the staff will perform the review, and its evaluation will be provided in a supplement to the SER."

By letter dated April 11, 1986, the licensee submitted the test results of Waste-Chem system to demonstrate the ability of the asphalt binder to maintain the stability of solidified waste products required in accordance with 10 CFR 61.

The staff is currently reviewing the test results which were submitted on a generic basis (Waste-Chem topical report on 10 CFR 61 waste form conformance). The staff's preliminary review indicates that the topical report, for the most part, contains sufficient information for conducting a detailed and final review. The staff anticipates completing its review by September 1987.

11.4.1 System Description

The Waste-Chem system installed at NMP-2 utilizes an extruder/evaporator to reduce waste volume. The reduction is achieved by evaporation while mixing the waste residual with an asphalt (bitumen) binder.

11.4.2 Evaluation Findings

A detailed evaluation of the Waste-Chem solidification system is contained in the NMP-2 SER (NUREG-1047, February 1985).

The staff has reviewed the preoperational test results on the installed Waste-Chem system provided by the licensee with its transmittal letter dated February 6, 1986. The test results indicate that the system design meets the staff's acceptance criteria delineated in Standard Review Plan (SRP) Section 11.4 (NUREG-0800).

Furthermore, in August 1986, the State of South Carolina, Department of Health and Environmental Control, issued a license amendment to the Barnwell Waste Management Facility (Radioactive Material License No. 097). The amendment permits the facility to receive for disposal, oxidized bitumen (asphalt) solidified radwaste.

Therefore, the staff finds that the installed Waste-Chem system, which uses an asphalt (oxidized bitumen) binder to produce a freestanding monolithic product, is acceptable for operation on an interim basis. The interim acceptance will be in effect until the staff completes its review of the Waste-Chem topical report "10 CFR 61 Waste Form Conformance Program for Solidified Process Waste Produced by a Waste-Chem Corporation Volume Reduction and Solidification System," dated May 1986.

In the letter dated February 6, 1987, the licensee also stated that it wishes to continue to use the NUS solidification service in conjunction with the Waste-Chem asphalt system. In SER Supplement 3 (July 1986), the staff approved the use of the NUS solidification system on an interim basis until the staff completes its review of a separate NUS topical report, entitled "Topical Report on 10 CFR 61 Qualified Radioactive Waste Form."

The staff's acceptance of the interim use of the NUS system was based on its approval of NUS Process Service Corporation Topical Report, PS-53-0378, Rev. 0, dated April 1983, entitled "Radwaste Solidification System." The licensee stated in its May 19, 1986 letter that the NUS system will be used in full compliance with NUS Topical Report PS-53-0378, Rev. 0.

Therefore, the staff finds that continued use of the NUS system as a backup to the installed Waste-Chem system is acceptable. This acceptance is also on an interim basis, effective until the staff completes its review of the separate NUS topical report on 10 CFR 61 qualified radioactive waste form.

11.4.3 Conclusions

On the basis of the above evaluation, the staff concludes that the operation of the installed Waste-Chem system and continued use of the NUS system are acceptable on an interim basis. This will be effective until the staff completes its review of (1) Waste-Chem topical report "10 CFR 61 Waste Form Conformance Program for Solidified Process Waste Produced by a Waste-Chem Corporation Volume Reduction and Solidification System," dated May 1986 and (2) NUS "Topical Report on 10 CFR 61 Qualified Radioactive Waste Form." The specific bases for the interim acceptance, as described in the staff's evaluation (above) are (1) acceptable preoperational test results on the Waste-Chem system installed at NMP-2, (2) the staff's previous approval of the NUS Topical Report PS-53-0378, "Radwaste Solidification System," Rev. 0 (approved by the NRC staff with limitations by letter from Cecil O. Thomas to Raymond H. J. Powell, May 30, 1985), (3) the staff's previous approval of the Waste-Chem topical report, "10 CFR 61 Waste Form Conformance Program for Solidified Waste Product Produced by a Waste-Chem Corporation Volume Reduction and Solidification System" (approved by the NRC staff by letter from Karl Kniel to Richard Doyle, April 12, 1978), and (4) acceptance of bitumen solidified radwastes by the licensed burial sites (States of South Carolina and Washington).

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 System Description

SSER 5 provided a revised Table 11.5, "Continuous Monitors (revised from SER)." The following changes to that table were requested by the licensee as clarifications or corrections and are acceptable to the staff.

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  (a) The main stack exhaust monitor type is online.

  (b) The offgas pretreatment trip/high setpoint is identified in the ODCM.

- Page 11-5

  (a) The liquid radwaste effluent trip/high setpoint is identified in the ODCM.

  (b) The turbine plant closed loop cooling water range is $\leq 1.1 \times 10^{-3}$.

  (c) Note ** should read "mr/hr (millirem per hour)."

14 INITIAL TEST PROGRAM

14.1 Preoperational Test

The low-power license issued October 31, 1986 stated that for the schedular exemptions granted in the license, the licensee, in accordance with its letter of October 31, 1986, shall certify that all systems, components, and modifications have been completed to meet the requirements of the regulations for which the exemptions have been granted and shall provide a summary description of actions taken to ensure the regulations have been met. This certification and summary shall be provided 10 days before the expiration of each exemption period as stated in the license.

Below is a list of certification letters supplied in compliance with that license requirement for deferred preoperational tests.

November 26, 1986	Reactor Coolant and Emergency Core Cooling System (ECCS) Leak Detection System
December 5, 1986	Containment Atmospheric Monitoring System
December 5, 1986 and January 31, 1987	Design-Basis Accident (DBA) Hydrogen Recombiner System
January 30, 1987	Offgas System
January 30, 1987	Turbine Electrical Hydraulic Control System

Exemptions for these systems were contained in Section 2.D.v of the low-power license. On the basis of the licensee's certification letters as listed above, these exemptions will not be included in the license for operation above 5% of rated power.

15 ACCIDENT ANALYSES

15.8 Anticipated Transients Without Scram

15.8.1 Required Actions Based on Generic Implication of Salem ATWS Events

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant failed to open upon an automatic reactor trip signal from the reactor protection system. This incident occurred during the plant startup, and the reactor was tripped manually by the operator about 30 seconds after the initiation of the automatic trip signal. The failure of the circuit breakers has been determined to be related to the sticking of the undervoltage trip attachment. Before this incident, on February 22, 1983, at Unit 1 of the Salem Nuclear Power Plant, an automatic trip signal was generated due to a steam generator low-low level during plant startup. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip.

Following these incidents, on February 28, 1983, the NRC Executive Director for Operations (EDO), directed the staff to investigate and report on the generic implications of these occurrences at Unit 1 of the Salem Nuclear Power Plant. The results of the staff's inquiry into the generic implications of the Salem incidents are reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." As a result of this investigation, the NRC requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns. These concerns are categorized into four areas: (1) Post-Trip Review, (2) Equipment Classification and Vendor Interface, (3) Post-Maintenance Testing, and (4) Reactor Trip System (RTS) Reliability Improvements. Within each of these areas, various specific actions were delineated.

This supplement addresses the following action items of Generic Letter 83-28:

- 3.1.1 and 3.1.2 under "Post Maintenance Testing" (Reactor Trip System Components)

- 3.2.1 and 3.2.2 under "Post Maintenance Testing" (All Other Safety-Related Components)

- 4.5.1 under "Reactor Trip System Reliability" (System Functional Testing)

By letters dated April 10, 1984, December 20, 1985, April 15, 1986, and March 18, 1987, the licensee described its planned and completed actions regarding the above items for Nine Mile Point Unit 2.

15.8.1.5 Items 3.1.1 and 3.1.2, Post-Maintenance Testing (Reactor Trip System Components)

Items 3.2.1 and 3.2.2, Post-Maintenance Testing (All Other Safety-Related Components)

Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to ensure that post-maintenance operability testing of safety-related components in the reactor trip system (RTS) is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Licensees and applicants shall submit the results of their check of vendor and engineering recommendations (regarding safety-related components in the RTS) to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to ensure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Licensees and applicants shall submit the results of their check of vendor and engineering recommendations (all other safety-related components) to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

In a letter dated April 15, 1986, the licensee stated that the Administrative Procedure AP-2, "Production and Control of Procedure," requires review of test and maintenance procedures and Technical Specifications to ensure that the post-maintenance operability testing of safety-related components including reactor trip system components is conducted. The review of test and maintenance procedures is effected by interdisciplinary review and cross disciplinary review. The review ensures that the test procedures demonstrate that the equipment is capable of performing its intended safety functions before its return to service. The licensee also stated that all tests in maintenance procedures and Technical Specification changes will be reviewed before implementation.

The licensee's departmental procedures S-IDP-PO, "Outline for I&C Procedures" and S-MI-GEN-002, "Maintenance Instructions for Writing Procedures," control the development of maintenance procedures. These two procedures require post-maintenance testing and are used by the reviewers to ensure that appropriate post-maintenance testing has been incorporated.

The licensee's procedure AP-3.4.2 provides for the administrative control and evaluation of vendor information and recommendations. Accordingly, all NMP-2 related information recommendations from the reactor trip system supplier are reviewed and evaluated by the Independent Safety Engineering Group (ISEG). In addition, NRC I&E Notices and Bulletins and INPO's Significant Event Reports and Significant Operating Experience Reports collectively provide a comprehensive and timely mechanism to ensure that information pertaining to problems with safety-related equipment are identified and corrected. Also, through active participation in General Electric Operations Engineers Program the licensee has enhanced plant performance awareness, and has analyzed, evaluated, and implemented General Electric recommendations as applicable to NMP-2. The program was designed to provide assistance in general plant operations and maintenance; provide assistance in interpretation of service information letters, backfits and other modifications; and increase flow and assimilation of information.

In a letter dated March 18, 1987, the licensee stated that all procedures required at present for electrical maintenance, mechanical maintenance, and instrumentation and control maintenance have been issued and are in effect. These procedures were reviewed and approved in accordance with the licensee's Administrative Procedure AP-2. Currently, the licensee is in the process of replacing all of Stone & Webster's project procedures by site service procedures and Niagara Mohawk departmental procedures, as applicable, and will complete this task by the time of commercial operation.

In the foregoing letter, the licensee also stated that the latest revision of Administrative Procedure AP-2 as well as all site administrative procedures are now applicable to boths Unit 1 and 2. This fulfilled the licensee's commitment in the Unit 2 FSAR (Section 13.5.1.2), which called for incorporation of Unit 2 into the existing site administrative procedures.

The licensee's present classification of safety-related components includes their subcomponents as well. NMP-2 utilizes the quality group classification system as delineated in FSAR Section 3.2. The quality group classification applies to all NMP-2 structures, systems, and components which are required to remain functional during and following a design-basis event to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and to prevent or mitigate the consequences of accident that could result in potential offsite radiation exposure.

The licensee's Engineering Assurance Procedure 3.1 describes review, control, and update of the procurement specifications of safety-related items. The procurement specifications include requirements for qualification testing, review, receipt, and approval of testing documentation and vendor manuals. Maintenance and surveillance data extracted from the vendor documentation are transmitted to NMPC Project Engineering via the Equipment Qualification Maintenance Program Data Sheet (EQMPDS). EQMPDS information is transferred to onsite maintenance management for incorporation into maintenance procedures in accordance with maintenance instruction MI-4.0.

The licensee had actively participated in the Nuclear Utility Task Action Committee (NUTAC) formed to control and utilize information regarding safety-related components. The exchange of information provides a mechanism for interchanges among utilities/vendors and utilities/regulator and established the Significant Event Evaluation and Information Network (SEE-IN) and Nuclear Plant Reliability Data System (NPRDS) programs managed by INPO. The licensee's Procedure TDP-6, "Nuclear Plant Reliability Data System Failure Reporting," delineates NUTAC Vendor Equipment Technical Information Program (VETIP) to contribute information to the SEE-IN program via the NPRDS.

The licensee has stated that all corrective maintenance on safety-related equipment at NMP-2 is performed in accordance with Administrative Procedure AP-5.2, "Procedure for Repair," which specifies the requirements for post-maintenance testing (PMT) following any corrective maintenance. TDP-8, "Post-Maintenance Testing Criteria," provides guidance for the type of testing required based on the type of components and associated maintenance. Appendix C of AP-5.2 provides the pre- and post-maintenance testing criteria which establish the extent of testing following a maintenance activity.

The maintenance or repair work is initiated through the station Work Requests (WRs) in accordance with AP-5. The maintenance supervisor determines the availability and the adequacy of maintenance and test procedures to meet requirements of TDP-8. Upon completion of the work, the WR is returned to the control room for a review by the Station Shift Supervisor. Successful completion of the WR and required post-maintenance testing results in acceptance of the system or components for return to service by the operations department.

The licensee also stated that all correspondence from Niagara Mohawk Project Engineering to the NMP-2 Station Superintendent were reviewed and cognizant personnel were interviewed to establish if any additional testing recommendations were still outstanding. No additional testing recommendations were identified.

Based on the above, the staff concludes that the licensee's actions are consistent with the NRC staff positions for Items 3.1.1, 3.1.2, 3.2.1, and 3.2.2 of Generic Letter 83-28 and are, therefore, acceptable.

15.8.1.6 Item 4.5.1, Reactor Trip System Reliability (System Functional Testing)

On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W, and CE plants; the circuitry used for power interruption with the silicon-controlled rectifiers on B&W plants; and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

The NMP-2 reactor trip design features include a pair of dc solenoid-operated backup scram valves. These valves are normally deenergized. At NMP-2 the scram pilot air system controls and supplies air to operate the scram valves and the scram discharge volume vent and drain valves through the two backup scram and two redundant reactivity control system (RRCS) solenoid-operated air valves. In an unlikely event, if the scram pilot valve fails to function, the action of the backup scram valves ensures that the control rods insert, thus enhancing the reliability of the reactor trip function.

In a letter dated April 15, 1986, the licensee stated that current testing of the scram pilot valve is accomplished through the existing surveillance program. Accordingly, the trip system is functionally tested from the sensing instrument, through the trip logic circuitry, to the scram pilot valves. The surveillance procedures are written to test the one-out-of-two-taken-twice logic in such a manner that the channels are tested independently. This allows one-half of the necessary logic to "makeup," actuating the entire trip channel up to and including one out of the two scram pilot valves on every control rod's scram inlet and discharge valves in each channel.

In the plant Technical Specifications, the licensee indicated that the scram test will be performed each operating cycle to demonstrate operability and reliability of the system. The frequency of testing will be as follows:

(1) for all control rods prior to thermal power exceeding 40% of rated thermal power following core alterations or after a reactor shutdown that is greater than 120 days

(2) for specifically affected individual control rods following maintenance or a modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods

(3) for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation

In the April 15, 1986 letter, the licensee indicated that the reactor trip system at NMP-2 is not designed for on-line testing of the backup scram valves. The current design would result in a full scram of one-half the control rods, if one of the backup scram valves were energized. Thus, functional testing of these valves during plant operation would require a plant scram, a significant challenge to plant safety systems and, therefore, a potential degradation of plant safety. The backup scram valves are non-safety-related additions employed to enhance the reliability of the safety-related reactor trip system. Based on the redundancy of the backup scram valves and the scram pilot valves, the licensee established that modifications to permit on-line testing of the backup scram valves are not warranted. However, in response to Generic Letter 83-28, Action Item 4.5.2, the licensee indicated that the scram pilot valves are tested weekly during average power range monitor half scram test, and in accordance with the NRC guidance, the backup scram valves will be tested during each refueling outage.

Based on the above, the staff concludes that the licensee's actions in this regard are consistent with the NRC staff position for Action Item 4.5.1 of Generic Letter 83-28 and, therefore, acceptable.

15.9 TMI Action Plan Requirements

15.9.2 NUREG-0737 Item II.K.1, IE Bulletins on Measures To Mitigate Small-Break LOCAs and Loss-of-Feedwater Transients

NUREG-0737 Item II.K.1.10, Review and Modify (as Required) Procedures for Removing Safety-Related Systems From Service (and Restoring to Service) To Assure Operability Status Is Known

In SER Supplement 5 (SSER 5) the staff concluded that the licensee met the requirements of Item II.K.1.10 based on NMP-2 Administrative Procedure (AP) AP-3.3.1 "Control of Equipment Markups," Revision 1, July 2, 1985. In its letter dated November 26, 1986 (NMP-2 L-0943), the licensee informed the staff that the referenced administrative procedure has been revised and the staff's

SER should be changed to reflect the changes in the administrative procedure. The licensee provided additional information and clarification in its letter dated January 23, 1987 (NMP-2 L-0978).

NMP-2 Administrative Procedure AP-3.3.1, "Control of Equipment Markups," Revision 2, July 10, 1986, paragraphs 3.1.1 and 3.1.2 requires that testing or verification of operability (in accordance with the Technical Specifications) shall be performed on the remaining redundant system before a safety-related system is removed from service. In addition, a licensed operator, independent of the person performing the test or verification, shall verify that the equipment is correctly returned to the normal operable status.

Technical Specifications do not always require performance of an operability test on the redundant system before removing a safety-related system from service. Availability of the redundant system is verified by the use of an Equipment Status Log which is maintained in the control room. Operability requirements for the redundant systems will be met according to the plant Technical Specifications. This conforms with the requirements of TMI-2 Item II.K.1.10, and the licensee's position is acceptable.

15.9.4 NUREG-0737 Item III.D.1, Primary Coolant Outside Containment

NUREG-0737 Item III.D.1.1, Integrity of Systems Outside Containment Likely To Contain Radioactive Material for Pressurized-Water Reactors and Boiling-Water Reactors

In a letter dated January 12, 1987, the licensee provided a summary of the initial leak test results. The leak tests were performed for systems outside the NMP-2 containment that are likely to contain radioactive materials under reactor accident conditions. This submittal is in response to license condition 2.C.(12) of Nine Mile Point, Unit No. 2 Facility Operating License No. NPF-54.

The license condition states:

> In accordance with the schedule described in Niagara Mohawk Power Corporation's letter dated April 21, 1986, Niagara Mohawk Power Corporation shall submit, within two months after completing fuel loading, the initial leak test results for systems outside containment, with the exception of the Reactor Core Isolation Cooling (RCIC) System, along with descriptions of corrective maintenance performed as a direct result of the Niagara Mohawk Power Corporation's evaluation of the leakage program. The leak test results for the RCIC system shall be provided within five months after first exceeding five percent of rated power.

Item III.D.1.1 of NUREG-0737 requires a nuclear power plant to implement a program for reducing leakage from systems outside the containment. Specifically, Item III.D.1.1 requires that the leakage be reduced to as-low-as-practical levels for those systems outside the containment which would or could contain highly radioactive fluids during a serious transient or an accident. The program should include:

(1) the implementation of all practical leak reduction measures,
(2) the measurement of leakage rates with systems in operation, and
(3) the implementation of a program for preventive maintenance to reduce leakage to as-low-as-practicable levels.

The licensee submitted a leakage reduction program for NMP-2 with its letter dated January 16, 1986. In the letter, the licensee stated that the initial leak rate test results, along with the corrective maintenance to be performed as a direct result of the licensee's evaluation of the leakage program, would be submitted to the NRC for review at a later date.

The staff concluded in SSER 3 (July 1986) that the licensee's program met the requirements in NUREG-0737, except for the initial leak rate test results. Therefore, the staff accepted the NMP-2 Leak Rate Reduction Program, with the requirement that leak rate test results would be submitted in accordance with license condition 2.C.(12). The basis for the staff's acceptance was that the licensee's program included the measurement of leak rate and implementation of reductions to as-low-as-practicable levels as described in items 1, 2, and 3 above.

In a letter dated January 12, 1987, the licensee submitted the initial leakage rate test results for (1) the residual heat removal (RHR) system (train A, B, and C; shutdown cooling; and steam condensate return), (2) the low-pressure core spray (CSL), (3) the high-pressure core spray (CSH), (4) the reactor coolant sampling system, (5) the standby gas treatment and containment purge systems, (6) the reactor core isolation cooling system, and (7) the primary containment monitoring system. In performing leakage rate measurements and detection, the licensee used a helium tracer for the air and gas systems and visual methods for the liquid systems.

As a direct result of the leak rate measurement program, the licensee has replaced the valve packings in the RHR system, as well as the flex hoses in the seal line to the governing valve and the trip throttle valve in the reactor core isolation cooling system. The licensee was able to reduce leakage to zero for most of the systems. For those systems in which the leakage was not zero (namely, the RHR-A, RHR-B, and CSH systems), the leakage rate was reduced to less than five drops per minute.

The staff finds that the licensee has demonstrated a reasonable effort and employed reasonable means for keeping the leakage rate to as-low-as-practicable levels for systems outside the containment. The leakage rates, measured with the systems being in operational conditions, and with corrective maintenance having been performed, are also acceptable.

On the basis of the above evaluation, the staff concludes that the licensee's initial leak rate test results, as summarized in its January 12, 1987 letter, meet the staff position delineated in Item III.D.1.1 of NUREG-0737. Therefore, the licensee's summary of the initial leak rate test results is acceptable and satisfies license condition 2.C.(12).

16 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the staff. The Nine Mile Point Nuclear Station, Unit No. 2 (NMP-2), Technical Specifications are included as Appendix A to the operating license. Included in the Technical Specifications are sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

Technical Specifications for NMP-2 were issued as part of the license limited to 5% of rated power on October 31, 1986. Subsequent to that, the licensee requested changes to the Technical Specifications in letters dated December 9 and 10, 1986; January 15 and 23, February 3, and March 11, 1987 (see Table 16.1). These requested changes are discussed in the following sections.

16.1 Changes to SORC and SRAB Reporting Times (TS Sections 6.5.3.10.a and 6.5.3.10.c)

By letter dated December 10, 1986, the licensee proposed changes to NMP-2 Technical Specifications 6.5.3.10.a and 6.5.3.10.c that would lengthen the time allowable for issuance of minutes and audit reports of the Safety Review and Audit Board to 30 days and 90 days, respectively. The changes were requested (1) in order to make the Technical Specifications for NMP-2 consistent with those for NMP-1 in this area and (2) because the licensee believes that the 14-day and 30-day requirements are too short to permit adequate review of minutes and audit reports before their issuance.

The licensee stated that these particular Technical Specifications should be the same for Unit 1 and Unit 2. However, Unit 1 received an operating license more than 17 years ago. Since then, regulatory requirements have changed for the licensing of more current plants. The requirements applicable to Unit 2 are those presented in the Standard Technical Specifications: 14 days for meeting minutes (Specification 6.5.3.10.a) and 30 days for audit reports (Specification 6.5.3.10.c). Although there may be some benefit to the licensee in making these specifications consistent between the two units, there is no licensing requirement for consistency and the licensee has not justified the large proposed increases in reporting times.

The licensee's claim that the 14-day and 30-day requirements are too short has not been justified, and the staff is not aware that other licensees have found these requirements too restrictive.

The staff concludes, therefore, that the licensee's proposed changes for Unit 2 have not been adequately justified and, therefore, are not acceptable.

16.2 Suppression Pool Temperature Alarm Setpoints

By letters dated December 10, 1986 and January 15 and March 27, 1987, the licensee requested changes to Technical Specification 4.6.2.1.c in the area of suppression-pool, high-temperature, alarm setpoints to make this Technical Specification consistent with the Final Safety Analysis Report, Question and Response Volume 3, Humphrey Concerns, Section 9.3, page HC-16.

NUREG-0783, "Suppression Pool Temperature Limit for BWR Containment" (November 1981), identifies the requirements which are necessary to support a conclusion that a manual scram can be accomplished when the suppression pool temperature reaches 110°F, as indicated in the NMP-2 Technical Specifications. These requirements are:

(1) Install alarms/displays in the control room to give the operator immediate and unambiguous indications of a stuck-open safety relief valve (SRV).

(2) Provide alarms/displays to alert the operator about the suppression pool temperature. Set the alarm at TS 1 (maximum pool temperature for continued power operation (90°F)) and TS 3 (pool temperature limit for reactor scram (110°F)).

(3) Provide clear instructions in the operating procedures to prohibit the operator from prolonging the initiation of a manual scram. For example, the operational procedures should specify the maximum number of attempts the operator will be allowed to use to reclose a stuck-open SRV.

Items 1 and 2 (above) have been provided at Nine Mile Point Unit 2. Item 3 is covered in an Emergency Operating Procedure (EOP). The EOP requires placing the reactor mode switch in the shutdown position if the SRVs cannot be closed within 5 minutes.

To make the Technical Specification consistent with the NUREG-0783 requirements, one alarm should be set at the maximum Technical Specification pool temperature limit, TS 1, continued power operation. This value is 90°F. The second alarm should be set at the temperature limit TS 3, for reactor scram. This value is 110°F. The changes proposed to Technical Specification 4.6.2.1 in the licensee's letter of March 27, 1987 conform to the NUREG-0783 guidelines.

On the basis of the staff's evaluation, the proposed changes in the licensee's letter of March 27, 1987 to page 3/4 6-17 of the Nine Mile Point Unit 2 Technical Specifications are acceptable.

16.3 Main Steamline Drain Valve Type C Leak Testing

By letter dated February 3, 1987 (NMP2L 0994), the licensee requested a change to the Technical Specifications concerning leak rate testing of four main steam line drain valves at Nine Mile Point Unit 2. The present Technical Specifications state on pages 3/4 6-28 and 3/4 6-35 (footnote (n)) that valves 2MSS*SOV97A, B, C, and D are Type C tested in the reverse direction. The proposed change would indicate that these valves are Type C tested and may be tested in the reverse direction.

The proposed Technical Specification change will clarify that Type C testing of the main steam line drain valves is allowable in either the postaccident direction or in the reverse direction. 10 CFR 50 Appendix J, Section III.C.1, requires Type C testing in the same direction as that when the valve would be required to perform its safety function, unless it can be determined that the results from the tests for a pressure applied in a different direction will provide equivalent or more conservative results. At the time the present Technical Specifications were prepared, the staff had reviewed the basis used as justification for reverse direction testing of the main steam line drain valves and concluded that it was acceptable. The staff, therefore, finds that the proposed Technical Specification clarifies that reverse testing of the main steam line drain valves is permissible and may be considered in lieu of a forward test direction.

On the basis of the staff's evaluation, Technical Specification Table 3.6.3-1, page 3/4 6-35, note (n) will be revised to read, "These valves are Type C tested and may be tested in the reverse direction."

16.4 Fire Protection

The NRC in Generic Letter 86-10 stated that licensees may add the Fire Protection Program into the Final Safety Analysis Report (FSAR).

This proposal was made because the fire protection license conditions vary widely from plant to plant and these variations have created problems in identifying the operative and enforceable fire protection requirements at each facility.

In certain situations the fire protection license conditions also create difficulties because they do not specify when a licensee may make changes to the approved Fire Protection Program without requesting a license amendment.

These problems exist because of the many submittals that constitute the fire protection program for each plant. The staff believes that the best way to resolve these problems is to incorporate the Fire Protection Program and major commitments, including the fire hazards analysis, by reference into the FSAR for the facility. In this manner, the Fire Protection Program, including the systems, the administrative and technical controls, the organization, and other plant features associated with fire protection would be on a consistent status with other plant features described in the FSAR. Also, the provisions of 10 CFR 50.59 would then apply directly for changes the licensee desires to make in the Fire Protection Program that would not adversely affect the ability to achieve and maintain safe shutdown. In this context, the determination of the involvement of an unreviewed safety question defined in 10 CFR 50.50(a)(2) would be made based on the "accident...previously evaluated" being the postulated fire in the fire hazards analysis for the fire area affected by the change. The staff also believes that a standard license condition, requiring licensees to comply with the provisions of the Fire Protection Program as described in the FSAR, should be used to ensure uniform enforcement of fire protection requirements.

The NMP-2 low-power license contains the standard license condition discussed above. This license condition will be included in the full-power license, but will be revised to incorporate later licensee submittals.

With the inclusion of the Fire Protection Program into the FSAR and the inclusion of the license condition as noted above in the license, the sections of the Technical Specifications that deal with fire protection are unnecessary in that document and may be deleted from it.

The licensee, by letters dated December 9, 1986; April 10, 1987; and May 20, 1987, requested that the fire protection sections be removed from the Technical Specifications and informed the NRC that the fire protection requirements contained in the Technical Specifications will be incorporated into the FSAR consistent with the guidance contained in Generic Letter 86-10.

The licensee's existing Fire Protection Program consists of FSAR Section 9.5.1, FSAR Appendix 9A, the fire protection sections of the Technical Specifications and the Operational Quality Assurance Manual, and the implementing administrative, maintenance, and surveillance procedures.

The licensee proposed to delete the following Technical Specifications sections containing the fire protection elements and include the applicable requirements into the FSAR, Appendix 9A as follows:

Delete TS section:	Title:	Include in FSAR section:
3/4.7.8	Fire Barriers	9A.3.5.1.1
3.3.7.8	Fire Detection	9A.3.6.1
3/4.7.7	Fire Protection Water Supply System	9A.3.6.2
3.7.7.1	Fire Pumps	9A.3.6.2.6
3.7.7.6	Outside Hose Stations	9A.3.6.2.7
3.7.7.2	Sprinkler and Water Spray System	9A.3.6.3.3
3.7.7.5	Manual Hose Installations	9A.3.6.3.4
3.7.7.4	Halon 1301 Suppression	9A.3.6.4
3.7.7.3	Co_2 Suppression System	9A.3.6.5

Under the guidance of Generic Letter 86-10, these requirements can be deleted from the Technical Specifications provided that they are described or referenced in the FSAR and that equivalent controls are established.

The staff has reviewed the licensee's proposal and determined that when this proposal is implemented the applicable Technical Specifications requirements, Action Statements, and Surveillance Requirements will be incorporated entirely into the FSAR.

The implementing administrative, maintenance, and surveillance procedures of the fire protection program will remain in effect.

By letter dated May 20, 1987, the licensee proposed to add the following to the Technical Specifications:

The Fire Protection Program is a program to implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report as amended and as approved in the Safety Evaluation Report (NUREG-1047) dated February 1985 as supplemented. The noncompliances with the above Fire Protection Program that affect the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with the requirements of 10 CFR 50.73.

With the addition of this section into Technical Specifications, equivalent controls of the Fire Protection Program will be maintained. The licensee's actions described above are consistent with the guidance of Generic Letter 86-10.

On the basis of the above, the staff finds that;

(1) The existing FSAR sections of the Fire Protection Program (FPP) together with the proposed revisions satisfy the guidance in Generic Letter 86-10 for incorporating the FPP into the FSAR.

(2) The deletion of the Technical Specifications sections are in accordance with the guidance in Generic Letter 86-10.

(3) The existing Technical Specifications requirements that deal with fire protection are incorporated entirely into the FSAR, and equivalent administrative controls exist to control these activities.

(4) Adequate administrative controls exist to determine if a proposed FPP change would adversely affect the ability to achieve and maintain a safe shutdown in the event of a fire.

On this basis, it is concluded that the licensee's request to delete the fire protection elements from the Technical Specifications meets the guidance in Generic Letter 86-10, and there is reasonable assurance that the health and safety of the public will not be endangered by operation in this manner and such activities will be conducted in compliance with the Commission's regulations and the approval of these actions by the Commission will not be inimical to the common defense and security or to the health and safety of the public.

16.5 Scram Discharge Volume Vent and Drain Valves

By letter dated December 9, 1986, the licensee proposed a change to Technical Specification 4.1.3.1.4.a. The change would eliminate surveillance of the scram discharge volume (SDV) vent and drain valves under operating conditions of pressure and temperature normal to a scram operation. The intent of the surveillance was derived from the "Generic Safety Evaluation Report for BWR Scram Discharge System," dated December 1, 1980. Surveillance Criterion 3 of the report states the following:

The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control-rod density.

The basis for the acceptance of Surveillance Criterion 3 is that it will check the proper operation of system components under operating conditions normal to a scram operation. The staff finds the licensee's proposed changes unacceptable because this surveillance would be eliminated.

Table 16.1 Technical Specification changes

Date of licensee's request (letter number)	Description of change	Status
December 9, 1986 (NMP2L-0949)	Licensee requested that typographical errors be corrected and minor clarifications be made to the Technical Specifications.	These changes will be made as requested.
December 9, 1986 (NMP2L-0951); April 10, 1987 (NMP2L-1021); and May 20, 1987 (NMP2L-1036)	Licensee requested deletion of the fire protection related sections of the Technical Specifications.	See Section 16.4 of this supplement.
December 9, 1986 (NMP2L-0948)	Licensee requested section 4.1.3.1.1 of the Technical Specifications regarding surveillance requirements for the scram discharge volume drain and vent valves be revised.	Denied. See Section 16.5 of this supplement.
December 9, 1986 (NMP2L-0950)	Licensee requested Technical Specification 4.6.6.1.b.1 regarding calibration of the containment hydrogen recombiner thermocouples be revised.	Licensee withdrew request on January 30, 1987 (NMP2L-0986)
December 10, 1986 (NMP2L-0952)	Licensee requested changes to Section 6.5.3.8 of the Technical Specifications regarding reporting times for the Safety Review and Audit Board.	Denied. See Section 16.1 of this supplement.
December 10, 1986 (NMP2L-0953); January 15, 1987 (NMP2L-0972); and March 27, 1987 (NMP2L-1010)	Licensee requested Section 4.6.2.1 of the Technical Specifications concerning suppression pool temperature alarms be revised.	See Section 16.2 of this supplement.

Table 16.1 (Continued)

Date of licensee's request (letter number)	Description of change	Status
January 23, 1987 (NMP2L-0977)	Licensee requested a clarification to the Technical Specifications concerning Type C testing of main steam isolation ball valves.	On March 11, 1987, the licensee notified the NRC of its decision to replace these valves. This clarification is no longer applicable.
February 3, 1987 (NMP2L-0994)	Licensee requested a clarification to Technical Specification Table 3.6.3-1 regarding Type C testing of main steam isolation drain valves.	See Section 16.3 of this supplement.
March 11, 1987 (NMP2L-1004); March 16, 1987 (NMP2L-1006); March 18, 1987 (NMP2L-1007); March 31, 1987 (NMP2L-1014) April 2, 1987 (NMP2L-1015); April 3, 1987 (NMP2L-1016); April 7, 1987 (NMP2L-1019); April 23, 1987 (NMP2L-1026); and April 28, 1987 (NMP2L-1029)	Licensee requested changes to Technical Specification Tables 2.2.1-1, 3.6.1.2-1, and 3.6.3-1 to reflect changes relating to replacing the main steam isolation ball valves with main steam isolation wye-pattern globe valves.	Issued by License Amendments Nos. 1 and 2 to Facility Operating License No. NPF-54, dated May 11 and 15, 1987, respectively.
April 28, 1987 (NMP2L-1028)	Licensee requested a change to Technical Specification 4.4.3.2.1 related to leakage-detection capabilities of the containment airborne gaseous and particulate radioactive detection and the reactor vessel head flange leak-detection systems.	Licensee withdrew request on May 27, 1987 (NMP2L-1040).

18 HUMAN FACTORS ENGINEERING

18.1 Detailed Control Room Design Review

In SER Supplement 5, Section 18.1, the staff stated that the operating license would be conditioned to require that temporary meter banding is to be completed before exceeding 5% of rated power. In a letter dated February 26, 1987, the licensee stated that, in accordance with its letter dated August 4, 1986, this work has been completed. This license condition therefore will not be continued in the license issued to permit operation above 5% of rated power.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF NRC STAFF RADIOLOGICAL REVIEW OF NINE MILE POINT NUCLEAR STATION, UNIT 2

October 17, 1986	Letter from licensee forwarding revised FSAR page regarding temperature performance of waterstop material per request. Revision will be incorporated in future FSAR update.
October 17, 1986	Letter from licensee requesting amendment to October 15, 1986 request for schedular exemption permitting completion of construction, testing, and post-test review for installation of Class 1E protective devices. Amendment will incorporate request for exemption from 10 CFR 50.55a(h).
October 17, 1986	Letter from licensee advising staff that FSAR will be amended within 6 months of receipt of operating license (OL). Amendment will incorporate information submitted between Amendment 27 and issuance of license and information committed to resolve 10 CFR 50.55(e) report on main steam isolation valves (MSIVs).
October 17, 1896	Letter from licensee forwarding draft initial prototype test program for MSIVs per October 15, 1986 meeting on construction deficiency report. Program will confirm correlation between across-seat and between-seat leak testing methods.
October 17, 1986	Generic Letter 86-17 sent to all boiling-water reactor (BWR) licensees and applicants regarding availability of NUREG-1169, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods."
October 20, 1986	Letter from licensee withdrawing May 7, June 18, and July 3, 1986 requests for schedular exemption regarding deferral of preoperational tests until after fuel load. Because of MSIV schedule impact, automatic depressurization system preoperational test is complete.
October 21, 1986	Letter from licensee forwarding list of comments on SSER 4, reflecting difference between SER and SSERs, FSAR through Amendment 27, and subsequent letter updates. Comments do not address status of open items or affect Technical Specifications.

October 22, 1986	Letter from licensee forwarding response to request for additional information regarding residual heat removal heat exchanger (RHR HX) outlet temperature indication qualification. Control room valve position indication is provided for each bypass valve associated with RHR HX bypass line.
October 22, 1986	Letter from licensee forwarding marked-up FSAR Page 9A.3-23 and Figures 9A.3-4, 9A.3-5, and 9A3-6. Change on Page 9A.3-23 deletes reference to high-pressure core spray (HPCS) room and 600-volt switchgear room. Changes will be incorporated in future FSAR revision.
October 24, 1986	Letter from licensee certifying that with the exception of items on enclosed list of SER/TS/FSAR differences, final draft Technical Specifications reflect as-built configuration of plant, FSAR through Amendment 27, SER through Supplement 4, and all subsequent changes as of October 24, 1986.
October 24, 1986	Letter from licensee forwarding commitments regarding control room ambient temperature limit, per NRC's October 22, 1986 request. Commitments should permit NRC to resolve concerns about reliability of panel-mounted electronic equipment in event of cooling of room.
October 28, 1986	Letter from licensee advising staff that design, construction, and preoperational testing required for fuel loading is complete and unit is ready to load fuel by October 29, 1986. Requests that NRC complete all steps necessary for issuance of operating license to permit fuel loading and low-level operation.
October 28, 1986	Letter to licensee forwarding updated draft License NPF-54, without attachments or appendices, for review and comment.
October 29, 1986	Letter to licensee forwarding October 21, 1986 notices of environmental assessment and finding of no significant impact regarding requests for exemptions from certain requirements of 10 CFR 50. Notices were forwarded to Office of Federal Register (FR) for publication.
October 31, 1986	Letter from licensee discussing utility's request for schedular exemptions to permit operation of facility before certain activities are completed. Utility officer will certify to NRC, 10 days before expiration date, that required activities have been completed.
October 31, 1986	Letter to licensee forwarding License NPF-54, FR notice of issuance of licensee, and Amendment 16 to Indemnity Agreement B-36. Pending Commission approval, operation is restricted to power levels not to exceed 5% of rated power.

November 11, 1986	Letter from licensee forwarding affidavit and additional information regarding MSIV removal and installation and schedule of activities. Actuator modifications will be made concurrently with fuel loading operations. Detailed MSIV modification schedule is also enclosed.
November 17, 1986	Letter from licensee discussing plan to remove actuators from installed MSIVs used to maintain secondary containment integrity when required. Ball will not be unseated when actuator is removed from the stem of the valve.
November 17, 1986	Letter from licensee forwarding response to Brookhaven National Laboratory (BNL) questions regarding 10 CFR 50.55(e) report on MSIV actuators and related information, per November 6, 1986 telephone request.
November 18, 1986	Summary of September 9, 1986 meeting with utilities regarding improving lines of communication between NRC and licensees, including Sholly process, interpretation of 10 CFR 50.59, Technical Specification improvement program, and discretionary enforcement.
November 20, 1986	Summary of October 15, 1986 meeting with utilities and Stone and Webster (S&W) regarding MSIV leakage problems. Hydraulic cylinders on modified MSIV actuators are not safety related. Utility is requested to perform an additional leak test.
November 20, 1986	Letter to licensee regarding removal of MSIV actuators from valves being used to maintain secondary containment integrity.
November 21, 1986	Letter from licensee advising staff that conclusion on acceptability regarding suppression pool bulk-to-local temperature remains unchanged, per NRC's November 6, 1986 request to review analysis documented in FSAR Section 6A.10. Analysis is performed in conformance with NUREG-0783.
November 24, 1986	Letter from licensee informing staff of changes in plant corporate structure providing more effective and efficient operation. Position of manager of corporate security has been abolished because of imminent retirement of J. J. Sunser. J. P. Beratta should receive correspondence on security.
November 25, 1986	Letter from licensee advising staff of change of address effective on December 1, 1986.
November 26, 1986	Letter from licensee certifying that reactor coolant and emergency core cooling system (ECCS) leak detection system is complete and operable, per requirement of schedular exemption to notify NRC 10 days before expiration of exemption regarding status of activity.

November 26, 1986	Letter to licensee requesting information to assess utility progress in completing actions required to meet GDC 2 and to schedule NRC review of downcomer issue, no later than 10 working days before Commission briefing. Information should include progress since February 18, 1986.
November 26, 1986	Letter from licensee forwarding comments on SSER 5, reflecting differences between SER and SSERs, FSAR through Amendment 27, and Technical Specifications. Comments do not address open issues. Enclosed FSAR changes will be incorporated into FSAR Amendment 28.
December 1, 1986	Letter to licensee forwarding "Interfacing System LOCA at BWRs," draft letter report for May 1986.
December 2, 1986	Letter from licensee forwarding proposed changes to site security and safeguards contingency plan. Changes are withheld (reference 10 CFR 73.21).
December 5, 1986	Letter from licensee providing status and confirmation of remaining commitments regarding solid radwaste system and schedule for completion. Full-scale preoperational test of asphalt-based volume reduction and solidification system will be performed.
December 5, 1986	Letter from licensee certifying completion of containment atmospheric monitoring system 10 days before expiration period as specified in listed exemptions. Actions are taken to ensure regulations regarding systems operability have been met as described.
December 5, 1986	Letter from licensee requesting exemption to defer operability of design-basis accident (DBA) hydrogen recombiner system until before exceeding 5% of rated power.
December 8, 1986	Letter from licensee forwarding additional information regarding 10 CFR 50.55(e) report on MSIV actuators. "MSIV Hydraulic Actuator Qualification," "Qualification of Modified MSIV Actuator, Specification P303D," and "Design-Report and Seismic Analysis of Solenoid-Operated Valve" enclosed.
December 9, 1986	Letter from licensee applying for amendment to License NPF-54, revising Technical Specifications regarding drywell and suppression chamber hydrogen recombiner system. Incorporation of amendment into full-power operating license upon issuance is requested.
December 9, 1986	Letter from licensee applying for amendment to License NPF-54, revising Technical Specifications regarding scram discharge volume operability. Change should be included in issuance of full-power operating license.

December 9, 1986	Letter from licensee applying for amendment to License NPF-54, revising Technical Specifications to correct typographical errors and provide clarification regarding isolation actuation instrumentation setpoints.
December 9, 1986	Letter from licensee forwarding attachment and associated FSAR change pages providing detailed discussion of current and proposed fire protection program. Enclosure addresses requirements of Generic Letter 86-10. Changed pages will be incorporated in first FSAR update.
December 10, 1986	Letter from licensee applying for amendment to License NPF-54, changing Technical Specification 4.6.2.1.c to reflect response to Humphrey concerns found in FSAR. Requests that amendment be incorporated into full-power license when it is issued.
December 10, 1986	Letter from licensee applying for amendment to License NPF-54, revising Technical Specifications 6.5.3.10.a and 6.5.3.10.c to increase maximum allowable time for issuance of safety review and audit board minutes and audit reports. Incorporation of change into full-power operating license is requested.
December 11, 1986	Summary of December 4, 1986 meeting with utility and S&W to discuss concerns regarding containment hydrogen recombiners and MSIVs.
December 16, 1986	Letter from licensee certifying completion for postaccident sampling system. Preoperational test of system completed, reviewed, and approved. Affidavit of Certification is enclosed.
December 16, 1986	Letter from licensee certifying completion of preoperational testing for containment inerting system, per April 7 and June 13, 1986 requests for deferral of preoperational testing of several systems. Preoperational testing is completed, reviewed, and approved.
December 16, 1986	Letter from licensee forwarding information on several activities associated with repair and modifications of MSIVs, per request. Revision 0 to Preoperational Test Procedure N2-POT-1-2, "MSIVs," is also enclosed.
December 18, 1986	Letter from licensee responding to utility's October 24, 1986 commitments per NRC's October 22, 1986 request regarding panel-mounted electronic equipment reliability in event of loss of redundant cooling and ventilation systems to control and relay rooms.
December 18, 1986	Letter from licensee requesting that issuance of full-power license not be delayed to incorporate six proposed Technical Specification requests submitted in December 1986. List of proposed Technical Specification submittals being withdrawn is enclosed.

December 19, 1986	Letter to licensee forwarding, for information, safety evaluation regarding BWR Owners Group document NEDE-31096-P, "ATWS; Response to NRC ATWS Rule, 10 CFR 50.62." Confirmation of applicability of topical report and response to listed concerns are requested within 90 days.
December 30, 1986	Letter from licensee forwarding Revision 2 to quality assurance (QA) program topical report, incorporating changes made since the December 1985 approval. Changes involve revisions to organization titles and responsibilities, and addition of new titles and responsibilities.
December 30, 1986	Letter from licensee submitting information on resolution of downcomer problem, including proposed schedules and discussion of options, per November 26, 1986 request.
January 2, 1987	Summary of December 18, 1986 meeting with utility and S&W regarding logic modifications to MSIV actuators due to December 3, 1986 full scram as result of loss of power to both reactor protection system scram sensor busses. Utility design modifications are unacceptable, per Regulatory Guide 1.6.
January 7, 1987	Letter from licensee forwarding response to NRC questions based on review of utility's December 16, 1986 letter regarding MSIVs and Section VIII concerning prototype testing schedule.
January 7, 1987	Letter from licensee informing staff of withdrawal of December 5, 1986 request for schedular exemption for DBA hydrogen recombiner systems. Final report 55(e)-86-22, dated January 6, 1987, regarding insufficient flow rate has been submitted to Region I. Requirements have been met. Affidavit is enclosed.
January 8, 1987	Generic Letter 87-01 to all power reactor licensees and applicants for operating license regarding public availability of NRC operator licensing examination question bank.
January 8, 1987	Letter from licensee forwarding Revision 0 to Procedure N2-MPM-R18, "MSIVs 2MMS HYV 6A, B, C, D and 2MSS HYV 7A, B, C, D," and Revision 0 to Procedure N2-CSP-17, "Hydraulic and Lubrication Oil Chemical Maintenance at Nine Mile Point Unit 2," per December 15, 1986 request.
January 12, 1987	Letter from licensee discussing integrity of systems outside containment, per License Condition 2.C(12). Summary of water testing results and corrective maintenance are per Technical Specification 6.8.4. Leak test results for reactor core isolation cooling (RCIC) are included.

January 14, 1987	Letter from licensee advising staff of status and future action regarding main steamline isolation ball valves. Confirmatory testing and evaluation program are continuing to ensure valves meet design function over lifetime.
January 14, 1987	Letter to licensee forwarding safety evaluation regarding MSIV logic modifications resulting from December 3, 1986 full scram, per December 18, 1986 and January 6, 1987 meetings. Modifications do not meet requirements of 10 CFR 50, Appendix A, GDC 21 or IEEE Standard 279. Issue must be resolved before initial criticality.
January 15, 1987	Letter from licensee forwarding design change for power supply to MSIV actuator solenoids, per IEEE Standard 279 and GDC 21 to 10 CFR 50, Appendix A. Response to NRC's January 9, 1987 concerns are also enclosed. Information will be incorporated into next revision to FSAR.
January 15, 1987	Letter from licensee clarifying December 10, 1986 request for change to Technical Specification Surveillance Requirement 4.6.2.1.c regarding operator action in response to second high-temperature alarm, per December 16, 1986 request. Immediate reactor trip in response to alarm is not required.
January 15, 1987	Letter from licensee forwarding revised list of submitted letters entitled, "Technical Specification Changes Not Impacting Full Power License." Revision supersedes and clarifies intent of attachment to December 18, 1986 submittal.
January 19, 1987	Letter from licensee informing staff that preoperational testing of asphalt-based volume reduction and solidification system is in final stages. Results will be submitted to NRC by February 2, 1987 and system will be declared operational on March 4, 1987.
January 23, 1987	Letter from licensee forwarding final summary report on MSIVs. Requests that 12-scfh acceptance criteria for each MSIV when tested between seats be reflected in Technical Specifications issued with full-power operating license. MSIVs are completed to meet requirements of regulations for exemption.
January 23, 1987	Letter from licensee responding to NRC's December 22, 1986 request for explanation of how redundant safety systems are verified as operable before taking sister safety systems out of service, per utility's November 26, 1986 request that NRC revise Page 15-2 of SSER 5.

January 27, 1987	Letter from licensee forwarding response to January 20, 1987 request for additional information on MSIV hydraulic actuator and acceptability of actuator hydraulic fluid when used with revised actuator design. Revised actuator operations manual will be provided to utilities by February 15, 1987.
January 27, 1987	Letter from licensee clarifying commitment regarding operation of MSIVs per Technical Specifications through first refueling outage. Balls and seats of valves are in substantially better condition than test ball and seats described in utility's January 14, 1987 letter.
January 27, 1987	Letter from licensee forwarding Revision C to "MSIV Phase I Test Specification Nine Mile Point Unit 2," for MSIV testing scheduled for February 2, 1987 at Crosby facilities. Comments are requested before test date.
January 29, 1987	Letter from licensee notifying staff that hydrogen recombiner system will be tested by metered makeup testing to comply with NUREG-0737, TMI Action Plan Item III.D.1.1 and FSAR Page 1.10-125, Item 2.
January 30, 1987	Letter from licensee certifying completion of turbine electrical hydraulic system and summarizing actions taken to ensure regulations are met regarding operability of system, per October 31, 1986 letter.
January 30, 1987	Letter from licensee requesting withdrawal of December 9, 1986 proposed change to Technical Specification 4.6.6.1.b.1. According to an NRC telephone conversation of January 13, 1987 with M. Haughey, the proposed change is not necessary for safe operation.
January 30, 1987	Letter from licensee certifying completion, review, and approval of preoperational test of offgas system regarding utility's May 2 and July 3, 1986 requests for schedular exemptions to defer operability of system until MSIVs opened following plant startup, per October 31, 1986 commitment.
February 3, 1987	Letter from licensee submitting changes to January 27, 1987 MSIV prototype test specification, per January 30, 1987 discussion with NRC staff. Revised specification and test results will be submitted by May 15, 1987.
February 4, 1987	Letter from licensee responding to request for additional information regarding maximum credible fault test of J-10 relay. Test procedure and summary of test results addressing NRC questions are enclosed.
February 6, 1987	Letter from licensee regarding MSIV leakage data.

February 6, 1987	Letter from licensee forwarding results of asphalt-based volume reduction and solidification system preoperational test.
February 26, 1987	Letter from licensee informing that temporary zone banding of certain meters was completed before exceeding 5% of rated power.
March 11, 1987	Letter from licensee committing to replace present main steam isolation valves with wye-pattern globe valves.
March 16, 1987	Letter from licensee requesting amendment to license to accommodate installation of new MSIVs.
March 25, 1987	Letter from licensee forwarding revised pages to FSAR Chapter 14, representing changes to initial startup test program.
March 27, 1987	Letter from licensee requesting supplemental application for amendment, superseding December 10, 1986 change request for Technical Specification Surveillance 4.6.2.1.c regarding suppression pool high-temperature-alarm setpoints.
March 31, 1987	Letter from licensee forwarding "Assessment Report on Need for Leakage Control System for Nine Mile Point Unit 2."
April 3, 1987	Letter from licensee forwarding "Assessment Report for Leakage Control System for Nine Mile Point Unit 2."
April 3, 1987	Letter from licensee responding to December 19, 1986 request for additional information regarding standby liquid control system.
April 7, 1987	Letter from licensee forwarding "Comparison of Electrical Design of Wye Pattern Globe Valve, Hanford 2 and River Bend Design."
April 7, 1987	Letter from licensee providing additional information regarding utility position for alternative to MSIV leakage control system.
April 10, 1987	Letter from licensee forwarding technical specification changes regarding fire protection for incorporation into FSAR.
April 13, 1987	Letter from licensee forwarding Revision 5 to Emergency Action Procedure EAP-1, "Activation and Direction of Emergency Plan."
April 13, 1987	Letter from licensee forwarding revised Technical Specifications in response to NRC letter dated March 4, 1987.

April 14, 1987	Letter to licensee regarding draft safety evaluation report for main steam isolation valve leakage control system.
April 16, 1987	Letter from licensee submitting additional information regarding drywell spray analysis.
April 16, 1987	Letter from licensee regarding commitments to take additional precautionary measures regarding control room ambient temperature limit.
April 23, 1987	Letter from licensee forwarding updated page and drawings to report related to electrical design of facility wye-pattern globe valves presently being installed.
April 24, 1987	Letter from licensee informing of utility's intention to delay submittal of amendment to FSAR.
April 28, 1987	Letter from licensee certifying that all transients and accident analysis affected by change from ball- to wye-pattern main steam isolation valves were performed by NRC-approved methodology and computer codes in FSAR.
April 28, 1987	Letter from licensee submitting application for amendment revising surveillance requirements regarding leakage-detection capabilities of containment airborne gaseous and particulate radioactivity detection and reactor vessel head flange leak-detection system.
April 30, 1987	Letter from licensee requesting amendment to Technical Specifications to increase accessibility of General Superintendent, Nuclear Generation position.
May 11, 1987	Letter to licensee issuing Amendment No. 1 to license regarding deletion of License Condition 2.C.(14) on the main steam isolation valves.
May 15, 1987	Letter to licensee issuing Amendment No. 2 to license related to the main steam isolation valves.
May 15, 1987	Letter from licensee forwarding copy of final report, "Downcomer Reanalysis Report."
May 18, 1987	Letter from licensee forwarding copy of report regarding failure modes and effects analysis (FMEA).
May 20, 1987	Letter from licensee requesting change to the "Administrative Controls" section of the Technical Specifications
May 20, 1987	Letter to licensee forwarding copy of draft full-power license for comment.

May 27, 1987	Letter from licensee regarding Technical Specification Surveillance 4.4.3.2.1.
May 27, 1987	Letter from licensee concerning withdrawal of April 28, 1987 requested change to Technical Specification Surveillance Requirement 4.4.3.2.1.
May 28, 1987	Letter from licensee enclosing Amendment 28 to the Final Safety Analysis Report.
May 28, 1987	Letter from licensee concerning Amendment 28 to the Final Safety Analysis Report.
May 29, 1987	Letter to licensee concerning IE Information Notice 86-98, regarding offsite medical services.
June 1, 1987	Letter from licensee responding to Generic Letter 87-06, "Periodic Verification of Leak Tight Integrity of Pressure Valves."
June 1, 1987	Letter from licensee concerning revised emergency procedures.
June 5, 1987	Letter from licensee concerning affidavit of service regarding Amendment 28 to the Final Safety Analysis Report.
June 11, 1987	Letter from licensee concerning certification of Appendix A, Technical Specifications.
June 15, 1987	Letter from licensee concerning status of arrangements for medical services in the offsite plans.
June 16, 1987	Letter from licensee providing commitments and FMEA related to electrical components in the Reactor Protection System.
June 23, 1987	Letter from licensee concerning request for exemption relating to the installation of certain redundant Class 1E protective devices and components.
June 23, 1987	Letter from licensee concerning electrical components with the General Electric scope of supply.
June 23, 1987	Letter from licensee concerning review of application of IEEE-279 isolation criteria.
June 25, 1987	Letter from licensee concerning additional information relating to neutron-monitoring system exemption request dated June 23, 1987.

APPENDIX B

REFERENCES

Institute of Electrical and Electronics Engineers, IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations," 1971.

NUS, "Topical Report on 10 CFR 61 Qualified Radioactive Waste Form," June 1984.

NUS Process Services Corporation, Topical Report PS-53-0378," Radwaste Solidification System," Rev. 0, April 1983.

U.S. Nuclear Regulatory Commission, Generic Letter 86-10, "Implementation of Fire Protection Requirements," April 28, 1986.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

---, NUREG-0783, "Suppression Pool Temperature Limit for BWR Containments," November 1981.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1981.

Waste-Chem Corporation, "10 CFR 61 Waste Form Conformance Program for Solidified Process Waste Product Produced by a Waste-Chem Corporation Volume Reduction and Solidification System," May 1986.

---, "Radwaste Volume Reduction and Solidification System," 1978.

APPENDIX D

ACRONYMS AND INITIALISMS

ADS	automatic depressurization system
AP	administrative procedure
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
B&W	Babcock & Wilcox
BWR	boiling-water reactor
CE	Combustion Engineering
CFR	Code of Federal Regulations
CHS	high-pressure core spray
CSL	low-pressure core spray
DBA	design-basis accident
DCRDR	detailed control room design review
ECCS	emergency core cooling system
EDO	NRC Executive Director for Operations
EOP	emergency operating procedure
EQMPDS	Equipment Qualification Maintenance Program Data Sheet
ESF	engineered safety feature
FMEA	failure modes and effects analysis
FPP	Fire Protection Program
FR	Federal Register
FSAR	Final Safety Analysis Report
GDC	general design criterion
GE	General Electric
HPCS	high-pressure core spray
HX	heat exchanger
I&E	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
INPO	Institute of Nuclear Power Operations
ISEG	Independent Safety Engineering Group
LFMG	low-frequency motor generator
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
MSIV	main steam isolation valve

NMP-1	Nine Mile Point Nuclear Station, Unit No. 1
NMP-2	Nine Mile Point Nuclear Station, Unit No. 2
NMPC	Niagara Mohawk Power Corporation
NMS	neutron-monitoring system
NPRDS	Nuclear Plant Reliability Data System
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NUTAC	Nuclear Utility Task Action Committee
PGCC	power generation control complex
P&ID	piping and instrumentation diagram
PMT	post-maintenance testing
QA	quality assurance
RCIC	reactor core isolation cooling
RHR	residual heat removal
RPS	reactor protection system
RRCS	redundant reactivity control system
RTS	reactor trip system
RWCU	reactor water cleanup
SDV	scram discharge volume
SEE-IN	Significant Event Evaluation and Information Network
SER	Safety Evaluation Report
SPDS	safety parameter display system
SRP	Standard Review Plan
SRV	safety/relief valve
SSE	safe shutdown earthquake
SSER	supplement to Safety Evaluation Report
S&W	Stone and Webster
TS	Technical Specifications
VETIP	Vendor Equipment Technical Information Program
WR	work request

APPENDIX E

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APPENDIX Q

ERRATA TO SAFETY EVALUATION REPORT AND ITS SUPPLEMENTS

Supplement 4

Page	Paragraph	Line	Change
1-4	--	--	Change SER Section "2.5.6.2.4" to "2.4.10" for Issue (2).

Supplement 5

Page	Paragraph	Line	Change
6-1	1	1	Change "quality" to "quantity".
9-3	3	2	Change "grap" to "grab".
10-1	5	2	Change "October 2, 1986" to "September 18, 1986".
11-3	Table 11.3		The number "1" should be listed for number of Floor Drain Collector Subsystem flatbed filters.
11-3	Table 11.3		The phase separator subsystem sample tank should be deleted.

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER (Assigned by TIDC, add Vol No., if any)
NUREG-1047
Supplement No. 6

2. TITLE AND SUBTITLE
Safety Evaluation Report related to the operation of Nine Mile Point Nuclear Station, Unit No. 2

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
July	1987

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
July	1987

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 7, above.

11a. TYPE OF REPORT
Technical

b. PERIOD COVERED (Inclusive dates)
October 1986 - July 1987

12. SUPPLEMENTARY NOTES
Docket No. 50-410

13. ABSTRACT (200 words or less)

This report supplements the Safety Evaluation Report (NUREG-1047, February 1985) for the application filed by Niagara Mohawk Power Corporation, as applicant and co-owner, for the license to operate Nine Mile Point Nuclear Station, Unit 2 (Docket No. 50-410). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near Oswego, New York.

This report supports the issuance of the full-power license for Nine Mile Point Nuclear Station, Unit No. 2.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

*U.S. GOVERNMENT PRINTING OFFICE: 1987-181-682:60152