December 21, 1992

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION

In the Matter of	)	
NIAGARA MOHAWK POWER COMPANY (Nine Mile Point Nuclear Station, Unit 1)	)	Docket No. 50-220 DPR-63

RESPONSE OF NIAGARA MOHAWK POWER COMPANY TO BEN L. RIDINGS' § 2.206 PETITION TO SHUT DOWN NINE MILE POINT UNIT 1

I. INTRODUCTION

On October 27, 1992, Ben L. Ridings ("Petitioner") filed a "Petition for Emergency Enforcement Action and Request for Public Hearing" ("Petition") in the captioned matter. Petitioner requests, pursuant to 10 C.F.R. § 2.206, that the Nuclear Regulatory Commission ("NRC") take emergency enforcement action against Niagara Mohawk Power Corporation ("Niagara Mohawk"), including issuance of an immediately effective order requiring shutdown of the Nine Mile Point Nuclear Station, Unit 1 ("NMP-1") and a public hearing prior to the resumption of operation. By letter dated November 19, 1992, the NRC requested that Niagara Mohawk provide its views regarding Mr. Ridings petition.

Petitioner asserts that Niagara Mohawk is not in compliance with NRC regulations governing availability of an emergency core cooling system ("ECCS") because the plant's high-pressure coolant injection ("HPCI") system is neither classified as safety-related nor supported by an onsite backup power supply. Petitioner cites, as the basis for its assertions, 10 C.F.R. § 50.46 and several of the General Design Criteria ("GDC") in 10 C.F.R. Part 50, Appendix A. In addition, Petitioner alleges administrative deficiencies in Niagara Mohawk's treatment of certain containment isolation valves at NMP-1.

On December 4, 1992, Thomas E. Murley, Director, Office of Nuclear Reactor Regulation, wrote to Mr. Ridings stating that the Commission had declined to take direct review of the Petition and had referred the Petition to the NRC Staff for consideration pursuant to 10 C.F.R. § 2.206. After discussing the protection of NMP-1 against a loss of coolant accident (LOCA), Dr. Murley concluded "that there is no basis to issue an immediately effective order to shut down NMP-1 because of the unavailability of an engineered safety feature system grade HPCI system." Dr. Murley similarly concluded that an immediately effective order to shut down NMP-1 was not warranted on the basis of the identified administrative deficiencies with the containment isolation valves.

This response provides Niagara Mohawk's technical and legal bases for the conclusion that, as to the matters alleged in the Petition, NMP-1 has been and is being safely operated. In

particular, it provides the basis for the conclusion that the NMP-1 ECCS system meets all applicable NRC requirements and is capable of performing its intended function. The response also sets forth the basis for the conclusion that the administrative deficiencies relating to certain valves have been or are being corrected and in any event, do not affect their ability to operate. Thus, the health and safety of the public is protected. The response also refutes the Petitioner's allegation that Niagara Mohawk was notified of certain of these issues in January 1990 and took no action. For the reasons stated herein, the Petition should be denied.

II. LEGAL STANDARD FOR REVIEW OF 2.206 REQUESTS

The institution of a proceeding or the initiation of an enforcement action in response to a request for action under 10 C.F.R. § 2.206 is appropriate only when substantial health and safety issues have been raised.[1] Arizona Public Service Co. (Palo Verde Nuclear Generating Station, Units 1, 2, and 3), DD-92-1, 35 N.R.C. 133, 143-44 (1992); Consolidated Edison Co. of New York

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[1] Petitioner's letter included a request for an immediately effective shutdown order. Under NRC regulations, the Commission may issue an immediately effective order if it finds "that the public health, safety, or interest so requires or that the violation or conduct causing the violation is willful." 10 C.F.R. § 2.202(a)(5). Because the Commission has already determined that Petitioner's allegations do not merit such extraordinary relief (see Letter to Ben L. Ridings from Thomas E. Murley, Director, NRC Office of Nuclear Reactor Regulation, dated December 4, 1992), we do not further discuss the application of this standard.

(Indian Point, Units 1, 2, and 3), CLI-75-8, 2 NRC 173, 176 (1975); Washington Public Power Supply System (WPPSS Nuclear Project No. 2), DD-84-7, 19 N.R.C. 899, 923 (1984). This standard has been recognized by the courts. Florida Power & Light Co. v. Lorion, 470 U.S. 729, 732 (1985). ("Commission interprets §2.206 as requiring issuance of an order to show cause when a citizen petition raises substantial health of safety issues.") "A mere dispute over factual issues does not suffice." Northern Indiana Public Service Co. (Bailly Generating Station, Nuclear-1), CLI-78-7, 7 NRC 429, 433 (1978).

In considering a request under 10 C.F.R. § 2.206, the NRC Staff's proper focus is on its overriding regulatory responsibilities to ensure adequate protection of the public health and safety. See Houston Lighting and Power Co. (South Texas Project, Unit 1), DD-88-9, 27 N.R.C. 648, 649 (1988), citing Power Reactor Development Co. v. Int'l Union of Elec., Radio, and Machine Workers, 367 U.S. 396, 406 (1961). These cases make clear that the initiation of a hearing or an extreme enforcement action such as a shutdown order or license suspension is not justified by the mere allegation of a problem. Rather, the standard is whether reasonable assurance continues to exist that operation can be conducted safely.

The Petitioner in the present case asserts that NMP-1 is in violation of NRC requirements for its failure to maintain a safety-related HPCI system as part of the plant's ECCS system for

coping with a loss-of-coolant accident. As is demonstrated below, Unit 1 is in compliance with all applicable regulations. Contrary to Petitioner's claims, the NRC's General Design Criteria are not applicable to Unit 1. Even if it were true that the plant is in violation of NRC requirements, a point which is clearly incorrect, the Petitioner's further assertion, that shutdown of NMP-1 is appropriate because compliance with NRC regulations is a prerequisite to safe operation of a nuclear power plant, runs counter to the prior decisions of the agency. The Commission has plainly stated, to the contrary, that "a violation of a regulation does not of itself result in a requirement that a license be suspended." Petition for Emergency and Remedial Action, CLI-78-6, 7 N.R.C. 400, 405 (1978); Petition for Shutdown of Certain Reactors, CLI-73-31, 6 A.E.C. 1069, 1071 (1973).[2]

Petitioner cites two NRC Atomic Safety and Licensing Appeal Board decisions, Maine Yankee Atomic Power Co., 6 A.E.C. 1003, and Vermont Yankee Nuclear Power Corp., 6 A.E.C. 520, (Petition, at p. 14), apparently in support of the proposition that failure to comply with an NRC regulation requires immediate plant shutdown. However, those decisions establish that "in order for a facility to be licensed to operate, the applicant must establish that the facility complies with all applicable regulations."

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[2] See also NRC General Enforcement Policy, 10 C.F.R. Part 2, Appendix C, Section VI.C(2) (February 1992) ("Ordinarily, a licensed activity is not suspended . . . for failure to comply with requirements where such failure is not willful and adequate corrective action has been taken").

Vermont Yankee Nuclear Power Corp., 6 A.E.C. at 528 (emphasis supplied). These decisions, from initial licensing proceedings, are clearly not of relevance to the matter at hand, which involves a licensed, operating reactor. To the contrary, as noted above, in a Section 2.206 setting, neither a violation of a regulation nor a degraded condition would of itself automatically result in a requirement that the plant be shut down. See Ohio v. Nuclear Regulatory Comm'n, 814 F.2d 258, 264 (6th Cir. 1987) ("we recognize that it may be more difficult to shut a plant down than to prevent initial licensing") (NRC acted reasonably in denying petitioner's request to intervene in full-power licensing proceeding). Precedent further establishes that, when faced with an identified safety concern, so long as the NRC (and licensee) is pursuing measures to address the concern, it is proper for the NRC to allow plants to continue operation -- provided there is continued assurance of adequate protection of the public health and safety while the concern is addressed. See Nader v. NRC, 513 F.2d 1045 (D.C. Cir. 1975).

In the present case, Petitioner has failed to meet the threshold required by NRC precedents to justify the extraordinary relief requested. Furthermore, NMP-1 remains in compliance with applicable regulations and license conditions. The Petitioner has identified no information that was not already available to Niagara Mohawk and the NRC Staff. Neither has he provided any basis to call into question the Licensee's compliance with the regulations.

With the above considerations in mind, Niagara Mohawk addresses below each of the Petitioner's allegations to demonstrate that no "substantial health and safety issue" has been raised and that the initiation of a proceeding to further consider these matters is therefore not warranted.

III. HISTORY OF PETITIONER'S ALLEGATIONS

Petitioner's assertions are not unfamiliar to Niagara Mohawk; he made similar allegations in 1990 following his brief involvement as an employee of a contractor at NMP-1. And, contrary to Petitioner's assertions, Niagara Mohawk devoted considerable resources to addressing those concerns. In 1989, Niagara Mohawk contracted with MDM Engineering for a review of the NMP-1 surveillance procedures to determine whether they met Technical Specification requirements and to factor equipment qualification preventative maintenance requirements into the existing Technical Specification surveillance testing matrix. Petitioner was employed by MDM Engineering in November 1989 to assist in these efforts. Petitioner's assignment was always considered short term. The portion of the project on which he worked was to be completed by December 31, 1989.

Niagara Mohawk's Regulatory Compliance Group was responsible for evaluating and responding to the reports and items identified by MDM. Technical reviews were conducted to determine the validity of the matters raised. Some issues were evaluated through the issuance of problem reports and others by the

Regulatory Compliance Group. MDM Engineering's work was completed in January 1990. All concerns were resolved, or an appropriate resolution was initiated.

It is Niagara Mohawk's practice to encourage personnel at exit interviews to make use of Niagara Mohawk's Quality First Program ("Q1P") if concerns exist. Niagara Mohawk's Q1P program is designed to give employees a confidential forum for reporting of potential problems that affect quality or safety on the job. However, Mr. Ridings did not take advantage of the opportunity to raise any issues at that time. It was not until July 31, 1990, six months after his assignment ended, that Q1P was approached by the Petitioner with concerns relating to Inservice Testing (IST) program deficiencies. At the time, Petitioner stated that his concerns warranted an immediate investigation and response by Niagara Mohawk. On August 6, 1990, Q1P notified the NRC Resident Inspector's office of the receipt of this Q1P concern.

The Q1P group performed a comprehensive investigation of Petitioner's July 31, 1990 claims. The review included interviews with personnel, review of data and programs, along with telephone conversations with the concernee. The Q1P review observed that where differences in the FSAR and Technical Specifications existed, the more stringent requirement was used for IST. Based on interviews and documented corrective action measures by Niagara Mohawk, the Q1P group concluded that Petitioner's concerns had been adequately addressed. The Q1P group informed the Petitioner of

these conclusions, and he acknowledged the feedback received.

In addition, from September 17, 1990 through October 5, 1990, approximately the same time the concern was being investigated, the Nine Mile Point Quality Assurance Department performed an audit of the Nine Mile Point ASME Section XI Program. The Unit 1 portion of this audit focused on IST Code compliance. The audit team found no violations of the Code. It also should be noted that numerous reviews were performed during our extended refueling outage (January, 1988- July, 1990) relating to Specific Issue 17 of the NMP-1 Restart Action Plan. The reviews verified that the IST program for pumps and valves was in compliance with the applicable requirements of ASME Section XI.

On December 20, 1990, Petitioner sent another letter to Niagara Mohawk, this time requesting copies of proposed FSAR changes, the Q1P investigation results, and other information concerning his allegations. Although it is not the policy of the Q1P program or Niagara Mohawk to provide such documentation, Niagara Mohawk attempted numerous times to contact Petitioner to provide additional verbal feedback on the investigation results. Niagara Mohawk personnel left a number of messages on Petitioner's telephone answering machine. The last call documented was on April 1, 1991. Petitioner never responded.

Contrary to Petitioner's assertion in the Petition that Niagara Mohawk took no action on the issues he raised in 1990, Niagara Mohawk promptly and aggressively pursued such matters. The

Q1P investigation concluded that appropriate actions had been taken for each of the issues raised by Petitioner at that time. Thus, this allegation is without merit.

IV. DESCRIPTION OF NMP-1 HEAT REMOVAL SYSTEMS

The design of NMP-1, as well as all other light water reactor nuclear power plants, provides for heat removal systems as part of the reactor coolant system to assure that heat may be removed from the reactor under normal operating conditions, during transients and postulated accidents and under shutdown conditions. Specifically, NMP-1 is equipped with an isolation condenser with redundant components to remove heat from the reactor system following isolation from the main condenser, and a core spray system to remove heat from the core following a loss-of-coolant accident.[3/] In order to understand the role of the HPCI mode of the feedwater system at NMP-1, it is necessary to review the role of both these systems.

[3/] Loss-of-coolant accidents are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a double-ended rupture of the largest pipe in the reactor coolant system. 10 C.F.R. § 50.49(c)(1).

A. Emergency Cooling System -- Isolation Condenser

The emergency cooling system provides for decay heat removal from the reactor fuel in the event that reactor feedwater capability is lost and the main condenser is not available. While a number of boiling water reactors rely on systems having active components such as steam turbines and pumps for this function, at NMP-1 the isolation condenser operates by natural circulation[4/] to provide an alternate heat sink to the main condenser. The emergency cooling system operates automatically upon receipt of a high-reactor-pressure or low-low reactor water level signal. It may also be initiated manually from the main control room or from two remote shutdown panels (one panel for each emergency cooling loop). FSAR Section V, p. V-29.

One half of the isolation condenser system is adequate for decay heat removal following isolation of the reactor from the main condenser. Should one of the two independent cooling loops fail, half of the system can be isolated while the other half remains fully functional. Even in the event of a total loss of ac power with no makeup whatsoever to the reactor vessel, operation of the isolation condenser would proceed normally. FSAR Section V, p. V-30.

4/ During operation of the emergency cooling loops, steam rises from the reactor vessel to the condenser tubes where it is condensed by boiling the condenser shell water. As the water condenses, it returns by gravity flow to the suction of a reactor recirculating pump and from there to the reactor vessel.

Since the isolation condenser system is passive and does not require makeup water for several hours after its initiation, it is extremely reliable. In fact, a similar system is being proposed for inclusion in the design for certain new reactors. In any event, the NMP-1 emergency cooling system meets Commission requirements and assures the protection of the public health and safety.

B. Core Spray System

The principal emergency core cooling system (ECCS) at NMP-1, the core spray system, is designed to prevent overheating of the reactor fuel following a postulated loss-of-coolant accident. This system consists of two separate and independent core spray loops; each of those loops contains redundant active components. Core spray water comes from the suppression chamber which is within primary containment and returns to the suppression chamber after cooling the core. Makeup is available to the suppression chamber pool from the condensate storage and transfer system if necessary. In the event of a postulated total loss of the core spray primary water source (loss of suppression pool water below the core spray pump suction level), lake water can be supplied to the core spray nozzles to provide an alternate source of core cooling. FSAR Section VII, p. VII-4.

The core spray system is designed to respond to a full range of loss-of-coolant accidents, from the smallest to the largest line break. Following a large break LOCA, reactor pressure

decreases rapidly and the reactor water level drops. This triggers a signal which would initiate operation of the core spray system.

For a certain range of small line breaks which are larger than the capacity of the control rod drive pumps, reactor pressure may not decrease rapidly enough to allow timely operation of the core spray system. The plant is therefore equipped with an automatic depressurization system ("ADS") to depressurize the reactor more rapidly so that operation of the core spray system can be initiated more rapidly. The ADS, like the core spray system, is a safety-related system. Redundancy is built into the ADS, with the ADS logic circuit supplied from two 4160V power boards. If ac power were lost to either power board, one logic system would become inoperable, but the other would remain operable and capable of initiating automatic depressurization. The ADS can also be initiated manually. The plant's FSAR, Section XV-C, shows that NMP-1 has the capability to maintain adequate core cooling over the entire range of analyzed breaks. Thus, the assertion by Petitioner that the HPCI system is required for a range of line breaks is without basis. Similarly, Petitioner's assertion that without the HPCI system there would be a "meltdown" is completely lacking foundation.

The ECCS is designed to prevent the fuel cladding from exceeding the limits specified in 10 C.F.R. § 50.46. The ECCS helps limit the adverse effects of a LOCA by mitigating fuel damage which in turn provides protection to the public against release of

radioactive material to the environment. As the Petitioner recognizes, the HPCI mode of the feedwater system is not credited nor needed to demonstrate compliance with the 10 C.F.R. § 50.46 acceptance criteria. To reiterate, NMP-1 was evaluated by the NRC and found to meet the ECCS acceptance criteria with reliance on ECCS equipment other than the HPCI and these systems provide adequate protection during all LOCAs.

C. HPCI Mode of Feedwater System

As noted by the Petitioner, the High Pressure Coolant Injection system at NMP-1 is not part of the emergency core cooling system. The HPCI system at NMP-1 is a mode of the feedwater system. It is not considered an engineered safety feature and for the purposes of evaluation of accident scenarios its operation is not assumed. Furthermore, it is not relied upon to demonstrate compliance with NRC regulations governing ECCS capability, including 10 C.F.R. § 50.46 and Appendix K to Part 50. Rather, it serves to complement the existing safety systems and in some circumstances obviate their use. The HPCI system has two condensate pumps, two feedwater booster pumps, and two motor driven feedwater pumps available. In the HPCI mode, one train (a condensate pump, a feedwater booster pump and a motor driven feedwater pump) is utilized with the other redundant train available as backup. The feedwater pumps can be supplied with normal off-site power from either of the two 115 kV systems.

Therefore, the HPCI system at Nine Mile Point Unit 1 has a high degree of redundancy.

The HPCI system has the ability to provide large amounts of makeup water into the reactor vessel at reactor pressure. HPCI provides a reliable high pressure source of makeup water to the reactor and will thus maintain water level during a small-break LOCA. Therefore, for a certain range of accident sizes, the HPCI system could minimize the need to use the ADS, which would be otherwise required to enable operation of the core spray system following a small-break LOCA.

Because the HPCI system is not credited as a source of makeup water during a LOCA, it need not meet the requirements for an engineered safety feature, be powered by onsite power sources, or be included in the IST program. Nonetheless, the HPCI mode of the feedwater system is a highly reliable source of makeup water in the event of an accident. First, the feedwater system is normally operating during power operation and its components have redundancy and a high degree of reliability; thus its active components would be available to function as an HPCI system if called upon to do so. Even though HPCI is a non-safety related system, it is treated as quality related ("Q") to identify its inclusion in the Niagara Mohawk's Quality Related Program for the Nine Mile Point Nuclear Stations. The facility Technical Specifications have limiting conditions of operation for this system and thus availability of components is further assured. Operation and surveillance of the

HPCI mode of the feedwater system is conducted in accordance with procedures N1-OP-16, "Feedwater System Booster Pump to Reactor", N1-ST-Q3, "High Pressure Coolant Injection Pump and Check Valve Operability Test", and N1-ST-C3, "High Pressure Coolant Injection (HPCI) Automatic Initiation Test". Moreover, the reliability of offsite power at Unit 1 is quite high; since January 1, 1990 both 115 kV lines were simultaneously out of service for a total of only 45 seconds. This exhibits a 99.99995% availability of offsite power over that time frame. In the event of a loss of off-site 115 kV lines the system would realign itself automatically to the Bennetts Bridge Hydro Station to restore power. The coincidence of a LOCA with a loss of offsite power would be an extremely rare event. In any event, in that situation, the ADS and Core Spray systems would work in conjunction to assure adequate core cooling.

V. NMP-1 ECCS LICENSING BASIS

On August 22, 1969, Niagara Mohawk was issued a Provisional Operating License ("POL"), authorizing it to operate NMP-1 at power levels not to exceed 1538 megawatts (thermal) in accordance with the provisions of the license and the Technical Specifications. The results of the evaluation of the Staff of the Atomic Energy Commission ("AEC"), predecessor regulatory agency to the NRC, of Nine Mile Point Unit 1 was provided to the AEC's Advisory Committee on Reactor Safeguards ("ACRS") on March 24, 1969. With regard to the Commission's General Design Criteria, it stated:

> We recognize that the NMP facility was not designed in accordance with the current set of the Commission's General Design Criteria. However, as discussed in our evaluation, the inherent features and capability provide a basis for reasonable assurance that the facility design meets the intent of the criteria.

In subsequent licensing submittals, most notably the Technical Supplement to Petition for Conversion from the Provisional Operating License to the Full Term Operating License, submitted on June 27, 1972, Niagara Mohawk provided information relating to the extent to which NMP-1 conforms to the GDC. At no time did Niagara Mohawk commit to full compliance with the GDC.

On January 4, 1974, the Commission published its acceptance criteria for Emergency Core Cooling Systems for light water power reactors (39 Fed. Reg. 1003). This new rule added Appendix K to 10 CFR Part 50 which specifies analytical techniques to be employed for the evaluation of ECCS acceptability. The NMP-1 ECCS reanalyses utilized a Commission accepted ECCS evaluation model. See NRC Safety Evaluation Report December 27, 1974. In Amendment No. 5 to the Nine Mile Point Unit 1 Facility Operating License, dated November 13, 1975, which further evaluated NMPC's ECCS reanalysis, the NRC determined that operation of NMP-1 reactor will meet the requirements of Appendix K to 10 CFR Part 50, and 10 C.F.R. § 50.46. That conclusion was reached without taking credit for the HPCI System.

This conclusion has been reconfirmed by General Electric Company's ("GE") loss-of-coolant accident analysis for each

subsequent fuel cycle. Each analysis, including the one for the present fuel cycle, did not take credit for the HPCI mode of the feedwater system in determining compliance with 10 CFR § 50.46. For the present cycle, GE's analysis was prepared in response to the requirements of Nine Mile Point Unit 1 Technical Specification 6.9.1.f, "Reporting Requirements, Core Operating Limits Report."

Petitioner's assertion which permeates the Petition that "[t]here is a minimum requirement for a High Pressure Core Injection ECCS Safeguard System at the Nine Mile Point Unit 1 facility" is simply incorrect. LOCA analyses performed by Niagara Mohawk to demonstrate compliance with 10 C.F.R. § 50.46 do not take credit for the HPCI system. FSAR Section VII, P. VII-61. Nevertheless, the plant's ECCS configuration satisfies the requirements of § 50.46 and 10 C.F.R. Part 50, Appendix K. Thus, the fact that the HPCI system is not safety-related and relies on offsite power sources does not detract from the finding that the NMP-1 ECCS provides reasonable assurance of the public health and safety.

Contrary to Petitioner's assertions, the NMP-1 emergency core cooling system is not subject to the General Design Criteria contained in 10 C.F.R. Part 50, Appendix A. Petitioner refers to GDC 33, 35, 36, and 37, which like the other GDCs, became effective on May 21, 1971, after NMP-1 was licensed. In September 1992, the NRC Commissioners explicitly reiterated that the General Design Criteria do not apply to plants such as NMP-1 that received

construction permits prior to May 21, 1971.[5] As noted by the Commissioners:

> [E]ach plant licensed before the GDC were formally adopted was evaluated on a plant specific basis, determined to be safe, and licensed by the Commission. Furthermore, current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDCs. Backfitting the GDC would provide little or no safety benefit . . . Plants with construction permits issued prior to May 21, 1971, do not need exemptions from the GDC.

Id. The primary issue under consideration at the time of these statements was whether the GDC should be applied to plants with operating licenses issued after May 21, 1971, or should be limited to those plants with construction permits issued after May 21, 1971. The Commission adopted the narrower interpretation. In any event, NMP-1 received its construction permit on April 12, 1965 (Construction Permit No. CPPR-16) and its initial operating license on August 22, 1969 (Provisional Operating License No. DPR-17). Thus, the Commission's choice of interpretations had no impact on the well-established fact that NMP-1 is not subject to the GDC. The Commissioners' decision stands in sharp contrast to petitioner's statement that NMP-1 noncompliance "has been acknowledged by the NRC Staff and is demonstrated unequivocally by the evidence in the public record."

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[5] See Memorandum for James M. Taylor, NRC Executive Director for Operations, from Samuel J. Chilk, Secretary to the Commission, "SECY-92-223 -- Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992.

VI. RESPONSE TO SPECIFIC CONCERNS

Petitioner expressed concerns regarding primary containment isolation valve administrative deficiencies. In Attachment 1 to this response, Niagara Mohawk has responds to each of the notes provided in Attachment 5 to the Petition. As noted by these responses, Nine Mile Point Unit 1 is in compliance with 10 C.F.R. Part 50 Appendix J. As previously noted exceptions to the GDC have been taken for NMP-1; therefore, no specific reasons to Petitioner's references to certain GDC is contained in Attachment 1. In addition, all of the valves designated as containment isolation boundaries are tested in accordance with the IST and Technical Specification Surveillance Requirement Programs. These programs demonstrate that the valves are capable of physically isolating any containment boundary during a post-accident condition. It was Niagara Mohawk's original design basis that systems such as core and containment spray be recognized as closed systems and as such constituted extensions of containment. The FSAR tables were written prior to issuance of 10 C.F.R. 50 Appendix J and reflect the original design basis of NMP-1. The necessary changes to reflect compliance with Appendix J will be incorporated when the NRC approves the February 7, 1992 Technical Specification application. Over the past several years, Niagara Mohawk and the NRC have attempted to resolve differences regarding implementation of the Appendix J program. As noted in Attachment 1, the administrative deficiencies have either been corrected or are

scheduled to be corrected shortly. In the meantime, these administrative deficiencies do not prevent the valves from performing their intended safety function or detract from the conclusion that NMP-1 can be safety operated in a manner which ensures adequate protection of the public health and safety.

As previously noted, HPCI is not credited in determining conformance with 10 C.F.R. § 50.46. In addition, since the HPCI is not required to shutdown the reactor and maintain it in a safe shutdown condition or prevent or mitigate the consequences of an accident, it is not generally included in the Nine Mile Point Unit 1 IST program. Containment isolation valves (31-01R, 02R, 07 and 08) which are part of the HPCI system are included in the IST program.

As requested by the NRC Project Manager, Attachment 2 to this response provides a more legible copy of the component listings and notes provided in Attachment 5 of the Petition. Note 17 was missing from the list provided to Niagara Mohawk.

VII. CONCLUSION

Petitioner has wholly failed to meet the standards for the institution of a proceeding to consider his allegations, let alone to require an immediate shutdown of the unit. The Petition fails to demonstrate that a substantial health and safety issue exists. In fact, Petitioner has not demonstrated the existence of any issue whatsoever aside from some administrative discrepancies which have been or are in the process of being resolved. The ECCS

for NMP-1 meets all regulatory requirements and assures that the health and safety of the public is protected. While the HPCI mode of the feedwater system is not relied on to meet the acceptance criteria for an ECCS, it has the capability to augment the ECCS in a reliable manner. Petitioner's request that the NRC issue a shutdown order under 10 C.F.R. § 2.206 and hold a hearing before restart should be denied. Furthermore, there is no basis for the allegation that Niagara Mohawk took no action after being notified by Petitioner of alleged problems in January 1990. For the foregoing reasons, the relief requested by Petitioner should be denied.

NOTE 1: FSAR Section VII requires these valves to go open within 20 sec Hi Drywell or low-low Rx level RPS signal and this times fails to appear in either TS Table 3.3.4 or FSAR Table VI-3a. Also, these valves are 10CFR50 Appendix A Criterion 55 valves and are not being tested accordingly.

RESPONSE: Core spray valves 40-01, 09, 10, 11 are located inside primary containment. It was Niagara Mohawk's position in the late 1970's and early 1980's that Core Spray was a closed system which constituted an extension of containment. Therefore, these valves were not included in the Tech. Spec. and FSAR Tables. The NRC Safety Evaluation Report (SER) of May 6, 1988 concluded that these valves meet Appendix J criteria for a water seal. Niagara Mohawk's application of November 20, 1990, superseded on February 7, 1992, revises Tech. Spec. Table 3.3.4 to include these valves. Valves 40-02 and 12 are outside primary containment valves that are locked open and do not require Appendix J testing. These valves are also included in our February 7, 1992 submittal. The correct closing time for all of the above core spray valves is 22.5 seconds which appears in the FSAR Section VII and the proposed Technical Specifications application. When this Tech. Spec. application is approved, the FSAR table will be updated to include the valve and its closure time.

NOTE 2: Containment Spray Test line currently does not receive RPS signal to go closed. The effectiveness of one containment spray pump is lost until operator response manually closes valve should the accident occur during testing of containment spray pumps. Also, this is a criterion 56 valve and is not being tested accordingly and should appear in TS 3.2.7 and FSAR Table VI-3b.

RESPONSE: Niagara Mohawk Safety Evaluation No. 89-13 provided a discussion with respect to loss of containment spray system flow. It concluded that sufficient flow is available even if valve 80-118 failed open. Valve 80-118 is a flow control valve which is inboard of valves 80-114 and 115. Valves 80-114 and 115 are Appendix J isolation valves and meet the criteria for a water seal specified in the NRC SER of May 6, 1988.

NOTE 3: FSAR Table VI-3b shows these valves receive no RPS signal. TS Table 3.3.4 shows these valves receive signal to open. P&ID C18012C shows RPS logic to these valves. Also, these are criterion 56 valves and are not being tested accordingly.

RESPONSE: The containment spray system was considered a closed system which constituted an extension of containment. The NRC SER of May 6, 1988 concluded that these valves meet Appendix J criteria for a water seal. Containment spray valves 80-15, 16, 35 and 36 are remote manual valves and do not receive RPS signal. This change is reflected in our February 7, 1992

license amendment request. The current P&ID C-18012-C does not show RPS logic to these valves.

NOTE 4: FSAR Table VI-3a shows a close stroke time of 18 seconds while TS Table 3.2.7 shows 10 second closure. Even though this is more conservative, the discrepancy came about as an error because components are not individually listed in tables.

RESPONSE: The stroke time for Scram Discharge Volume valves 44.2-15 and 16 was corrected in the FSAR in the June 1990 update. The change was a result of Safety Evaluation No. 89-033 which was approved on December 18, 1989. The FSAR and Tech. Spec. now agree.

NOTE 5: FSAR Table 3a shows RPS logic to close with core spray actuation while TS Table 3.2.7 does not.

RESPONSE: Core spray system valves 40-30, 31, 32, and 33 close on low-low water level or high drywell pressure, which is the core spray actuation signal. Both the FSAR and the Tech. Spec. are correct; however, the FSAR will be changed to eliminate the core spray actuation signal since it is redundant.

NOTE 6: FSAR Table 3b shows these valves with a 70 second and 90 second stroke time. These valves should also appear on TS Table 3.3.4.

RESPONSE: The stroke time for valves 80-114, 115 is 60 seconds. These valves were considered an extension of containment. The NRC SER dated May 6, 1988 concluded that these valves meet Appendix J criteria for a water seal. The FSAR Table will be updated to include the correct stroke time and eliminate the redundant line shown in the FSAR table upon approval of the license amendment.

NOTE 7: P & ID 18014C sht 2 shows these valves receive an RPS signal however, FSAR Table VI-3b and TS Table 3.3.4 fail to include these penetrations and stroke times.

RESPONSE: Valves 201.7-08, 09 are in the Appendix J program. These valves are included in our February 7, 1992 submittal to reflect their inclusion in Appendix J. The FSAR will be updated when the license amendment is approved.

NOTE 8: These valve are criterion 56 valves which appear in FSAR Table VI-3b. These valves may or may not (see note 12) appear in TS Table 3.3.4. TS as written, it is impossible to distinguish however these valves are identified in surveillance test (N1-ST-Q5) as TS acceptance criteria.

RESPONSE: Valves 201.2-23, 24, 25, 26, 27, 28, 29, 30 were associated with the original gas analyzer system. These valves are currently listed in Table 3.3.4 of the Technical Specifications. The valves under note 11 were added during the process of obtaining a full term operating license. These valves were included in a license amendment request dated March 21, 1978 and are included in our February 7, 1992 submittal.

NOTE 9: FSAR Table VI-3a shows RPS logic to close however TS Table 3.2.7 does not identify these valves. Also, valves (*) appear on P & ID C18006C with no RPS logic while they are identified with RPS logic on P & ID C18017C.

RESPONSE: Valves 05-03, 02, 39-11, 12, 13, 14, 110-127, 128 were considered an extension of containment. No testing is required for System 05 and 39 valves listed above. In as much as they are not containment isolation valves since they are outside of the containment boundary which is at valves 01-03, 39-07R, and 39-08R. The 110 system valves above are included in the revised Appendix J submittal of Feb. 7, 1992. A Document Change Request will be issued to revise P&ID C-18006-C to identify the RPS logic shown on P&ID C-18017-C.

NOTE 10: These valves are deactivated and the TS and appropriate FSAR sections should be revised to reflect this change.

RESPONSE: Valve 34-01 does not require testing as the line has been blank flanged. The Tech Specs are being corrected in our Feb. 7, 1992 Appendix J submittal. Valve 58.1-01 is beyond the containment boundary which is at spectacle flange 58.1-07 and therefore does not require testing. The FSAR will be updated when the license amendment is approved.

NOTE 11: These valves are identified on P & ID C18014C sht 1 as receiving RPS logic yet do not appear in FSAR Table VI-3b or TS Table 3.3.4.

RESPONSE: These valves are the isolation valves for the redundant CAD system. See response to note 8. They were included in a March 21, 1978 licensee amendment application and are currently in our Appendix J submittal dated Feb. 7, 1992. The FSAR table will be updated when the amendment is approved.

NOTE 12: FSAR Table VI-3b show RPS logic to close however TS table 3.3.4 does not identify these valves. Effects surveillance program and procedure revision.

RESPONSE: Valves 202-07, 08, 35 and 36 receive a Reactor Building Protection System signal and not a Reactor Protection Signal. They are not primary containment isolation valves and therefore Appendix J is not applicable. The May 6, 1988

NRC SER concluded that valves 63-04 and 05 meet Appendix J criteria for a water seal.

NOTE 13: P & ID C18005C sht 1 show HPCI logic to close yet are not identified in TS or FSAR. Also, not identified in IST Program.

RESPONSE: These valves are in the HPCI system and are not containment isolation valves since they are outside of the containment boundary which is at valves 31-01R and 31-02R. They are therefore not required to be reflected in the FSAR or Tech. Spec. tables. HPCI valves are non safety-related and are generally not part of the IST program.

NOTE 14: FSAR Table VI-3b show RPS logic to close however TS Table 3.3.4 does not identify these valves. Also, tested IAW NI-ST-Q5, current procedure 5 sec TS acceptance criteria that does not exist. Also, these valves do not appear on drawings C18014C as identified in IST plan.

RESPONSE: Valves TIP-1, 2, 3 and 4 are being added to the Technical Specification as a result of an NRC inspection. They are included in our February 7, 1992 submittal. Niagara Mohawk is aware that these valves are not on P&ID C-18014-C, sheet 2. In fact, the Nine Mile Point Unit 1 Pump and Valve Inservice Testing Program Plan lists these valves and notes that the valves are not shown on the P&ID.

NOTE 15: Currently tested IAW N1-ST-Q7 with IST acceptance criteria of 60 sec. No FSAR or TS Stroke times identified.

RESPONSE: The correct valve numbers are 70-92, 93, 94, 95. The February 7, 1992 Technical Specification application reflects these valves and stroke times. The FSAR will be updated appropriately.

NOTE 16: FSAR VI-3c identifies these valves as criterion 57 valves. TS Table 3.3.4 identifies these valves as both criterion 56 and 57 valves. This is physically impossible. Secondly, these valves are not tested to either criterion.

RESPONSE: The correct valve numbers are 70-92, 93, 94, 95. The February 7, 1992 Technical Specification application reflects these valves and stroke times. The FSAR will be updated appropriately.

NOTE 18: FSAR Table VI-3b and TS Table 3.3.4 identify these valves as criterion 56 valves however are not being test accordingly.

RESPONSE: The NRC SER of May 6, 1988 concluded that valves 81-01, 02, 21, 22 meet Appendix J criteria for a water seal.

COMPONENT ID #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	UNIDENTIFIED IN TS	TS SIGNAL	TS ST	PRINTS IN PRS LIST	02743.3499
05-03	Y	CLOSE	5	N	—	—	*	9
05-02	Y	CLOSE	5	N	—	—	*	9
39-12	Y	CLOSE	5	N	—	—	Y	9
39-11	Y	CLOSE	5	N	—	—	Y	9
39-13	Y	CLOSE	5	N	—	—	Y	9
39-14	Y	CLOSE	5	N	—	—	Y	9
40-12	Y	OPEN	—	N	—	—	Y	1
40-01	Y	OPEN	—	N	—	—	Y	1
40-09	Y	OPEN	—	N	—	—	Y	1
40-11	Y	OPEN	—	N	—	—	Y	1
40-10	Y	OPEN	—	N	—	—	Y	1
40-02	Y	OPEN	—	N	—	—	Y	1
58.1-01	N	—	—	Y	—	60	Y	10
80-118	N	—	—	N	—	—	—	2
30-31	N	—	—	N	—	—	HPCI	13
30-32	N	—	—	N	—	—	HPCI	13
201.1-14	N	—	—	N	—	—	Y	11
201.1-16	N	—	—	N	—	—	Y	11
201.1-09	N	—	—	N	—	—	Y	11
201.1-11	N	—	—	N	—	—	Y	11
63-04	Y	CLOSE	30	N	—	—	Y	12
63-05	Y	CLOSE	30	N	—	—	Y	12
80-15	Y	—	60	Y	OPEN	60	Y	3/1
80-16	Y	—	60	Y	OPEN	60	Y	3/1
80-35	Y	—	60	Y	OPEN	60	Y	3/1
80-36	Y	—	60	Y	OPEN	60	Y	3/1

* - see note 9

COMPONENT ID #	IDENTIFIED IN PBAR	PBAR SIGNAL	PBAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	PBID OPMTS ID PS LOGIC	NO. JES
201.2-109	Y	CLOSE	60	N	—	—	Y	11
201.2-112	Y	CLOSE	60	N	—	—	Y	11
44.2-15	Y	CLOSE	18	Y	CLOSE	10	Y	4
44.2-16	Y	CLOSE	18	Y	CLOSE	10	Y	4
40-30	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-31	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-32	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-33	Y	CLOSE	30	Y	CLOSE	50	Y	5
122-03	Y	CLOSE	30	N	—	—	Y	12
110-127	Y	CLOSE	20	N	—	—	Y	9
110-128	Y	CLOSE	20	N	—	—	Y	9
202-07	Y	CLOSE	60	N	—	—	Y	12
202-08	Y	CLOSE	60	N	—	—	Y	12
202-35	Y	CLOSE	60	N	—	—	Y	12
202-36	Y	CLOSE	60	N	—	—	Y	12
80-114	Y	CLOSE	90/70	N	—	—	Y	6
80-115	Y	CLOSE	90/70	N	—	—	Y	6
201.7-08	N	—	—	—	—	—	Y	7
201.7-09	N	—	—	—	—	—	Y	7
201.2-25	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-26	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-27	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-28	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-29	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-30	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-23	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-24	Y	CLOSE	60	Y	CLOSE	60	Y	8

(39)

COMPONENT ID #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	PRID PRIOR ID RPS LOGIC	NOTES
201.2-01	Y	CLOSE	60	N	—	—	Y	11
201.2-02	Y	CLOSE	60	N	—	—	Y	11
201.2-03	Y	CLOSE	60	N	—	—	Y	11
201.2-04	Y	CLOSE	60	N	—	—	Y	11
201.2-10	Y	CLOSE	60	N	—	—	Y	11
201.2-11	Y	CLOSE	60	N	—	—	Y	11
201.2-110	Y	CLOSE	60	N	—	—	Y	11
201.2-111	Y	CLOSE	60	N	—	—	Y	11
34-01	Y	—	30	Y	—	30	—	10
29-51	N	—	—	N	—	—	HPCI	13
29-51	N	—	—	N	—	—	HPCI	13
TIP-1	Y	CLOSE	60	N	—	—	**	14
TIP-2	Y	CLOSE	60	N	—	—	**	14
TIP-3	Y	CLOSE	60	N	—	—	**	14
TIP-4	Y	CLOSE	60	N	—	—	**	14
70-94	Y	—	30	Y	—	30	N	15/16
70-92	Y	—	30	Y	—	30	N	16
70-93	Y	—	—	Y	—	—	N	16
70-99	Y	—	—	Y	—	—	N	16
80-17	Y	—	—	Y	—	—	N	17
80-18	Y	—	—	Y	—	—	N	17
80-37	Y	—	—	Y	—	—	N	17
80-38	Y	—	—	Y	—	—	N	17
80-65	Y	—	—	Y	—	—	N	17
80-66	Y	—	—	Y	—	—	N	17
80-67	Y	—	—	Y	—	—	N	17
80-68	Y	—	—	Y	—	—	N	17

(4)

COMPONENT ID #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	PB-FD PRINTS ID RPS LOGIC	NOTES
80-01	Y	–	70	Y	–	70	N	17
80-02	Y	–	70	Y	–	70	N	17
80-21	Y	–	70	Y	–	70	N	17
80-22	Y	–	70	Y	–	70	N	17
81-01	Y	–	70	Y	–	70	N	18
81-02	Y	–	70	Y	–	70	N	18
81-21	Y	–	70	Y	–	70	N	18
81-22	Y	–	70	Y	–	70	N	18

NOTES:

① FSAR section III requires these valves to go open within 20 sec Hi Drywell or Lo Rx Level RPS signal and this time fails to appear in either TS Table 3.3.4 or FSAR Table VI-3a. Also, these valves are 10CFR50 Appendix A Criterion 55 valves and are not being tested accordingly.

② Containment Spray Test Line currently does not receive RPS signal to go closed. The effectiveness of one containment spray pump is lost until operator response manually closes valve should the accident occur during testing of containment spray pumps. Also, this is a criterion 56 valve and is not being tested accordingly and should appear in TS 3.2.7 and FSAR Table VI-3b.

③ FSAR Table VI-3b shows these valves recieve NO RPS signal. TS Table 3.3.4 shows these valves recieve signal to OPEN. P&ID C18012C shows RPS logic to these valves. Also, these are Criterion 56 valves valves and are not being tested accordingly.

④ FSAR Table VI-3a shows a close stroke time of 18 seconds while TS Table 3.2.7 shows 10 second closure. Even though this is more conservative, the discrepancy came about as an error because components are not individually listed in tables.

Notes Cont.

(5) FSAR Table 3a shows RPS logic to close with core spray actuation while TS Table 3.2.7 does not.

(6) FSAR Table 3b shows these valves with a 70 second and 90 second stroke time. These values should also appear on TS Table 3.3.4.

(7) P&ID 180146 sht 2 shows these valves receive an RPS signal however FSAR Table VI-3b and TS Table 3.3.4 fail to include these parameters and stroke times.

(8) These value are Criterion 5c valves which appear in FSAR Table VI-3b. These valves may or may not (see note 12) appear in TS Table 3.3.4. TS as written, it is impossible to distinguish however these values are identified in surveillance test (N1-ST-Q5) as TS acceptance criteria.

(9) FSAR Table VI-3a shows RPS logic to close. however TS Table 3.2.7 does not identify these values. Also, values (*) appear on P&ID C18006C with no RPS logic while they are identified with RPS logic on P&ID C18017C.

(10) These values are deactivated and the TS and appropriate FSAR sections should be revised to reflect this change.

NOTES CONT.

(11) These valves are identified on P&ID C18014C sht 1 as receiving RPS logic yet do not appear in FSAR Table VI-3b or TS Table 3.3.4.

(12) FSAR Table VI-3b show RPS logic to close however TS Table 3.3.4 does not identify these valves. Affects Surveillance Program and procedure revision.

(13) P&ID C18005C Sht 1 show HPCI logic to close yet are not identied in TS or FSAR. Also not idudified in IST Program.

(14) FSAR Table VI-3b show RPS logic to close however TS Table 3.3.4 does not identify these valves. Also, tested IAW N1-ST-Q5, current procedure 5 sec TS acceptance criteria that does not exist. Also these valves do not appear on drawings C18014C as identified in IST Plan.

(15) Currently tested IAW N1-ST-Q7 with IST acceptance criteria of 60 sec. No FSAR or TS stroke times identified.

(16) FSAR VI-3c identifies these valves as criterion 57 valves. TS Table 3.3.4 identifies these valves as both criterion 56 and 57 valves. This is physically impossible. Secondly, these valves are not tested to either criterion.

5055.3505 (7) 020

NOTES CONT

(18) FSAR Table VI-3b and TS Table 3.3.4 identify these values as criterion 56 values, however ~~[illegible]~~ being tested accordingly.

(8)

NMP1 2.206 Acknowledgement Letter
Date December 4, 1992

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

December 4, 1992

Docket No. 50-220

Ben L. Ridings
P.O. Box 1101
Kingston, Tennessee 37763

Dear Mr. Ridings:

On October 27, 1992, you filed a "Petition for Emergency Enforcement Action and Request for Public Hearing" (Petition) regarding Nine Mile Point Nuclear Station Unit No. 1 (NMP-1) with the Nuclear Regulatory Commission (NRC or Commission). You requested that the Commission take direct review of the Petition. The Commission has declined to take direct review of your Petition and has referred the Petition to me for consideration pursuant to 10 CFR 2.206.

The Petition requests that the NRC immediately order Niagara Mohawk Power Corporation (NMPC) to cease power operation of NMP-1 and place the reactor in a cold shutdown condition. The Petition also requests that the Commission hold a public hearing before authorizing resumption of plant operation. You seek relief based on allegations that: (1) NMP-1 does not meet NRC requirements for an engineered safety feature system (ESFS) grade high-pressure coolant injection (HPCI) system, (2) 45 percent of the containment isolation valves have administrative deficiencies, and (3) NMPC, NMPC's quality assurance group, and the NRC have reviewed these safety concerns and, contrary to any practical justification, have remained silent.

With respect to the lack of an ESFS grade HPCI system, you had two concerns: (1) you stated that the feedwater system, which can operate in an HPCI mode, is not an acceptable alternative system because it does not have a backup electrical power supply provided by an onsite emergency diesel generator and (2) you stated concern about using the feedwater system in an HPCI mode because some 44 out of 47 valves in the feedwater injection flow path are not included in the NMP-1 Inservice Test Program for pumps and valves.

Although NMP-1 does not have an ESFS grade HPCI system, the plant was designed and licensed by the NRC with other emergency core cooling system (ECCS) equipment that provides adequate protection against all loss-of-coolant accidents. The Commission's regulations in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors," require that licensees provide their plants with ECCS's designed to meet the criteria set forth in that section.

In response to your request for an immediate shutdown of NMP-1, my staff has reviewed the NRC's Safety Evaluation Report for NMP-1 dated December 27, 1974, and General Electric's (GE) loss-of-coolant accident analysis (NEDC-31446P) for the current fuel cycle (Cycle 10) concerning NMP-1's conformance to the requirements of 10 CFR 50.46.

The December 27, 1974, Safety Evaluation Report concluded that the NMP-1 ECCS satisfies the requirements of 10 CFR 50.46. Furthermore, the Updated Final Safety Analysis Report (UFSAR) states that the HPCI is not an engineered safety feature system and, therefore, is not relied on in meeting the criteria of 10 CFR 50.46. This conclusion was reaffirmed in GE's loss-of-coolant accident analysis for the current fuel cycle as well as in the previous reload cycles. GE's analysis was prepared in response to the requirements of NMP-1 Technical Specification 6.9.1f, "Reporting Requirements, Core Operating Limits Report." The NMP-1 ECCS satisfies the requirements of 10 CFR 50.46 by utilizing the automatic depressurization system (ADS) and the core spray system (CSS), both of which have redundancy and are supplied backup electrical power by the NMP-1 onsite emergency diesel generators. The CSS in conjunction with the ADS is designed to accommodate the range of loss-of-coolant accidents from the smallest up to the largest line break. For line breaks smaller than 0.30 square foot, reactor pressure may not decrease rapidly enough to prevent clad overheating if there is no feedwater flow. Therefore, the ADS is provided to depressurize the reactor so that the CSS can inject water into the reactor. Because operation of the feedwater pumps in the HPCI mode is not required to meet the requirements of 10 CFR 50.46, an onsite emergency electrical power supply for the feedwater pumps is not required. The NMP-1 Technical Specifications require the feedwater system to be operable in the HPCI mode as the normal means for core cooling; however, this system is not relied on to satisfy the requirements of 10 CFR 50.46. Furthermore, the valves in the feedwater flow path are not required to be included in the NMP-1 inservice testing program because the feedwater system is not required to meet 10 CFR 50.46.

You also asserted that the NMP-1 feedwater system operating in the HPCI mode fails to meet GDC 33, 35, 36 and 37. As stated in a Staff Requirements Memorandum dated September 18, 1992, the Commission has determined that the General Design Criteria in 10 CFR Part 50, Appendix A, do not apply to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. While compliance with the intent of the GDC is important, each plant licensed before the GDC were formally adopted was evaluated on a plant specific basis, determined to be safe, and licensed by the Commission. Furthermore, current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDC. Plants with construction permits issued prior to May 21, 1971, do not need exemptions from the GDC.

On the basis of the foregoing discussions, I have concluded that there is no basis to issue an immediately effective order to shut down NMP-1 because of the unavailability of an ESFS grade HPCI system.

Your Petition also stated that 45 percent of the primary containment isolation valves at NMP-1 had administrative deficiencies as indicated in Attachment 5 to your Petition. However, some of the valve identification numbers listed in Attachment 5 are not fully legible, and for these valves, we were unable to evaluate your concerns. Note 17 applicable to valves listed on pages 1, 3, and 4 of Attachment 5 was not provided; for these valves, my staff reviewed the existing regulatory requirements and NMPC's procedures and programs for implementing those requirements and found no deficiencies. The NRC staff had previously identified, through its inspection program, administrative deficiencies, similar to those identified in Attachment 5, with reactor coolant system isolation valves and containment isolation valves listed in the NMP-1 Technical Specifications and the UFSAR. In a safety evaluation dated May 6, 1988, the NRC staff requested NMPC to resolve these administrative deficiencies. Subsequently, by letter dated November 20, 1990, as superseded by letter dated February 7, 1992, NMPC submitted a request for a license amendment to update the NMP-1 Technical Specifications to resolve these administrative deficiencies.

Our review of this request is in progress and although we have not yet completed our review, we have reviewed your concerns with respect to the current NMP-1 Technical Specifications, the UFSAR, and the most recent Inservice Testing Program for NMP-1 pumps and valves. Our preliminary review indicated that NMPC is implementing adequate surveillance testing and leakage-rate testing procedures to verify valve and containment operability. These procedures include functional testing required by the NMP-1 Technical Specifications to ensure that valves required to close during accident conditions function properly on receipt of a signal to close. Furthermore, periodic valve exercising, stroke-time testing, and leakage-rate testing ensure that the inservice testing program and applicable technical specification requirements are met. All the above testing provides reasonable assurance that NMP-1 can be operated without undue risk to the public health and safety in light of the described administrative deficiencies in isolation valves. In addition, our preliminary conclusions are that the current technical specifications, the license amendment request previously discussed, or the UFSAR address most of these administrative deficiencies. Based on the above, I have concluded that an immediately effective order to shut down NMP-1 on the basis of the identified administrative deficiencies with the containment isolation valves is not required.

As stated above, our review of the Petition has disclosed that some specific information in your Petition was not fully legible or not provided. The NRC staff has been unable to contact you by telephone to obtain the missing information. In order for the NRC to provide a complete review of your

concerns, we request that you provide the missing information promptly but, in any event, within 25 days of the date of this letter in order for us to consider it in our evaluation of your Petition. You may provide the missing information by contacting Mr. Donald S. Brinkman, the NRC's Project Manager for NMP-1 at (301) 504-1409.

With regard to your allegation that the NRC staff has previously reviewed these safety concerns and has remained silent, a copy of the Petition has been referred to the NRC Office of the Inspector General for whatever review and action the Inspector General deems appropriate.

The NRC staff will review your Petition in accordance with 10 CFR 2.206. I will issue a final decision with regard to your Petition within a reasonable time. A copy of the notice that is being filed for publication with the Office of the Federal Register is enclosed for your information.

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
Federal Register Notice

cc w/enclosure:
See next page

Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit No. 1

cc:

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Syracuse, New York 13212

UNITED STATES NUCLEAR REGULATORY COMMISSION

Docket No. 50-220

NIAGARA MOHAWK POWER CORPORATION

(Nine Mile Point Nuclear Station Unit No. 1)

RECEIPT OF PETITION FOR DIRECTOR'S DECISION UNDER 10 CFR 2.206

Notice is hereby given that by letter dated October 27, 1992, Ben L. Ridings (Petitioner) filed a "Petition for Emergency Enforcement Action and Request for Public Hearing" (Petition) regarding Nine Mile Point Nuclear Station Unit No. 1 (NMP-1) with the Nuclear Regulatory Commission. The Petition, which has been referred to me for consideration as a petition under 10 CFR 2.206, requests that the Nuclear Regulatory Commission immediately order Niagara Mohawk Power Corporation (NMPC) to cease power operation of NMP-1 and place the reactor in a cold shutdown condition. The Petition seeks relief on the basis of allegations that: (1) NMP-1 does not meet NRC requirements for an engineered safety feature system (ESFS) grade high-pressure coolant injection (HPCI) system, (2) 45 percent of the containment isolation valves have administrative deficiencies, and (3) NMPC, NMPC's quality assurance group, and the NRC have reviewed these safety concerns, and contrary to any practical justification, have remained silent.

With respect to the lack of an ESFS grade HPCI system, the Petitioner had two concerns: First, the Petitioner stated that the feedwater system, which can operate in an HPCI mode, is not an acceptable alternative system because it does not have a backup electrical power supply provided by an onsite emergency diesel generator; second, the Petitioner was concerned about using the feedwater system in a HPCI mode because some 44 out of 47 valves in the feedwater injection flow path are not included in the NMP-1 Inservice Test Program for pumps and valves.

For the reasons stated in a letter to the Petitioner dated December 4, 1992, Petitioner's request for immediate action was denied. Petitioner's request is being treated in accordance with 10 CFR 2.206 of the Commission's regulations. The NRC will take appropriate action on this request within a reasonable time.

A copy of the Petition is available for inspection and copying for a fee in the Commission's Public Document Room, 2120 L Street, NW., Washington, DC 20555 and at the Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 4thday of December 1992.

concerns, we request that you provide the missing information promptly but, in any event, within 25 days of the date of this letter in order for us to consider it in our evaluation of your Petition. You may provide the missing information by contacting Mr. Donald S. Brinkman, the NRC's Project Manager for NMP-1 at (301) 504-1409.

With regard to your allegation that the NRC staff has previously reviewed these safety concerns and has remained silent, a copy of the Petition has been referred to the NRC Office of the Inspector General for whatever review and action the Inspector General deems appropriate.

The NRC staff will review your Petition in accordance with 10 CFR 2.206. I will issue a final decision with regard to your Petition within a reasonable time. A copy of the notice that is being filed for publication with the Office of the Federal Register is enclosed for your information.

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,
Original Signed by
Thomas E. Murley
Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
Federal Register Notice

cc w/enclosure:
See next page

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*SEE PREVIOUS CONCURRENCE

PDI-1:LA	PDI-1:PM	PDI-1:D	TECH ED*	Region I*	SRXB*
CVogan	DBrinkman:smm	RACapra		WHehl	RJones
11/30/92	11/30/92	11/30/92	11/25/92	11/25/92	11/25/92
SPLB*	EMEB*	OGC*	ADRI*	D:DRPE	ADPR/NRR
CMcCracken	JNorberg	JGoldberg	JCalvo	SVarga	JPartlow
11/25/92	11/25/92	11/27/92	11/25/92	11/30/92	11/30/92
D:NRR					
TMurley					
12/4/92					

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