NUREG-1047
Supplement No. 1

Reg Files

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

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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 the 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. Subject to favorable resolution of the issues discussed in this report, the NRC staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

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APPENDICES

FIGURE

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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 applicant) for a license to operate the Nine Mile Point Nuclear Station Unit 2 (NMP-2). This document is the first supplement to that SER (SSER 1). It includes the report of the Advisory Committee on Reactor Safeguards (ACRS). This supplement also provides the staff evaluation of outstanding issues that have been resolved and addresses changes to the SER that have resulted from the receipt of additional information.

Each of the sections and appendices of this supplement is designated the same as the related portion of the SER. Appendix A is a continuation of the chronology of this safety review and Appendix B lists reference materials cited in this document. Appendix D lists acronyms used in this supplement, and Appendix E lists the principal staff contributors. Appendices C, F, and G have not been changed by this supplement; however, a new appendix, Appendix H, has been added. Appendix H is a copy of the ACRS report, which is addressed in Section 19.

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 Ms. Mary F. Haughey. She may be reached by telephone at (301) 492-7897 or by mail at the following address:

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1.8 Outstanding Issues

The SER identified certain outstanding issues in the staff review that had not been resolved with the applicant at the time of issuance of that report. The list of those issues is reproduced in Table 1.3 with the current status of each issue.

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 has not yet been provided by the applicant. In these instances, the applicant has 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 open and the NRC staff will report on its resolution in a supplement to this report. Table 1.4 contains a list of confirmatory issues and their current status.

1.10 License Condition Items

There are certain issues for which a license condition may be desirable to ensure that staff requirements are met during plant operation. The license condition may be in the form of a condition in the body of the operating licenses, or a limiting condition for operation in the Technical Specifications appended to the licenses. One item (6) has been added to the list of license conditions. These currently defined license condition items are listed in Table 1.5.

Table 1.3 Outstanding issues

Issue		SER Section	Status
(1)	Snow loads	2.3.2	Open
(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	Open
(4)	Equipment qualification	3.10, 3.11	Open
(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	Open
(7)	Containment isolation	6.2.4	Open
(8)	Containment leak testing	6.2.6	Open
(9)	Containment fracture toughness (GDC 51)	6.2.7	Open
(10)	Postaccident monitoring instrumentation	7.5.2.2	Open
(11)	Separation criteria	8.4.5	Open
(12)	Safe and alternate shutdown	9.5.1.4	Open
(13)	Essential lighting	9.5.3	Open
(14)	Air start system	9.5.4, 9.5.6.	Open
(15)	Operations management	13.1, 13.4, 13.5	Closed, SSER 1
(16)	Procedures generation package	13.5.2	Open
(17)	Preoperational and startup test abstracts	14.0	Open
(18)	DCRDR and SPDS	18.1, 18.2	Open

Table 1.4 Confirmatory issues

Issue		SER Section	Status
(1)	Design of parapet scuppers on the roofs of safety-related buildings	2.4.2.2	Confirmatory
(2)	Construction quality control tests on revetment ditch	2.5.6.2.4	Confirmatory
(3)	Feedwater check valves	3.6.2	Confirmatory
(4)	Pipe break criteria	3.6.2	Confirmatory
(5)	Vertical floor flexibility	3.7.2, 3.7.3	Confirmatory
(6)	SRV/pool dynamic loads on containment interior structure	3.8.3	Confirmatory
(7)	Analytical results for the reactor internals for LOCA and SSE	3.9.2.4	Confirmatory
(8)	Results of Mark II hydrodynamic loads for NSSS piping, components, and equipment	3.9.3.1	Confirmatory
(9)	Leak rate test program	3.9.6	Confirmatory
(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	Confirmatory
(11)	Lead factors	5.3.1.2	Confirmatory
(12)	Verificaton of CONTEMPT LT/028 computer code	6.2.1.3	Confirmatory
(13)	Pool dynamics (a) Pool swell loads (b) Loads on submerged boundaries (c) Multivent, lateral load (d) CO and chugging loads inside the pedestal (e) Steam condensation submerged drag loads (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	6.2.1.7.3	Confirmatory

Table 1.4 (Continued)

Issue		SER Section	Status
(14)	Reverse flow testing	6.2.6	Confirmatory
(15)	Plant-specific LOCA analysis	6.3, 15.9.3	Confirmatory
(16)	Maximum hydrogen generation from the chemical reaction of the cladding with water or steam	6.3.5	Confirmatory
(17)	Instrument setpoints	7.2.2.3	Confirmatory
(18)	Anticipated transients without scram - mitigation system	7.2.2.4	Confirmatory
(19)	Minimum number of channels required to initiate protection actions	7.2.2.6	Confirmatory
(20)	Isolation of circuits	7.2.2.8	Confirmatory
(21)	Separation of Class 1E equipment and circuits	7.2.2.10	Confirmatory
(22)	Testing of protection systems instrumentation	7.3.2.5	Confirmatory
(23)	Manual initiation of RCIC	7.4.2.2	Confirmatory
(24)	Capability for safe shutdown following loss of electrical power to instrumentation and controls	7.4.2.4	Confirmatory
(25)	LPCI and LPCS injection valves interlocks	7.6.2.1	Confirmatory
(26)	Multiple control system failures	7.7.2.1	Confirmatory
(27)	High-energy-line breaks and consequential control systems failures	7.7.2.2	Confirmatory
(28)	Adequacy of station electric distribution system voltage	8.4.1	Confirmatory
(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	Confirmatory
(30)	Site visit confirmation that the 15-ft color-marking interval for cables is sufficient to verify their correct separation	8.4.5	Confirmatory

Table 1.4 (Continued)

Issue		SER Section	Status
(31)	Verification of the implementation of the electrical separation design criteria during site visit	8.4.5	Confirmatory
(32)	Review of analysis or design changes related to qualification of electrical equipment for flooding	8.4.7	Confirmatory
(33)	Portable radio communications demonstration	9.5.2	Confirmatory
(34)	Emergency lighting	9.5.3	Confirmatory
(35)	Procedures for filling fuel oil storage tanks	9.5.4.1	Confirmatory
(36)	Details of 1-in. vent line	9.5.4.1	Confirmatory
(37)	Division III diesel generator operation - severe conditions	9.5.4.1	Confirmatory
(38)	Fuel oil storage and transfer system - P&ID	9.5.4.2	Confirmatory
(39)	Procedures for maintaining diesel generator jacket water temperature	9.5.5	Confirmatory
(40)	Diesel generator interface on P&ID	9.5.5, 9.5.6	Confirmatory
(41)	Procedures for minimum loading of diesel generators	9.5.5	Confirmatory
(42)	Divisions I, II, and III diesel generator air-start systems	9.5.6	Confirmatory
(43)	Division III air dryer - installation and performance monitoring	9.5.6	Confirmatory
(44)	Fire damper control of combustion products	9.5.8	Confirmatory
(45)	Concrete dust control	9.5.8	Confirmatory
(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	Confirmatory
(47)	Alert and notification of the public within 15 minutes	13.3.2.5	Confirmatory

Table 1.4 (Continued)

Issue		SER Section	Status
(48)	EOF staffing	13.3.2.8	Confirmatory
(49)	Basis for recommendations for protective measures	13.3.2.10	Confirmatory
(50)	Compliance with ATWS rule (10 CFR 50.62)	15.8	Confirmatory
(51)	IE Bulletin 79-08 item 6 (II.K.1.5) and item 8 (II.K.1.10)	15.9.2	Confirmatory
(52)	Installation of equipment for the automatic restart of RCIC on low water level	15.9.3	Confirmatory
(53)	Modification of ADS logic (II.K.3.18)	15.9.3	Confirmatory
(54)	Installation of modification to RCIC pipe break detection circuitry (II.K.3.15)	15.9.3	Confirmatory
(55)	Integrity of systems outside containment likely to contain radioactive material (III.D.1.1)	15.9.4	Confirmatory

Table 1.5 License conditions

Issue		SER Section
(1)	Turbine system maintenance program	3.5.1.3
(2)	Thermal hydraulic stability analysis beyond Cycle 1	4.4.4
(3)	Fire protection	9.5.1.9
(4)	Operability of PASS system	9.3.2
(5)	Operation with partial feedwater	15.1
(6)	Operating experience on shift	13.1.2.1

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

In Section 3.6.2 of the Nine Mile Nuclear Station Unit 2 (NMP2) SER, the staff identified an open issue concerning a portion of the reactor-water cleanup (RWCU) piping system in the containment penetration area where the break exclusion criterion was applied. In accordance with Branch Technical Position (BTP) MEB 3-1 Position B.1.b, breaks need not be postulated in those portions of piping from the containment wall to and including the inboard and outboard isolation valves provided they meet the requirements of the ASME Code Section III Subarticle NE-1120 and the additional design requirements in BTP MEB 3-1 Position B.1.b. Specifically, Positions B.1.b(3) and B.1.b(4) respectively state that (1) the number of circumferential and longitudinal piping welds and branch connections should be minimized and (2) the length of these portions of piping should be reduced to the minimum length practical. The staff review of the NMP-2 RWCU piping in the break exclusion area did not appear to satisfy the above two staff positions and the staff requested that the applicant provide further justification for the extent of break exclusion applied to this piping. The staff was particularly concerned with the safety consequences related to the dynamic effects of a postulated break at the terminal end where the RWCU piping connects to the feedwater piping.

The applicant provided its response to the staff concerns in letters from C. V. Mangan (NMPC) to A. Schwencer (NRC) dated December 12, 1984 and January 21, 1985. Additionally, the staff met with the applicant on January 3, 1985 to discuss this issue in detail.

In the December 12, 1984 letter, the applicant provided the staff with the basis for determining that the length of the RWCU piping within the break exclusion area was the minimum length practical. Several studies were performed to demonstrate that the configuration of RWCU and feedwater piping provides the optimum operational flexibility while (1) maintaining low piping stresses and fatigue effects, (2) minimizing the number of fittings and length of pipe to the least practically achievable, and (3) meeting the guidelines of BTP MEB 3-1 Position B.1.b. Other configurations resulted in unacceptable piping stresses or would result in (1) significant changes in both the RWCU and feedwater system design or (2) a significant number of additional pipe supports and pipe whip restraints.

In the January 21, 1985 letter, the applicant evaluated the consequences of a postulated break at the thermal tee where the RWCU piping connects to the feedwater line. The conclusions of the applicant's study were as follows.

(1) The RWCU pipe would impact a maintenance platform at elevation 251 ft. This was found to be acceptable because the failure of the platform would

not result in secondary damage to other safety-related components and, thus, will not impact safe plant shutdown.

(2) The main steam containment penetration Z1A and the main steamline would be impacted by the whipping RWCU pipe. The effects were found to be acceptable and would not result in a rupture of the main steamline or containment penetration because of insufficient energy associated with the whipping pipe. Additionally, the applicant found that the function of the outside main steam isolation valve would not be impaired.

(3) The feedwater line was found to be the only potential target from jet impingement effects. The stresses in the feedwater line from jet impingement were found to be within the allowable stress limit and are, thus, acceptable. In addition, the applicant found that the feedwater isolation valves, containment penetration, and jet impingement wall would not be overloaded from the effects of jet impingement.

On the basis of results of the applicant's evaluations described above, the staff concludes that the applicant has sufficiently demonstrated that the extent of break exclusion applied to the RWCU piping in the containment penetration area is acceptable. The staff considers this item closed.

6 ENGINEERED SAFETY FEATURES

6.1 Materials

6.1.2 Protective Coating Systems (Paints) - Organic Materials

In the SER, the staff concluded in Section 6.1.2 that the organic materials inside containment were acceptable. The evaluation was based on the information provided in the FSAR through Amendment 13.

In Amendment 18, the applicant provided changes to the organic materials section. The changes consist of a reduction in the quantity of unqualified protective coatings inside containment.

A reduced quantity of unqualified coatings would produce less debris in a design-basis accident, thereby enhancing safety of the postaccident fluid systems. Therefore, the staff's previous conclusion in the SER that organic materials are acceptable remains the same.

6.2 Containment Systems

6.2.1.8 Steam Bypass of the Suppression Pool

The applicant provided, in FSAR Amendment 17, a revised analysis of the steam bypass to show that the containment design pressure will not be reached assuming that no containment spray operation occurs for the first 30 minutes following onset of a postulated accident. The Emergency Procedures Guidelines (EPG) will provide the operator with guidelines for actuation of the system consistent with the following assumptions which the applicant has provided as the basis for their analysis.

1. The postulated pipe break, consisting of (a) a limiting steam line break of 0.4 sq. ft., (b) loss of offsite power and (c) failure of Division II diesel generator, occur at time zero. The pipe break size was determined from an evaluation of a spectrum of breaks ranging in size from about 0.1 ft^2 to 3.0 ft^2.

2. The bypass capability of the wetwell is $A/\sqrt{K}$ = 0.05 sq. ft. in conformance with SRP (NUREG-0800) Section 6.2.1.1.C.

3. Feedwater is not added to the reactor vessel due to the loss of offsite power and unit trip.

4. Steam flow out the break is maximized by providing that the operator throttles ECCS flow starting at 10 minutes after the accident. It is also assumed to maximize steam flow that the operator does not initiate controlled reactor cooldown or actuate ADS. No operator action is assumed to occur that results in a temperature change in the reactor vessel of greater than 100°F/hr. Reactor pressure is maintained by automatic operation of

the SRV's in the power actuated relief mode. The applicant has also indicated that the containment pressure transient is abated at 30 minutes into the postulated LOCA solely by the containment spray actuation.

5. No heat or mass transfer takes place between the pool surface and the suppression chamber atmosphere.

6. Passive heat sinks absorb energy from the drywell and suppression chamber using the Uchida heat transfer correlation.

The results of the analysis indicate that containment sprays need to be fully operational at 30 minutes following the transient. The minimum design flow rate of 7,450 gpm (one loop available out of two) is required. Ninety five percent of the flow is directed into the drywell and five percent to the suppression chamber atmosphere. The manual initiation of the containment spray can be accomplished in approximately 2 to 4 minutes considering valve stroke times involved. This translates to the operator having to take actions at 26 to 28 minutes following onset of the postulated LOCA.

The results of this analysis indicate that the drywell reaches its peak pressure of 45 psig at 30 minutes into the transient while the wetwell peak pressure is 39 psig. Once the sprays are actuated, the drywell/wetwell pressures are reduced quickly to values well below the design limits.

To verify that this bypass capability will be available in the as-built condition, the applicant will perform a pre-operational leakage test at the design pressure differential between drywell and wetwell of 25 psid. The acceptance criteria should be 10 percent of $A/\sqrt{K}$ with $A/\sqrt{K} = 0.05\ ft^2$.

SRP Section 6.2.1.1.C also requires in-service verification of the bypass capability. Therefore, the staff will require a Technical Specification requirement that the applicant perform post-operational leakage tests at each refueling outage. The test differential pressure should be that corresponding to the vent submergence and the acceptance criteria should be 10 percent of the value assumed in the analysis, i.e., 10 percent of $A/\sqrt{K}$ of 0.05 sq. ft.

The staff has reviewed the analysis provided by the applicant including the assumptions listed above. The staff concludes that the steam bypass capability of Nine Mile Point, Unit 2, is acceptable. The Technical Specifications will contain provisions, as described above, to insure that the design bypass capability is achieved in the as-built condition.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Operations

13.1.2 Operating Organization

13.1.2.1 General

In its SER, the staff noted that the FSAR did not include the résumés of the individuals filling certain positions in the plant operating organization. FSAR Amendment 17 provided the missing résumés.

Of eight Station Shift Supervisors (SSSs), three have had senior reactor operator (SRO) licenses on the FitzPatrick plant or on the Nine Mile Point Nuclear Station Unit 1 (NMP-1) plant, and the others have all been licensed reactor operators (ROs) at those same plants. All eight Assistant Station Shift Supervisors (ASSSs) have baccalaureate degrees in technical or scientific disciplines, but none has commercial power reactor operating experience; only one has operational experience in the naval nuclear program. Of six Chief Shift Operators, four have experience in the operation of NMP-1. The Supervisor Radwaste Operations has a B.S. degree in chemistry and has 6 years' experience in health physics and radwaste systems.

In order to meet the guidelines of Generic Letter 84-16, "Adequacy of On-Shift Operating Experience for Near Term Operating License Applicants," the applicant should have, at the time of fuel load, each operating shift staffed with at least one senior operator with a minimum of 6 months of hot operating experience on a similar type plant, including startup and shutdown experience and at least 6 weeks' experience above 20% power. The staff's review of the résumés in the FSAR indicates that each of two SSSs has 3 years of hot operating experience as an SRO at FitzPatrick and three SSSs have from 2 to 4 years' hot operating experience as ROs, also at FitzPatrick. From the information provided in the FSAR, the staff is unable to determine how much hot operating experience the other three SSSs possess. It is apparent that at least five operating shifts can be staffed with SROs who meet the guidelines of Generic Letter 84-16.

On the basis of the résumés discussed above and those on which the staff reported in the SER, the staff concludes that NMP-2 will be operated by well-qualified individuals. However, to ensure that the required operating experience is represented on each shift at the time of fuel load, the staff will condition the operating license accordingly.

13.1.2.2 TMI Action Plan Items

NUREG-0737 Item I.A.1.1, Shift Technical Advisor

In its SER, the staff noted that the applicant had proposed to fulfill the NUREG-0737 requirements for a Shift Technical Advisor (STA) by using an Assistant Station Shift Supervisor (ASSS) in a dual role. The staff further noted that this would be acceptable provided that individuals used in the dual role have qualifications that meet the provisions of SECY-84-355. FSAR Amendment 18 includes a commitment by the applicant that the STA qualifications meet those provisions. Therefore, the staff concludes that the proposed dual role STA/SO is acceptable.

13.4 Operational Review

13.4.1 Review and Audit

In its SER, the staff noted that the review and audit functions proposed for NMP-2 were significantly different from those required for NMP-1 by the Unit 1 Technical Specifications and that they should be the same. FSAR Amendment 18 includes revisions to the functions of the Site Operations Review Committee (SORC) and the Safety Review and Audit Board (SRAB) that make these functions consistent between the two units. There are a few minor discrepancies, but the staff will eliminate these discrepancies from the Unit 2 Technical Specifications before the NMP-2 operating license is issued. Therefore, the staff considers this item closed.

13.5 Station Procedures

13.5.1 Administrative Procedures

13.5.1.1 General

In the SER, the staff noted that there was no indication in the FSAR that interdisciplinary review of procedures is conducted other than by the Site Operations Review Committee (SORC). FSAR Table 13.4-3, provided in FSAR Amendment 18, includes a description of the technical review and control process. This description will be the basis for a Technical Specification covering the review function. Included in FSAR Table 13.4-3 are statements to the effect that procedure review will include determination as to whether or not interdisciplinary review is necessary. If this additional review is needed, such review will be performed by the appropriate designated station personnel.

On the basis of the statements in FSAR Table 13.4-3, the staff concludes that there is reasonable assurance that the applicant will give adequate attention to interdisciplinary review of procedures. This item is closed.

The staff also noted in the SER that it was not clear in what areas of the control room the control room SRO could be located and still maintain appropriate supervisory contact with the reactor operator and the control and alarm panels. In FSAR Amendment 15, the applicant committed to assuring that the SRO would maintain visual and aural contact with all control room boards. The staff concludes that this commitment resolves the open item concerning where the control room SRO may be while performing his supervisory functions.

13.5.1.2 TMI Action Plan Items

NUREG-0737 Item I.C.5, Procedures for Feedback of Operating Experience

In the SER, the staff noted that the applicant had provided a detailed commitment to meet all of the requirements of TMI Action Plan Item I.C.5, but the information provided did not describe how outside information is obtained and who obtains it, how it is screened to ensure that it is sent to the people who need it, and how it is ensured that appropriate action has been taken by those to whom it has been sent. In FSAR Amendment 15, the applicant provided more detailed information about the process.

Operating experience assessment is performed by the Technical Support Group, the Operations Assessment Committee (OAC), and the Site Operations Review Committee (SORC). The Supervisor of the Technical Support Group is, as a member of OAC, the Operations Assessment Coordinator, and the OAC meets regularly with the SORC. An OAC designee is also a member of the Independent Safety Engineering Group. This organizational arrangement ensures appropriate coordination among these groups. The relationship between SORC and OAC is shown in Figure 13.1.

The OAC evaluates plant operations from a safety point of view, including equipment failures, design problems, operational errors, maintenance, testing, quality assurance, and procedures. Those involved in the assessment of operating experience will review information from a variety of sources, including operating information from the applicant's own plant(s), publications such as IE bulletins, circulars, and notices, and pertinent NRC or industrial assessments of operating experience.

The OAC prepares a draft report for presentation at a SORC meeting and a final report following the meeting. The report summarizes the applicable station operations experience that has been accumulated since the last meeting and documents the action taken at the station as a result of that experience. The final report is included in the minutes of the SORC for review by the Safety Review and Audit Board (SRAB) and members of the station staff. Action items from the final report are tracked on the SORC unfinished business items list. This list is followed up at later SORC meetings to ensure implementation.

On the basis of the additional information provided in FSAR Amendment 15, the staff concludes that the applicant's organizational arrangements will provide acceptable means for assuring that operating experience information will be provided to those who need it. This satisfies the requirements of TMI Action Plan Item I.C.5.

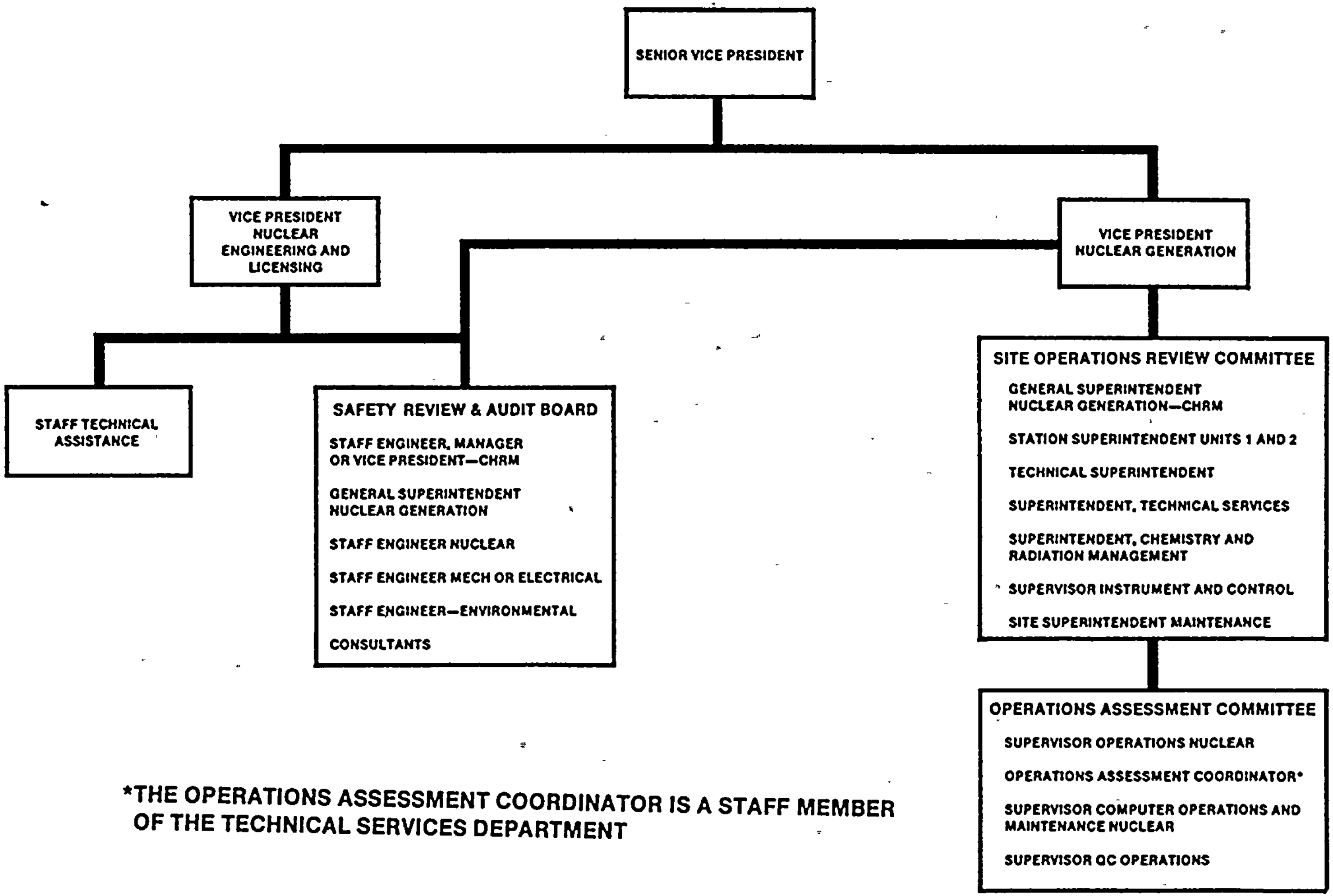

Figure 13.1 Safety review and audit of operations organization
Source: FSAR Figure 13.4-1

19 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) reviewed the application for an operating license for the Nine Mile Point Nuclear Station Unit 2 (NMP-2). A subcommittee toured the facility on February 20, 1985 and met in Syracuse, New York on February 20-21, 1985 to discuss the application. On March 7-9, 1985 the full Committee considered the NMP-2 application at its 299th meeting in Washington, D.C. A copy of the Committee's report to NRC Chairman Nunzio Palladino is reproduced as Appendix H to this SER supplement.

The Committee noted in its report to Chairman Palladino that the NRC staff had identified a number of outstanding issues that must be resolved before an operating license could be granted. These issues will be addressed in this and subsequent supplements to the SER.

The Committee also noted that NRC staff inspections in the period 1981-1983 revealed significant deficiencies in some areas of the construction quality assurance programs and some deficiencies in some areas of the construction of the plant itself. However, the Committee noted, too, that the applicant has taken extensive action to remedy these deficiencies. In addition, the Committee stated its belief that there is a reasonable basis for confidence that the quality of the completed plant will be adequate.

The Committee stated in its report the belief that, subject to the resolution of open issues identified by the NRC staff, and subject to the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Nine Mile Point Nuclear Station Unit 2 can be operated at power levels up to 3323 MWt without undue risk to the health and safety of the public.

APPENDIX A

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

December 12, 1984	Letter from applicant submitting a description of the reactor water cleanup and feedwater systems described in FSAR and a discussion of compliance with 3.6.2.
January 9, 1985	Generic Letter 85-01 to all power reactor licensees and all applicants for power reactor licenses regarding Fire Protection Policy Steering Committee report.
January 9, 1985	Summary of December 7, 1984 meeting with utility and S&W in Bethesda, MD regarding bypass leakage.
January 18, 1985	Letter to applicant responding to concerns in November 19, 1984 letter to facilitate decision of whether to appeal use of air dryers reviewed on plant-specific basis.
January 21, 1985	Letter from applicant providing study results of pipe whip and jet impingement consequences for break at terminal connection to feedwater thermal tee. Results will be incorporated in FSAR Amendment 18.
January 22, 1985	Letter to applicant advising that listed additional commitments regarding identification of damping values and assurance piping displacements can be accomodated with no adverse interaction with adjacent structures needed prior to completing review.
January 23, 1985	Letter from applicant forwarding information requested by Benedict during January 16 and 17, 1985 telcons. Matl describes functions of site operations review committee and safety review and audit board required by SECY84-355. Information will be incorporated in FSAR Amendment 18.
January 25, 1985	Letter from applicant forwarding July 18, 1984 LOCA and seismic analysis report for containment purge isolation valves. Valve orientation/travel limit recommendations will be implemented before fuel load. Nuclear seismic analysis report and seismic function test plan enclosed.
January 28, 1985	Generic Letter 85-03 to all BWR licensees and applicants regarding clarification of equivalent control capacity for standby liquid control system.
January 28, 1985	Letter to applicant forwarding request for additional information regarding April 1984 FSAR amendment revising

	snow loads for Category 1 structures. Response requested within 60 days of letter date.
January 29, 1985	Generic Letter 85-04 to all power reactor licensees and applicants for OL regarding operator licensing exams.
January 29, 1985	Letter to applicant forwarding request for additional information regarding compliance with TMI Action Item II.K.3.28 on verifying qualification of accumulator on automatic depressurization system valves. Response requested by 850325.
February 4, 1985	Letter from applicant informing that schedule for completion of remaining technical audits and additional information regarding design verification activities will be provided by February 11, 1985, per January 28, 1985 request.
February 6, 1985	Letter to applicant forwarding comments on detailed control room design review program plan, per NUREG-0737, Supplement 1 and NUREG-0700. Resolution of NRC concerns listed in enclosed recommended.
February 7, 1985	Letter from applicant forwarding information presented at January 28, 1985 meeting regarding engineering assurance in-depth technical audit program being performed by S&W. Final evaluation of audit data will be completed by 851007.
February 7, 1985	Letter from applicant forwarding "Design Action List (DAL)-17 Analyses of Non-Class 1E Devices Connected to Class 1E Power Supplies," summary report, per request to aid review of OL application. Information resolves FSAR Question 421.47.
February 11, 1985	Letter from applicant requesting NRC approval by March 1, 1985 of MSIV 7 exemption from 10 CFR 50.55a and ASME Section III requirements. Requested date based on evaluation of time to replace MSIV and still meet fuel load requirements.
February 11, 1985	Summary of January 3, 1985, meeting with applicant and S&W in Bethesda, MD regarding compliance with GDC 51 and break analysis of reactor water cleanup line.
February 12, 1985	Letter from applicant responding to January 22, 1985 letter, stating commitments regarding alternate damping values and verification of increased piping displacements. Commitments will be incorporated into FSAR Amendment 18.
February 14, 1985	Letter from applicant forwarding Amendment 17 to FSAR and Supplement 8 to environment report, incorporating certain responses and changes resulting from continuing review of FSAR and environment report.

February 15, 1985	Letter to applicant forwarding SER (NUREG-1047) and FR notice regarding application for OL. Continued work to resolve remaining open and confirmatory issues prior to ACRS meeting requested.
February 25, 1985	Letter from applicant forwarding change notice for FSAR Table 3.2.1 and additional clarification regarding refueling platform assembly installation requirements.
February 28, 1985	Letter from applicant forwarding Amendment 18 to FSAR for Nine Mile Point Unit 2.
March 4, 1985	Letter to applicant requesting additional information regarding TMI Action Item II.D.1 Description of plant piping configuration and description of how values regarding maximum backpressure as percentage of safety/relief valve and tee quencher flow obtained requested.
March 5, 1985	Letter from applicant forwarding results of QC test conducted upon completion of const of revetment ditch system, per Question F241.17. Results include quarry measurement and weight insp and sieve analyses conforming to ASTM C136-81.
March 5, 1985	Letter from applicant addressing concerns summarized in NRC February 6, 1985 letter regarding detailed control room design review program, per Supplement 1 to NUREG-0737.
March 5, 1985	Letter from applicant informing of plan to provide written submittal of plans for final S&W engineering assurance technical audit after March 15, 1985 meeting instead of on March 11, 1985 as indicated in previous correspondence.
March 5, 1985	Letter from applicant forwarding equipment qualification data packages, including mechanical equipment environment qualification package for PC vacuum relief valves and special service control valves.
March 7, 1985	Letter from applicant informing that Division I and II diesel generator air dryers will be installed per NRC January 1985 letter. Measures to ensure excess moisture does not get into air start system during preliminary tests stated.
March 15, 1985	Letter to applicant forwarding ACRS report on facility. Subject to resolution of open issues identified by staff and satisfactory completion of const staffing, testing, facility may operate up to power levels of 3323 MWt.
March 19, 1985	Letter to applicant forwarding request for additional information regarding response to Generic Letter 83-28, Items 2.1, 2.2.2, 3.1.3, 3.2.3 & 4.5.3. Response requested within 60 days for Items 2.1, 2.2.2, 3.1.3 & 3.2.3 and within 90 days for Item 4.5.3.

March 20, 1985	Letter to applicant forwarding basis for acceptance of measures to reduce amount of entrained water in air supply. Pressure in air receiver tank after condensate blowdown must be 25 psi below preblowdown pressure to assure air in air start system dry.
March 21, 1985	Letter to applicant forwarding preimplementation audit plan for evaluation of facility SPDS. Requests review of information and propose date for audit within 60 days.
March 22, 1985	Letter from applicant forwarding Vols 1 & 2 of S&W "Report of Findings of Independent Review of Key Technical, Interface and Const Concerns," per March 15, 1985 agreement.
March 26, 1985	Letter from applicant forwarding response to January 29, 1985 request for additional information regarding NUREG-0737, Item II.K.3.28, concerning automatic depressurization system valves. Information submitted will be incorproated into Amendment 19 to FSAR.
March 27, 1985	Letter to applicant approving application for authorization to utilize alternate to requirements of 10 CFR 50.55a(c)(1). Use of MSIV 7A acceptable as currently fabricated and installed. Actions to bring MSIV into total compliance with ASME Code listed.
March 29, 1985	Letter from applicant forwarding additional information regarding purge and vent valves, per March 29, 1983 request.
April 2, 1985	Letter to applicant forwarding approved ASME Code Case N-413, "Min Size of Fillet Welds for Linear Type Supports, Section III, Division I, Subsection NF," per February 25, 1985 request. Case Code N-413 may be used at facility for linear type supports with stated limitations.
April 3, 1985	Letter from applicant forwarding "Program for Completion of Engineering Assurance In-Depth Technical Audits, Nine Mile Point 2 Project" and overall audit plan for RCIC system and associated structures, in response to questions from February 27 and March 15, 1985 meetings.
April 4, 1985	Letter from applicant forwarding additional information regarding design basis for roof snow loading, in response to January 28, 1985 request. Responses 2 and 3 will be included in FSAR Amendment 19.
April 5, 1985	Letter to applicant discussing NRC review of deviations from BWR/5 STS submitted on November 20, 1984. Submittal should be revised to justify deviations based on FSAR, SER or as-built plant. Proposed Tech Spec review schedule enclosed.
April 8, 1985	Letter from applicant forwarding additional information regarding manual initiation of reactor core isolation

system for facility. Information will be incorporated into FSAR Amendment 20.

April 10, 1985	Letter from applicant clarifying February 25, 1985 letter regarding classification of refueling platform. Comparison of QA Category I requirements and QA activities actually employed on refueling platform provided in enclosed Table 1.
April 12, 1985	Letter from applicant forwarding package of updated seismic hydrodynamic qualification master list and response spectra. Equipment qualification audit regarding seismic and pump and valve operability requested in May 1985.
April 15, 1985	Letter to applicant advising that essential item of safety-related equipment not included in vital equipment list of physical security plan. Amendment requested within 45 days of date of letter.
April 16, 1985	Generic Letter 85-06 to all PWR licensees and all applicants for OLs regarding QA guidance for ATWS equipment not safety-related.
April 17, 1985	Letter to applicant forwarding SALP Report 50-410/85-99 for October 1983 - January 1985. Meeting scheduled for April 24, 1985 at Region I office to discuss assessment of SALP report and utility plans to improve performance.
April 19, 1985	Letter to applicant forwarding Mgt Meeting Report 50-410/85-09 on February 27 and March 15, 1985. Requests information regarding design audits performed to date on Reactor Controls, Inc. Forthcoming audit will be monitored at selected stages.
April 22, 1985	Letter from applicant forwarding revised Vols 1-4 of "Environment Qualification Document." More than 85% of required equipment qualified. Equipment qualification audit in May 1985 requested.
April 24, 1985	Letter from applicant advising that July 10 and 11, 1985 appropriate dates for preimplementation audit for SPDS requested in NRC March 21, 1985 letter.
April 30, 1985	Letter from applicant forwarding summaries of test reports and analyses from GE design record files. Tests and analyses furnished as additional response to NRC Question 421.47 and Confirmatory Item 21.
May 1, 1985	Letter from applicant forwarding Amendment 19 to FSAR for Nine Mile Point Unit 2.
May 2, 1985	Generic Letter 85-07 to all operating reactor licensees regarding implementation of integrated schedules for plant mods.

APPENDIX B

REFERENCES

U. S. Nuclear Regulatory Commission, Generic Letter 84-16, from D. G. Eisenhut to all licensees of operating reactors, applicants for operating license and holders of construction permits, "Adequacy of On-Shift Operating Experience For Near Term Operating License Applicants," June 27, 1984.

---, SECY-84-355, letter from W. J. Dircks to The Commissioners, "Final Policy Statement on Engineering Expertise on Shift," September 10, 1984.

APPENDIX D

ACRONYMS AND ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards, NRC
ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
ASSS	Assistant Station Shift Supervisor
B.S.	Bachelor of Science
BTP	branch technical position
ECCS	Emergency Core Cooling System
EPG	Emergency Procedure Guidelines
FSAR	Final Safety Analysis Report
LOCA	loss-of-coolant accident
MEB	Mechanical Engineering Branch
NMP-1	Nine Mile Point Nuclear Station Unit 1
NMP-2	Nine Mile Point Nuclear Station Unit 2
NRC	U.S. Nuclear Regulatory Commission
OAC	Operations Assessment Committee
RO	reactor operator
RWCU	reactor-water cleanup
SER	Safety Evaluation Report
SORC	Site Operations Review Committee
SRAB	Safety Review and Audit Board
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SSER	Supplement to the Safety Evaluation Report
SSS	Station Shift Supervisor
STA	shift technical advisor
SRV	Safety Relief Valve
TMI	Three Mile Island

APPENDIX E

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This supplement to the Safety Evaluation Report is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report.

NRC Staff	Title	Branch
M. Haughey	Project Manager	Licensing Branch 2
R. Benedict	Senior Nuclear Engineer	Licensee Qualification
J. Lane	Containment Systems Engineer	Containment Systems
D. Terao	Mechanical Engineer	Mechanical Engineering
F. Witt	Chemical Engineer	Chemical Engineering

APPENDIX H

ACRS REPORT ON THE NINE MILE POINT NUCLEAR STATION, UNIT 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 11, 1985

The Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE NINE MILE POINT NUCLEAR STATION, UNIT 2

During its 299th meeting, March 7-9, 1985, the Advisory Committee on Reactor Safeguards reviewed the application of Niagara Mohawk Power Corporation (the Applicant), acting on behalf of itself and as agent for Rochester Gas & Electric Corporation, Central Hudson Gas & Electric Corporation, New York State Electric & Gas Corporation, and Long Island Lighting Company, for a license to operate the Nine Mile Point Nuclear Station, Unit 2. The ACRS commented on the construction permit application for this Plant in a report dated July 17, 1973. Members of the Nine Mile Point, Unit 2 Subcommittee toured the facility on February 20, 1985 and met in Syracuse, New York on February 20-21, 1985 to discuss the application. During our review, we had the benefit of discussions with representatives and consultants of the Applicant, the General Electric Company, the Stone & Webster Engineering Corporation, and the NRC Staff. We also had the benefit of the documents listed.

Nine Mile Point Nuclear Station, Unit 2 is located in New York State on the shore of Lake Ontario immediately adjacent to Unit 1 and to the James A. FitzPatrick Nuclear Power Plant owned by the New York Power Authority. Unit 2 uses a boiling water reactor (BWR/5) with a rated power level of 3323 MWt. The nuclear steam supply system is similar to several other previously reviewed BWRs, such as Washington Public Power Supply System, Unit 2; Wm. H. Zimmer Nuclear Station, Unit 1; and La Salle County Station, Units 1 and 2. The primary containment is a Mark II, steel lined, reinforced concrete design with multiple downcomers connecting the drywell to the water-filled pressure suppression chamber. Condenser cooling for Unit 2 is provided from a counterflow, natural-draft, hyperbolic concrete cooling tower. The ultimate heat sink for emergency core cooling is Lake Ontario.

Construction of Unit 2 is about 86 percent complete and the Applicant currently estimates the fuel load date to be February 1986.

The Niagara Mohawk Power Corporation has operated Unit 1 for about 15 years and at one time operated the FitzPatrick Plant for the then Power Authority of the State of New York.

During our meeting, the NRC Staff identified a number of open issues that must be resolved prior to granting an operating license. We believe that these can be resolved in a manner satisfactory to the NRC Staff.

NRC Staff inspections in the period 1981-83 revealed significant deficiencies in some areas of the construction quality assurance programs, and some deficiencies in the construction of the plant itself. The Applicant has taken prompt and extensive measures to remedy these deficiencies. There have been major reorganizations and changes in personnel at all levels of the quality assurance organization, and the Applicant now has a strongly stated and clearly evident dedication to both quality and the assurance of quality. The NRC Region I Staff considers the current program to be generally acceptable but will, of course, continue its inspection and oversight functions. We believe that there is a reasonable basis for confidence that the quality of the completed plant will be adequate.

We believe that, subject to the resolution of open issues identified by the NRC Staff, and subject to the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Nine Mile Point Nuclear Station, Unit 2 can be operated at power levels up to 3323 MWt without undue risk to the health and safety of the public.

Sincerely,

David A. Ward
Chairman

References:

1. Niagara Mohawk Power Corporation, "Final Safety Analysis Report, Nine Mile Point Nuclear Station, Unit No. 2," Volumes 1-28 and Amendments 4-13 and 15-17
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Nine Mile Point Nuclear Station, Unit 2," USNRC Report NUREG-1047, dated February 1985

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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 a license to operate the 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. Subject to favorable resolution of the items discussed in this report, the NRC Staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

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