UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
Thomas E. Murley, Director

In the Matter of	)	
NIAGARA MOHAWK POWER CORPORATION	)	Docket No. 50-220
(Nine Mile Point Nuclear Station Unit No. 1)	)	(10 CFR 2.206)

DIRECTOR'S DECISION UNDER 10 CFR 2.206

INTRODUCTION

On October 27, 1992, Mr. Ben L. Ridings (Petitioner) filed a Petition for consideration in accordance with 10 CFR 2.206 with the Nuclear Regulatory Commission (NRC or Commission). The Petitioner requested that the Commission take direct review of the Petition. However, the Commission declined to take direct review and referred the Petition to the Director, Office of Nuclear Reactor Regulation (NRR), for consideration.

The Petitioner requested that the NRC issue an immediately effective order directing Niagara Mohawk Power Corporation (NMPC) to cease power operation of Nine Mile Point Nuclear Station Unit No. 1 (NMP-1) and place the reactor in a cold-shutdown condition. The Petition also asked the Commission to hold a public hearing before authorizing resumption of plant operation. As bases for these requests, the Petitioner asserted that (1) NMPC is operating NMP-1 in violation of the requirements for availability of an emergency core cooling system (ECCS) high-pressure coolant injection (HPCI) system including the failure to provide the mandatory emergency backup power to the 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.

The Petition was placed in the Public Document Room and a copy of the Petition was sent to NMPC in a letter of November 19, 1992, for NMPC's review and comments regarding the issues raised in the Petition. In a letter of December 21, 1992, NMPC commented on the issues raised in the Petition.

In a letter of December 4, 1992, I acknowledged receipt of the Petition, informed the Petitioner that the Commission had declined to take direct review of the Petition, denied Petitioner's request for immediate action, and told the Petitioner that a final decision on the Petition would be issued within a reasonable time. My December 4, 1992, letter to the Petitioner also requested that the Petitioner give the NRC some specific information that was not fully legible or not provided in the Petition.

In response to my request for specific information, the Petitioner submitted a document titled "Information Requested by Office of Nuclear Reactor Regulation" as an attachment to a letter received by the NRC Office of the Executive Director for Operations on January 5, 1993. In his response, the Petitioner also asserted that the NMP-1 facility will not meet the leakage limits of 10 CFR Part 50, Appendix J, when the leakage rates of Category A containment isolation valves are added to the leakage total for the NMP-1 containment building. (As defined by ASME Code Section XI, Category A valves are those for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their function.) In addition, the

Petitioner contends that NMPC's asserted failures to comply with the requirements of 10 CFR Part 50 precludes NMPC from operating NMP-1 with limited liability. A copy of the Petitioner's response was sent to NMPC in a letter of January 11, 1993, for NMPC review and comments regarding the issues raised in the response. In a letter of February 9, 1993, NMPC commented on the issues raised in the Petitioner's response. A copy of the Petitioner's response was also placed in the Public Document Room.

I have now completed my evaluation of the Petition and the Petitioner's response ("Information Requested by Office of Nuclear Reactor Regulation"). The Petitioner's request for correction of the NMP-1 Technical Specification (TS) to correctly list the NMP-1 containment isolation valves, their initiating signals, and their stroke times is granted. However, for the reasons given in the discussion below, the Petitioner's request for other actions is denied.

DISCUSSION

The NRC staff's evaluation of the Petitioner's assertions follows.

(1) NMP-1 does not meet NRC requirements for an ECCS HPCI system.

The Petitioner asserted that NMP-1 does not meet NRC requirements for an ECCS HPCI system for the following reasons:

(a) NMP-1 fails to meet General Design Criterion (GDC) 33, "Reactor coolant makeup"; GDC 35, "Emergency core cooling"; GDC 36,

"Inspection of emergency core cooling system"; and GDC 37, "Testing of emergency core cooling system," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 because NMP-1 does not have an ECCS HPCI system to provide abundant emergency core cooling in the event of a small-break loss-of-coolant accident (LOCA). Petitioner also asserted that the feedwater system operating in its HPCI mode is not an acceptable alternative system because it does not have a backup electric power supply from an onsite emergency diesel generator.

(b) Of the 47 valves in the feedwater injection flow path, 44 are not included in the NMP-1 inservice testing program for pumps and valves.

The NRC staff's review of these issues and conclusions are based on the original design and licensing basis of NMP-1, as follows.

NMP-1 is a General Electric boiling-water reactor with a Mark I containment. After appropriate review and evaluation by the staff of the U.S. Atomic Energy Commission (AEC), predecessor regulatory agency to the NRC, the NMP-1 Construction Permit was issued to NMPC on April 21, 1965.

On March 24, 1969, the AEC staff issued a report to the Advisory Committee on Reactor Safeguards in which the AEC staff 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.

The NMP-1 Provisional Operating License was issued to NMPC on August 22, 1969. The "Technical Supplement to Petition for Conversion From Provisional Operating License to Full-Term Operating License," dated July 1972, gave information related to the extent to which NMP-1 conforms to the GDC. The NRC did not require NMPC to design NMP-1 in accordance with the GDC because NMP-1 was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission.

The NRC staff also notes that NMP-1 received a construction permit on April 21, 1965, a date that preceded the issuance of the GDCs in Appendix A to 10 CFR Part 50. (The GDCs were issued on May 21, 1971.) In a September 18, 1992, staff requirements memorandum (SRM) to the NRC Executive Director for Operations, the Commission set forth its position that the NRC staff will not apply the GDCs to plants with construction permits issued before May 21, 1971. The SRM continued:

> 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. Backfitting the GDC would provide little or no safety benefit while requiring an extensive commitment of resources. Plants with construction permits issued prior to May 21, 1971, do not need exemptions from the GDC.

Therefore, GDC 33, 35, 36, and 37 do not apply to NMP-1.

The AEC published its acceptance criteria for emergency core cooling systems for light-water power reactors on January 4, 1974 (39 FR 1003). This then-new regulation added Appendix K to 10 CFR Part 50 which specifies analytical techniques to be employed for the evaluation of ECCS acceptability. NMP-1 was originally licensed to the Interim Acceptance Criteria of 10 CFR 50.46 which were effective while the AEC was promulgating this regulation. The AEC Safety Evaluation Report of December 27, 1974, concluded that the NMP-1 ECCS satisfies the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50 as finally promulgated. That conclusion was reached without relying on or taking credit for the feedwater system operating in its HPCI mode. Moreover, NMP-1 meets the intent of the GDC by providing redundant methods for reliably cooling the reactor core (and meeting the requirements of 10 CFR 50.46) under postulated accident conditions. The provisional operating license was converted to a full-term operating license on December 26, 1974.

The following is a summary of the NRC staff's analysis of how the NMP-1 ECCS satisfies NRC requirements. The NMP-1 ECCS includes the core spray system (CSS), consisting of two separate and independent loops, and an automatic depressurization system (ADS). The CSS and ADS are described in UFSAR Sections VII.A. and V.B.5.0, respectively. Each CSS loop consists of two 100-percent pump combinations (i.e., two core spray pumps and two core spray topping pumps). The maximum discharge pressure of each pump combination is approximately 350 psig. The four core spray pumps and four core spray topping pumps get electric power from offsite sources or from the onsite emergency diesel generators. The logic for the ADS is powered by ac emergency

power supplies, and the six electrically actuated relief valves get electric power from the dc emergency power supplies (station batteries in parallel with battery chargers).

The CSS is a safety-related system which is designed to accommodate the range of loss-of-coolant accidents from the smallest up to the largest line break. For large breaks, the CSS can maintain the peak cladding temperature within the acceptance criteria of 10 CFR 50.46 without assistance from the ADS because the reactor depressurizes sufficiently fast for the CSS to achieve rated flow before the criteria of 10 CFR 50.46 are exceeded. For small breaks, i.e., breaks below about 0.30 square foot, the ADS is provided and it will operate to depressurize the reactor to permit water injection by the CSS before the criteria of 10 CFR 50.46 are exceeded. The criteria of 10 CFR 50.46 are not exceeded, assuming a single failure that disables one of the two available CSS loops and without taking credit for operation of the feedwater system in the HPCI mode.

In addition to the CSS and ADS, NMP-1 also has and utilizes the feedwater system operating in a HPCI mode and two control rod drive pumps operating in the coolant injection mode to inject water into the reactor at reactor operating pressure in the event of a small-break LOCA. Successful operation of these systems is desirable since their proper operation may negate the need to unnecessarily actuate the ADS valves. However, the NMP-1 LOCA safety analyses do not rely on water injection by either the control rod drive pumps or the feedwater system operating in the HPCI mode to satisfy the requirements of 10 CFR 50.46.

The foregoing conclusion has been reaffirmed in the General Electric Company's (GE's) LOCA analysis for each subsequent fuel cycle. Each analysis was performed to demonstrate compliance with the requirements of 10 CFR 50.46 without taking credit for the feedwater system operating in the HPCI mode. The analysis for the current fuel cycle was prepared in response to the requirements of NMP-1 TS 6.9.1f, "Reporting Requirements, Core Operating Limits Report."

Therefore, the NRC staff concludes that the Petitioner's assertion that the NMP-1 HPCI system (feedwater system operating in the HPCI mode) must meet GDC 33, 35, 36, and 37 and that the HPCI system must be part of the ECCS and be supplied with backup electrical power from an onsite emergency diesel generator is incorrect. NMP-1 does not have and does not need an ECCS HPCI system because the existing NMP-1 ECCS satisfies the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, without reliance on the feedwater system operating in the HPCI mode.

<u>Non-Safety-Related Methods for Coolant Injection</u>

In addition to the CSS and ADS, NMP-1 has two control rod drive pumps which can be operated in the coolant injection mode and with a feedwater system which can be operated in an HPCI mode to inject coolant at reactor operating pressure. Each control rod drive pump is rated at 85 gpm at a head of 3760 feet. Operation of the control rod drive pumps in the coolant injection mode is described in Section X.C. of the UFSAR. Operability of the control rod drive pumps in the coolant injection mode is required by TS 3.1.6,

"Control Rod Drive Pump Coolant Injection." Electric power for the control rod drive pumps comes from either offsite sources or from the onsite emergency diesel generators.

Operation of the feedwater system in the HPCI mode is described in Section VII.I of the UFSAR. Operability of the HPCI system is required by TS 3.1.8, "High Pressure Coolant Injection." The HPCI system utilizes the two condensate storage tanks, the main condenser hotwell, two condensate pumps, condensate demineralizers, two feedwater booster pumps, feedwater heaters, two motor-driven feedwater pumps, an integrated control system, and all associated piping and valves. The HPCI system is capable of delivering 6840 gpm into the reactor vessel at reactor pressure when using two trains of feedwater pumps. The HPCI system gets electric power from normal offsite sources by either of the two 115-kV lines, but not from the onsite emergency diesel generators. NMP-1 also has the capability of automatically realigning the HPCI system to receive electric power from a dedicated generator at the Bennetts Bridge Hydro Station in the event of a loss of power to both 115-kV offsite lines. Although this hydrogenerator is not equivalent to an onsite emergency diesel generator, it is a highly reliable source of backup power.

Operation of the control rod drive pumps in the coolant injection mode and the feedwater system in the HPCI mode is described in the UFSAR. The control rod drive pumps and the HPCI system are required to be operable by the NMP-1 TS. The control rod drive pumps and the HPCI system are required to be operable by the TS to provide coolant injection without unnecessarily actuating the ADS valves. However, the NMP-1 safety analyses do not rely on

operation of the control rod drive pumps in the coolant injection mode or on operation of the feedwater system in the HPCI mode to provide emergency core cooling or to meet the acceptance criteria of 10 CFR 50.46.

NMP-1 was designed and constructed, and began operation (Provisional Operating License issued on August 22, 1969), before May 21, 1971, when the GDCs were issued. The final emergency core cooling acceptance criteria of 10 CFR 50.46 were satisfied by the NMP-1 ECCS (one out of two loops of the CSS operating in conjunction with the ADS) without reliance on either the control rod drive pumps operating in the coolant injection mode or the feedwater system operating in the HPCI mode. Therefore, neither the control rod drive pumps nor the feedwater system operating in the HPCI mode are required to meet 10 CFR 50.46 criteria for ECCS equipment.

Applicability of IST Program to Feedwater System

With regard to the Petitioner's concern regarding the failure to include 44 of the 47 valves in the feedwater injection flow path in the NMP-1 inservice testing (IST) program for pumps and valves, as discussed above, the NMP-1 safety analyses do not rely on feedwater system operation in the HPCI mode to provide emergency core cooling or to satisfy the criteria of 10 CFR 50.46. Furthermore, the feedwater system is not otherwise required to be a safety-related system. For nuclear power facilities whose construction permits were issued before January 1, 1971 (as was the case for NMP-1), Section (f) of 10 CFR 50.55a requires the IST programs for those facilities to include, to the extent practical, IST requirements for pumps and valves classified as ASME Code Class 1, 2, and 3. However, the NMP-1 feedwater system is not a safety-related system and is not classified as an ASME Code

Class 1, 2, or 3 system. Therefore, the Commission's regulations do not require these valves to be part of the NMP-1 IST program. However, the feedwater isolation valves (31-01R, 31-02R, 31-07, and 31-08) function as part of the reactor coolant system pressure boundary and are, therefore, included in the NMP-1 IST program for pumps and valves. These valves are also containment isolation valves and, as such, are included in TS Table 3.2.7.

HPCI System — Conclusion

In summary, the HPCI system is not required to meet GDC 33, 35, 36, and 37, it is not required to be part of the ECCS with backup electric power from an onsite emergency diesel generator, nor is its operation required to satisfy the emergency core cooling requirements of 10 CFR 50.46. The existing ECCS satisfies the emergency core cooling requirements of 10 CFR 50.46 without reliance on the non-safety-related feedwater system operating in the HPCI mode. The Petitioner does not raise any new issues regarding the design or operation of the feedwater system operating in the HPCI mode. Accordingly, I find that the Petition contains no basis to order a shutdown of NMP-1 or to institute such a proceeding as requested by the Petitioner; therefore, this portion of the Petition is denied.

(2) 45 percent of the containment isolation valves have administrative deficiencies. The NMP-1 facility will not meet the leakage limits of 10 CFR Part 50, Appendix J, when the leakage rates of Category A containment isolation valves are added to the leakage total for the NMP-1 containment building.

The Petitioner asserted that 45 percent of the NMP-1 containment isolation valves have administrative deficiencies. Attachment 5 to the

Petition listed 18 notes in which the Petitioner identified specific deficiencies associated with the containment isolation valves. The asserted deficiencies included:

1. failure to list certain containment isolation valves in the TS or UFSAR tables which list the containment isolation valves

2. failure to test the containment isolation valves in accordance with the requirements of 10 CFR Part 50, Appendix J

3. failure to test the containment isolation valves in accordance with the requirements of the NMP-1 IST program

4. inconsistencies in valve stroke time requirements between the TS tables and the UFSAR

5. inconsistencies in valve actuation signals as specified in the TS tables, the UFSAR tables, and on plant drawings

Primary reactor containments are required to meet the containment leakage test requirements given in Appendix J of 10 CFR Part 50. The purpose of containment leakage tests performed in accordance with the requirements of Appendix J are to ensure that (1) leakage through the primary reactor containment and systems and components penetrating primary containment do not exceed allowable leakage rate values as specified in the plant's technical

specifications or associated bases and (2) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. The maximum allowable leakage rate (La) for the NMP-1 primary containment is 1.5 weight percent of the contained air per 24 hours at a test pressure of 35 psig. Section III.C.3 of Appendix J further limits the combined leakage for all penetrations and valves subject to Types B and C tests (as defined in Sections II.G and II.H of Appendix J) to less than 0.60 La. Type C tests are intended to measure containment isolation valve leakage rates.

Containment isolation valves are provided on lines penetrating the drywell and pressure suppression chamber to ensure integrity of the containment when required during emergency and postaccident periods. Containment isolation valves which must be closed to ensure containment integrity immediately after an accident are automatically controlled by the reactor protection system.

The NRC staff has reviewed the deficiencies identified in Attachment 5 to the Petition. Each of the notes listed in Attachment 5 to the Petition and the NRC staff's corresponding specific findings are discussed in Attachment 1 to this Director's Decision. Several of the Petitioner's notes included comments regarding compliance with GDC 55, "Reactor coolant pressure boundary penetrating containment"; GDC 56, "Primary containment isolation"; and GDC 57, "Closed system isolation valves," of Appendix A to 10 CFR Part 50.

These comments are not individually addressed in the NRC staff findings since, as previously noted, the NRC staff has concluded that the GDCs of Appendix A to 10 CFR Part 50 are not applicable to NMP-1.

The NMP-1 containment isolation valves are listed in two tables in the NMP-1 operating license TS and in three tables in the NMP-1 UFSAR. NMP-1 TS Table 3.2.7 and NMP-1 UFSAR Table VI-3a listing containment isolation valves are titled "Reactor Coolant System Isolation Valves." NMP-1 TS Table 3.3.4 listing containment isolation valves is titled "Primary Containment Isolation Valves Lines Entering Free Space of the Containment." NMP-1 UFSAR Table VI-3b listing containment isolation valves is titled "Primary Containment Isolation and Blocking Valves Lines Entering Free Space of the Containment." NMP-1 UFSAR Table VI-3c listing containment isolation valves is titled "Primary Containment Isolation and Blocking Valves Lines with a Closed Loop Inside Containment Vessels."

The NRC staff had previously identified, through its inspection program, administrative deficiencies in the TS and UFSAR listings of the containment isolation valves similar to those identified in the Petition. An evaluation of NMP-1 compliance with the requirements of Appendix J to 10 CFR Part 50 was sent to NMPC in a letter and attached safety evaluation of May 6, 1988. The NRC staff letter and the attached safety evaluation stated that several changes were required to the NMP-1 TS and requested that NMPC submit a license amendment to revise the NMP-1 TS.

In a letter of November 20, 1990, NMPC submitted a proposed license amendment to update the containment isolation valve tables and to bring the TS into conformance with the requirements of 10 CFR Part 50, Appendix J, and the

NRC staff's safety evaluation of May 6, 1988. NRC staff and NMPC representatives discussed the contents of the November 20, 1990, submittal in a meeting held on March 5, 1991. Following this meeting, NMPC representatives requested that the NRC staff suspend review of the November 20, 1990, submittal, since NMPC would be revising and resubmitting the proposed TS based on comments from the March 5, 1991, meeting.

In a letter of February 7, 1992, NMPC submitted a proposed license amendment that superseded the November 20, 1990, submittal and incorporated the comments from the March 5, 1991, meeting between NMPC and NRC staff. The NRC staff reviewed the February 7, 1992, submittal and issued a request for additional information (RAI) to NMPC on November 30, 1992. NMPC responded to this RAI in a letter of January 29, 1993. The NRC staff conducted an onsite inspection of the containment isolation valve issue during the period February 1-5, 1993. The purpose of the onsite inspection was to obtain more information about the containment isolation valve issue. The findings of the onsite inspection are summarized below. The detailed results of that inspection are reported in combined Inspection Report No. 50-220/93-01 and 50-410/93-01, dated March 23, 1993.

Completeness of TS and UFSAR Tables of Containment Isolation Valves

During the February 1-5, 1993, onsite inspection, the NRC staff independently developed a list of containment isolation valves using plant drawings. In order to compare this list with TS Tables 3.2.7 and 3.3.4 of the February 7, 1992, license amendment request, the NRC staff needed to understand the criteria used by NMPC in the development of the tables. NMPC stated that the TS tables were developed to list any containment isolation

valves that received an automatic isolation signal from the reactor protection system (RPS). On the basis of a comparison using this criterion, the NRC staff concluded that the two lists were consistent, with two exceptions. Specifically, the proposed TS tables did not include valves 63-04 and 63-05 (postaccident sampling system return isolation valves) identified on Drawing F-45089-C, Sheet 8, Revision 3, as containment isolation valves.

. Following discussions with the NRC staff, NMPC changed the criterion for listing valves as containment isolation valves in the TS tables. The revised criterion included only those isolation valves closest to the containment. On the basis of this change, the following revisions were made in a February 18, 1993, supplement to the February 7, 1992, submittal:

- Valves 63-04 and 63-05 were not included in the TS table because they do not serve as containment isolation valves.

  The NRC staff verified that while these valves receive automatic isolation signals from the reactor protection system they are located in a branch line outside of containment isolation valves 63.1-01 and 63.1-02 that also receive automatic isolation signals from the RPS. Valves 63.1-01 and 63.1-02 are included in TS Table 3.3.4.

- Valves 05-02 and 05-03R (emergency cooling high point vent to main steam); 39-11R, 39-12R, 39-13R, and 39-14R (emergency cooling steam line drain to main steam); and 05-01R, 05-04R, 05-11, and 05-12 (emergency cooling high point vent line), were deleted from TS Table 3.2.7.

The NRC staff verified that containment isolation valves 39-03, 39-04, 39-05, 39-06, 39-07R, 39-08R, 39-09R, and 39-10R, are located between the subject valves and the reactor coolant system and are included in TS Table 3.2.7.

- Valves 80-114 and 80-115 (containment spray discharge to waste disposal system) were deleted from TS Table 3.3.4.

  The NRC staff verified that isolation valve 80-118 provides the containment isolation function for the subject penetration. NMPC updated TS Table 3.3.4 to include valve 80-118 in a February 18, 1993, supplement to the February 7, 1992, submittal.

On the basis of this review, the NRC staff agreed that the proposed change was appropriate, since the revision states that the valves located closest to the containment are to be considered the containment isolation valves rather than valves in branch lines that are outboard of valves closer to the containment. This revised criterion serves to minimize extensions of the containment and is, thereby, consistent with the intent of GDC 55, 56, and 57, even though the GDCs are not applicable to NMP-1.

The NRC staff also determined that the proposed TS tables did not include six normally closed manual isolation valves. NMPC stated that these valves had not originally been included because they were normally closed, manually operated valves that do not receive an automatic isolation signal from the RPS. However, NMPC committed to include four of these manual valves (72-479,

72-480, 114-114, and 114-116) in TS Table 3.3.4 so that all containment isolation valves will be listed in the TS tables. NMPC included these valves in TS Table 3.3.4 in the February 18, 1993, supplement to the February 7, 1992, submittal. The other two valves (110-165 and 110-166) were not included in the TS tables since this line has been capped and the penetration will be tested as part of Type B penetration testing. Therefore, these two valves are no longer classified as containment isolation valves.

In addition, the NRC staff independently reviewed the technical data in the TS tables. With the exception of three items described below, all entries were independently verified to be correct.

1. On proposed TS page 148, the bracket indicating that the listed initiating signal was indicated as being applicable to all four penetrations (drywell supply, suppression chamber supply, drywell return, and suppression chamber return) of the H_2/O_2 #12 sampling system was incorrectly drawn. The bracket erroneously indicated that the initiating signal was applicable to the self-actuating check valves when, in fact, the initiating signal was applicable only to the dc solenoid valves in the drywell supply and suppression chamber supply lines.

2. On proposed TS page 148, Note (1) was incorrectly applied to four places on the #11 H_2/O_2 sampling entries (drywell supply, suppression chamber supply, drywell return, and suppression chamber return). Note (1) states: "These valves do not have to be vented during the Type "A" test. However, Type C leakage from these valves is added to the Type A test results."

Since these lines are required to be vented during Type A tests, this note should not apply to these valves.

3. On proposed TS page 148a, Note (1) was also incorrectly applied to the containment atmosphere monitoring supply line entry since this line is required to be vented during Type A tests.

These administrative deficiencies were discussed with NMPC and were corrected in the February 18, 1993, supplement to the February 7, 1992, submittal. The NRC staff concluded that all other technical data entries in the TS tables were correct.

The NRC staff verified consistency between the pertinent elementary RPS wiring drawings and the valve isolation actuation signals listed in the February 7, 1992, license amendment request. The NRC staff reviewed a sample of recent test data to determine if these valves responded properly to their actuation signals. Specifically, test results from the most recent performance of Procedure N1-ST-R2, "Loss of Coolant Accident and Emergency Diesel Generator Simulated Automatic Initiation Test" (July 9–11, 1992), were reviewed. This test inserted low-low reactor water level and high drywell pressure signals (the most common actuation signals for containment isolation valves) and verified that the specified isolation valves closed. Review of the test results revealed that all valves listed in the procedure responded properly.

The NRC staff also verified that NMPC had similar test procedures in place for all containment isolation valves and that these procedures were being used to verify proper isolation valve response to other actuation signals.

Containment Leakage Rate Testing

Appendix J of 10 CFR Part 50 establishes the NRC requirements for containment leakage testing. Appendix J requires performance of three types of containment leakage tests (Type A Tests, Type B Tests, and Type C Tests). These three types of tests are explained in Sections II.F, II.G, and II.H. of Appendix J.

Type A Tests are tests intended to measure the primary reactor containment overall integrated leakage rate (1) after the containment has been completed and is ready for operation and (2) at periodic intervals thereafter.

Type B Tests are tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:

1. containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows; and electrical penetrations fitted with flexible metal seal assemblies,

2. air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary,

3. doors with resilient seals or gaskets, except for seal-welded doors, and

4. components other than those listed in 1, 2, or 3 (above) which must meet the acceptance criteria in Section III.B.3. of Appendix J (combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 La).

Type C Tests are tests intended to measure containment isolation valve leakage rates.

<u>Leakage Rate Testing of Containment Isolation Valves</u>

The NRC staff reviewed the most recent local leakage rate test (LLRT) results associated with Procedure N1-TSP-201-550, "Local Leak Rate Test — Summary (Type B and C Tests)." This procedure is used to track the combined leakage rate for all penetrations subject to Type B and C Tests following a Type A Test to verify that the measured combined leakage rate is less than the Appendix J allowable leakage rate of 0.60 La and that the leakage rate limits of TS 4.3.3.f(1)(b)(i) and (ii) and 4.3.3.f(1)(c) are not exceeded. The NRC staff determined that the leakage rate totals were consistent with the requirements of Appendix J and the TS as of January 29, 1993. The NRC staff verified that the leakage rates from all primary containment isolation valves requiring Type C testing were included. Independent calculations of the total Type C leakage rates, based on the test data in the procedure, confirmed the accuracy of the value determined by.NMPC.

Step 9.8 of the procedure indicated that the leakage rate limit of TS 4.3.3.f(1)(b)(i) applies to the sum of the leakage rates from testable penetrations and the isolation valves listed in the TS tables. Six normally closed manual isolation valves (72-479, 72-480, 114-114, 114-116, 110-165, and 110-166) were not included in the TS tables in the February 7, 1992, license amendment request. However, leakage rate values for these valves were properly included in the calculation for combined Type C leakage rates. This inconsistency was corrected by adding four of these manual isolation valves to TS Table 3.3.4 in the February 18, 1993, supplement to the February 7, 1992, submittal. The other two valves were not included in the TS tables and will be deleted from Type C testing since they are no longer classified as containment isolation valves; this penetration has been capped and will be tested as part of Type B testing.

The NRC staff reviewed Drawing F-45089-C, Sheets 8 through 10, and verified that test procedures have been identified for all of the containment isolation valves requiring Type C testing per Appendix J of 10 CFR Part 50.

<u>Leakage Rate Testing of Water-Sealed Containment Isolation Valves</u>

Section III.C.3 of Appendix J to 10 CFR Part 50, states that leakage from containment isolation valves that are sealed with fluid (water, for NMP-1) from a seal system may be excluded when determining the combined leakage rate for all penetrations and valves subject to Types B and C tests, provided that

- Such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the TS or associated bases, and

- The installed isolation valve seal-water system fluid inventory is sufficient to ensure the sealing function for at least 30 days at a pressure of 1.10 Pa.

The February 7, 1992, license amendment request which was supplemented by the February 18, 1993, submittal and approved by License Amendment No. 140 issued on April 12, 1993, specifies in the TS that the maximum allowable water leakage rate from water-sealed valves shall be limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm. These water leakage rate limits are consistent with the requirements of Paragraph 4.2.2.3(e) of the ASME Operations and Maintenance Standards Part 10 (OM-10) of the 1989 Edition of Section XI of the ASME Code which was incorporated by reference in Section (b) of 10 CFR 50.55a, effective September 8, 1992 (57 FR 34666).

The NRC staff reviewed the most recent leakage rate test results of valves designated in the TS as being subject to water-seal testing and determined that all such valves met their applicable leak test requirements. The NRC staff concluded that, based on the provisions of Section III.C.3. of Appendix J, the leakage rates from these water-sealed valves may be properly excluded when determining the combined leakage rate for all penetrations and valves subject to Types B and C tests.

The NRC staff reviewed Note (6) of TS Table 3.3.4 in the February 7, 1992, proposed license amendment. Note (6) states that the following valves have a water-seal capability and that no Appendix J or IST leakage rate testing is required:

- Valves 63.1-01, 63.1-02, 05-05, and 05-07 are properly excluded from Appendix J and IST leakage rate testing since these valves have no atmospheric leak path.

- Valves 80-15, 80-16, 80-17, 80-18, 80-19, 80-35, 80-36, 80-37, 80-38, 80-39, 80-65, 80-66, 80-67, and 80-68 have adequate water seals that did not require water leak rate tests according to the NRC staff's safety evaluation of May 6, 1988.

Therefore, the NRC staff concluded that these water-sealed valves are properly excluded from Appendix J and IST leakage-rate testing.

Inservice Testing of Containment Isolation Valves

The NRC staff reviewed Revision 3 of the Second 10-Year Inservice Testing Program Plan for NMP-1 and verified that the plan included the independently developed list of containment isolation valves and appropriate exercising and stroke time test requirements (for power-operated valves). The NRC staff reviewed the following two surveillance tests which implement the IST requirements:

- N1-ST-04, "Reactor Coolant System Isolation Valves Operability Test," performed November 16-18, 1992

- N1-ST-05, "Primary Containment Isolation Valves Operability Test," performed on November 7, 1992

This review revealed that all the isolation valves listed in the procedures had been exercised and, if required, stroke time tested. The procedures specified stroke time limits and the measured results were consistent with the IST program and, if specified, with the TS limits. On the basis of these reviews of IST data, the NRC staff concluded that all containment isolation valves listed in the procedures have been properly exercised and stroke time tested as part of the licensee's ongoing IST program.

The NRC staff also verified that test procedures are in place for all required IST testing of containment isolation valves.

UFSAR Update

In its January 29, 1993, letter, NMPC committed to update the UFSAR and correct deficiencies therein by June 30, 1993. The NRC staff will verify, as part of its routine reviews of UFSAR updates, that UFSAR Tables VI-3a, VI-3b, and VI-3c have been corrected.

Containment Isolation Valves — Conclusion

In summary, the NRC staff concluded that (1) the containment isolation valve deficiencies identified by the Petitioner were administrative in nature; (2) notwithstanding the administrative deficiencies, the operability of the containment isolation valves was being maintained in accordance with the requirements of the TS and IST program and the valves were being properly tested in accordance with all applicable regulatory requirements; (3) the leakage rates of water-sealed valves were properly excluded when determining the combined leakage rate for all penetrations and valves subject to Type B and C tests; (4) the licensee has committed to update the UFSAR by June 30,

1993, to correct the identified deficiencies; and (5) License Amendment No. 140, issued on April 12, 1993, to the Nine Mile Point Nuclear Station Unit No. 1 Facility Operating License DPR-63 corrected the administrative deficiencies related to the containment isolation valve listings in the TS. Therefore, to the extent that the Petitioner sought correction of the TS tables to correctly list the NMP-1 containment isolation valves, their initiating signals, and their stroke times, this relief has been granted. As stated above, NMPC has committed to correct the UFSAR tables by June 30, 1993. Action to require earlier change to the UFSAR tables is not needed in light of the NRC staff's confirmation of valve operability during an onsite inspection conducted February 1-5, 1993. Petitioner's request for other actions based on containment isolation valve deficiencies is denied.

(3) <u>NMPC, NMPC's quality assurance group, and the NRC have reviewed these safety concerns and, contrary to any practical justification, have remained silent</u>

The Petitioner was employed at NMP-1 as a contractor from November 13, 1989 to January 18, 1990. During that employment, the Petitioner expressed several concerns to NMPC regarding the design and operation of the NMP-1 feedwater system in its HPCI mode and what he believed were various inconsistencies in the listings of the containment isolation valves in the TS, in the UFSAR, and on the plant drawings.

The NRC staff has reviewed NMPC records regarding the processing of the Petitioner's concerns by the NMPC Regulatory Compliance Group and by the NMPC Quality First Program. A summary of NMPC's consideration of the Petitioner's concerns follows.

. Review of Concerns by NMPC Regulatory Compliance Group

The Petitioner initially submitted his concerns regarding the design and operation of the feedwater system in its HPCI mode and what he believed were various inconsistencies in the listings of the containment isolation valves in the TS, in the UFSAR, and on the plant drawings to the NMPC Regulatory Compliance Group during January 1990. In a letter dated July 31, 1990, to NMPC, the Petitioner subsequently also submitted these concerns to the NMPC Quality First Program (Q1P). The NRC staff review of NMPC records disclosed that the concerns the Petitioner submitted to the NMPC Regulatory Compliance Group and to the NMPC Q1P covered the same topics he submitted to the NRC in his 10 CFR 2.206 Petition dated October 27, 1992, and evaluated herein by the NRC staff.

NMPC evaluated the Petitioner's concerns regarding the feedwater system operating in the HPCI mode during February 1990 and determined that the NMP-1 accident analyses do not rely on the HPCI system for mitigation of any accidents. NMPC's conclusion regarding operation of the feedwater system in the HPCI mode was documented in an internal memorandum of February 28, 1990, and was consistent with the conclusion reached by the NRC staff in this Director's Decision. After reviewing NMPC records, the NRC staff concluded that NMPC had properly reviewed the Petitioner's concerns regarding the HPCI system.

The NRC staff reviewed NMPC records which showed that in January 1990, the Petitioner communicated to the NMPC Regulatory Compliance Group what he believed were various inconsistencies in the listings of containment isolation valves in the TS, in the UFSAR, and on the plant drawings. The Petitioner also expressed concerns about the performance of IST and leak tests according to the requirements of Appendix J. NMPC reviewed the Petitioner's concerns between January and July 1990. NMPC determined that some of the Petitioner's concerns had been previously reviewed and found acceptable in NRC staff-approved safety evaluations and that some of his concerns had been resolved by issuance of NRC staff-approved schedular exemptions. NMPC also referred the Petitioner's list of concerns to the NMPC Licensing group to ensure that applicable concerns would be addressed by including them in the license amendment then in preparation with the purpose of resolving deficiencies identified in the NRC staff safety evaluation of May 6, 1988. After reviewing NMPC records, the NRC staff concluded that the NMPC Regulatory Compliance Group had processed the Petitioner's concerns in an appropriate manner.

Review of Concerns by NMPC Quality First Program

The Petition stated that following a perceived period of inaction by NMPC, the Petitioner notified the NMPC Q1P of his concerns regarding (1) operation of the feedwater system in the HPCI mode and (2) the containment isolation valves.

The Q1P is an NMPC program designed to give its employees a confidential forum for reporting potential problems that affect quality or safety on the job. Q1P is directed by the NMPC Quality Assurance Department and applies to

the receipt, control, investigation, resolution, feedback to the originator, and reports to NMPC management of any concerns identified. Q1P is not governed by NRC regulatory requirements, except as related to protected activities by employees. Although NMPC employees are encouraged to report potential problems to the NMPC Q1P during their employment and upon termination of employment, NMPC representatives stated that NMPC personnel had searched the Q1P files and found no record of the Petitioner contacting the Q1P prior to receipt of a letter from the Petitioner dated July 31, 1990. NMPC informed the NRC staff that Q1P records were not considered plant records unless a valid quality concern was determined to exist. Therefore, it is possible that no records exist because previous contacts may have been made but had been treated as having no basis.

The NRC staff reviewed a copy of the letter NMPC received from the Petitioner dated July 31, 1990, in which concerns regarding operation of the feedwater system in the HPCI mode and the containment isolation valves were outlined. These concerns repeated the ones made previously by the Petitioner to the NMPC Regulatory Compliance Group.

The NRC NMP-1 resident inspectors were informed by a Q1P representative on August 6, 1990, that the July 31, 1990, letter had been received. According to records reviewed by the NRC staff, NMPC had reviewed the Petitioner's concerns between August and November 1990. These records showed that NMPC closed out these concerns on November 28, 1990, after contacting the Petitioner and obtaining his agreement for closure. NMPC again determined that the concerns regarding operation of the feedwater system in the HPCI mode had no basis since the NMP-1 safety analyses do not rely on operation of the

feedwater system in the HPCI mode to satisfy the emergency core cooling requirements of 10 CFR 50.46. The NMPC Licensing group received the concerns regarding the containment isolation valves for consideration in the proposed license amendment development. The NRC staff concluded that the NMPC Q1P organization processed the Petitioner's concerns appropriately.

As noted in the discussion of operation of the feedwater system in the HPCI mode, information regarding the design features of the NMP-1 feedwater system, including operation in the HPCI mode, has been readily available in the public records and the NRC staff was well aware of this information over the life of the NMP-1 plant. The NRC staff concerns about NMP-1 compliance with the requirements of 10 CFR Part 50, Appendix J, have been a matter of public record since May 6, 1988, when the NRC staff issued its letter with its attached safety evaluation. As noted above, the NRC staff concluded that NMPC's Regulatory Compliance Group and Q1P representatives handled the Petitioner's concerns in an appropriate manner. Therefore, I have concluded that the Petitioner's assertion that NMPC, NMPC's quality assurance group, and the NRC have known of these safety concerns and have remained silent has no basis. Accordingly, the Petitioner's request for enforcement action against NMP-1 on this part of the Petition is denied.

Although I have denied this portion of the Petition, 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.

<u>Insurance</u>

The Petition asserts that NMPC is not insured to operate NMP-1 in the manner described in the Petition. In order to operate a commercial nuclear

power plant within the United States with "limited liability," an NRC licensee must have and maintain financial protection (e.g., liability insurance). The Price-Anderson Act requires NMPC to provide $200 million in liability insurance for public liability claims that might arise from a nuclear accident at the NMP-1 site. In addition, NMPC (along with all other commercial nuclear power plant licensees) must participate in an industry self-insurance plan which subjects it to a potential liability of $63 million for each commercial nuclear power plant that it operates for public liability claims that might arise from a single nuclear accident at NMP-1 <u>or any other commercial nuclear power plant licensed by the NRC</u>. This liability insurance cannot be purchased from the nuclear liability insurance pools unless the pools are satisfied that a licensee is operating its commercial nuclear power plant in accordance with NRC regulations. Contrary to the assertions in the Petition, NMPC has obtained and is maintaining the appropriate amount of liability insurance.

CONCLUSION

The Petitioner requested that the NRC issue an immediately effective order directing NMPC to cease power operation of NMP-1 and to place the reactor in a cold-shutdown condition pending full compliance with NRC regulations. The Petition also asked the Commission to hold a public hearing before authorizing resumption of plant operation.

On April 12, 1993, the NRC staff issued License Amendment No. 140 to the NMP-1 Facility Operating License DRP-63. This license amendment corrects the NMP-1 TS tables which list the containment isolation valves, their initiating signals, and their stroke times. To the extent the Petitioner sought such

corrections, this relief has been granted. Further, NMPC has committed to update the UFSAR, by June 30, 1993, to list the containment isolation valves correctly. The NRC staff will verify this commitment as part of its routine reviews of UFSAR updates. The NRC staff views these changes as administrative corrections since the NRC staff has concluded that all relevant valves were appropriately tested. With regard to the other requests made by the Petitioner, an immediate shutdown of NMP-1 and the institution of a public hearing before authorizing resumption of plant operation, as requested by the Petitioner, is appropriate only where substantial health and safety issues have been raised. See *Consolidated Edison Company of New York* (Indian Point, Units 1, 2, and 3), CLI-75-8, 2 NRC 173, 175 (1975), and *Washington Public Power Supply System* (WPPSS Nuclear Project No. 2), DD-84-7, 19 NRC 899, 923 (1984). For the reasons discussed above, I find no basis for taking such actions. Rather, on the basis of the review efforts by the NRC staff, I conclude that no substantial health and safety issues have been raised by the Petitioner. Accordingly, the Petitioner's remaining requests for action pursuant to 10 CFR 2.206 are denied.

A copy of this Decision will be placed in the Commission's Public Document Room, Gelman Building, 2120 L Street, NW, Washington, DC 20555, and at the Local Public Document Room, Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

A copy of this Decision will also be filed with the Secretary for the Commission's review as stated in 10 CFR 2.206(c) of the Commission's regulations.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 9th day of May 1993.

REVIEW OF 18 NOTES IN ATTACHMENT 5 TO PETITION

Attachment 5 to the Petition contained a list of 88 containment isolation valves which were asserted by the Petitioner to have deficiencies as listed in the TS tables, the UFSAR tables, or on the plant drawings. The asserted deficiencies were summarized in the 18 notes of Attachment 5. Some specific information (valve identification numbers were not fully legible in Attachment 5 and Note 17 was missing from Attachment 5) was missing in the Petition. The illegible or missing information was submitted in the Petitioner's response received on January 5, 1993, or in a January 11, 1993, telephone conversation between the Petitioner and the NRC Project Manager for NMP-1. The following is a listing of the Petitioner's 18 notes and the NRC staff's findings for each of the notes.

Note 1: FSAR Section VII requires these valves to go open within 20 sec Hi Drywell or low-low reactor level RPS signal and this times [sic] fails to appear in either TS Table 3.3.4 or FSAR Table VI-3a. Also, these valves are 10 CFR 50 Appendix A Criterion 55 valves and are not being tested accordingly.

Findings: Note 1 applies to core spray valves 40-01, 40-02, 40-09, 40-10, 40-11 and 40-12. NMPC determined that the correct maximum opening time for these valves is 22.5 seconds, as indicated in Revision 8 of UFSAR Section VII.A.4.0 (page VII-9). License Amendment No. 140 (TS page 119) is consistent, requiring a 22.5-second opening time

on reactor water level low-low or high drywell pressure signals coincident with reactor vessel pressure less than 365 psig. Although the GDC do not apply to NMP-1, as stated in the Director's Decision to which this is attached, the NRC staff has verified that these valves were being properly tested in accordance with applicable requirements.

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.

Findings: Note 2 applies to the remote manual containment spray test valve 80-118. The NRC staff reviewed NMPC Safety Evaluation (SE) No. 89-13, which considered reduction of containment spray flow due to valve 80-118 remaining open. NMPC concluded that sufficient system flow is available under accident conditions even if valve 80-118 fails in the fully open position. The NRC staff reviewed NMPC's approved SE No. 89-13 (Revision 5) dated September 15, 1991, and concluded that it provided an appropriate basis for concluding that sufficient flow would be available. Valve 80-118 was added to TS Table 3.3.4 (TS page 148) in License Amendment No. 140. Although the GDC do not apply to NMP-1, as stated in the Director's

Decision to which this is attached, the NRC staff verified that these valves were being properly tested in accordance with applicable requirements. The licensee committed in its January 29, 1993, submittal to update the UFSAR.

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.

Findings: Note 3 applies to containment spray valves 80-15, 80-16, 80-35, and 80-36. These are normally open valves. License Amendment No. 140 (TS page 148) shows that these valves open on remote manual initiation (not RPS logic). This is consistent with Drawing C-18012C, Sheet 2, Revision 36. Although the GDC do not apply to NMP-1, as stated in the Director's Decision to which this is attached, the NRC staff verified that these valves were being properly tested in accordance with applicable requirements. The licensee committed in its January 29, 1993, submittal to update the UFSAR.

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.

Findings: Note 4 applies to scram discharge volume valves 44.2-15 and 44.2-16. The 18-second closing time previously listed in the UFSAR was recognized by the licensee as being in error. The error was corrected and Revision 9 of UFSAR Table VI-3a shows a closing stroke time for these valves of 10 seconds, consistent with TS page 119 of License Amendment No. 140. The NRC staff reviewed NMPC SE No. 89-033 which was approved by NMPC on December 18, 1989, and concluded that it provided an appropriate basis for this change.

Note 5: FSAR Table 3a [VI-3a] shows RPS logic to close with core spray actuation while TS Table 3.2.7 does not.

Findings: Note 5 applies to core spray high point vent valves 40-30, 40-31, 40-32, and 40-33. Revision 2 of UFSAR Table 3a (page VI-48) indicates that these valves close on low-low water level, high drywell pressure, or core spray actuation signals. License Amendment No. 140 (TS page 119) states that the valves close on low-low water level or high drywell pressure. Although both the UFSAR and license amendment are correct and in agreement, the licensee committed in its submittal of January 29, 1993, to change the UFSAR to eliminate reference to the core spray actuation signal since it is redundant. Core spray actuation is initiated by low-low water level or high drywell pressure signals.

Note 6: FSAR Table 3b [VI-3b] shows these valves with a 70 second and 90 second stroke time. These valves should appear on TS Table 3.3.4.

Findings: Note 6 applies to containment spray discharge to rad waste valves 80-114 and 80-115. The containment isolation function for this penetration is provided by valve 80-118 rather than by valves 80-114 and 80-115. Therefore, valves 80-114 and 80-115 are not included in the TS tables. Valve 80-118 is included in TS Table 3.3.4 (TS page 148) of License Amendment No. 140. The licensee committed in its January 29, 1993, submittal to update the UFSAR.

Note 7: P&ID 18014C sht [sheet] 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.

Findings: Note 7 applies to containment atmosphere monitoring valves 201.7-08 and 201.7-09. License Amendment No. 140 (TS page 148a) includes these valves and shows that they automatically close on low-low water level or high drywell pressure signals with a maximum allowable stroke time of 60 seconds, but they are not in the UFSAR. The licensee committed in its January 29, 1993, submittal to update the UFSAR.

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 [sic] these valves are identified in surveillance test (N1-ST-Q5) as TS acceptance criteria.

Findings: Note 8 applies to the #12 containment H_2/O_2 analyzer system valves 201.2-23 through 201.2-30 (8 valves). License Amendment No. 140 (TS page 147) includes these valves and shows that they close on low-low water level or high drywell pressure signals with a maximum stroke time of 60 seconds, which is consistent with the UFSAR.

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.

Findings: Note 9 applies to emergency cooling vent and drain valves 05-02, 05-03, 39-11, 39-12, 39-13, and 39-14 and recirculation system sampling valves 110-127 and 110-128. TS page 119a of License Amendment No. 140 includes valves 110-127 and 110-128 and shows that they have a maximum allowable closing time of 20 seconds and close automatically on specified signals. NMPC deleted valves 05-02, 05-03, 39-11, 39-12, 39-13, and 39-14 from TS Table 3.2.7 in order to minimize extensions to the containment. The licensee

committed in a December 21, 1992, letter commenting on the Petition, to issue a document change request to revise Drawing C-18006-C to properly identify the RPS logic inputs.

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

Findings: Note 10 applies to torus fill from condensate system valve 58.1-01 and head spray valve 34-01. Neither valve is included in License Amendment No. 140 since they are not containment isolation valves. The licensee committed in the January 29, 1993, submittal to update the UFSAR to reflect the current status of these valves.

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.

Findings: Note 11 applies to the #11 containment H_2/O_2 analyzer system valves 201.2-109, 201.2-110, 201.2-111, 201.2-112, 201.7-01, 201.7-02, 201.7-03, 201.7-04, 201.7-10, and 201.7-11 and post-LOCA vent valves 201.1-09, 201.1-11, 201.1-14, and 201.1-16. License Amendment No. 140 (TS pages 147 and 148a) includes these valves and indicates that they automatically close on low-low water level or high drywell pressure and have a maximum allowable stroke time of 60 seconds. The current revision (Revision 2) of pages VI-50 and VI-50A of UFSAR Table VI-3b also indicates that these valves close

automatically on low-low water level or high drywell pressure and have a maximum allowable stroke time of 60 seconds.

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

Findings: Note 12 applies to postaccident sampling valves 63-04, 63-05, and 122-03, and normal reactor building ventilation valves 202-07, 202-08, 202-35, and 202-36. Drawing C-18013C, Revision 23, shows that valves 202-07, 202-08, 202-35, and 202-36 are not primary containment isolation valves. These valves are actuated by signals from the reactor building protection system, rather than by the RPS. License Amendment No. 140 (TS page 119a) includes valve 122-03 and indicates that it closes automatically on specified signals with a maximum allowable stroke time of 30 seconds. Valves 63-04 and 63-05 are no longer considered containment isolation valves and, therefore, were not included in License Amendment No. 140 and should be deleted from UFSAR Table VI-3b. The licensee committed in its January 29, 1993, submittal to update the UFSAR.

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.

Findings: Note 13 applies to feedwater system valves 30-31, 30-32, and 29-51. These valves are not identified in the TS or UFSAR tables since they are not containment isolation valves, as shown on Drawing C-18005C, Sheets 1 and 2. These HPCI-related valves are beyond containment boundary valves 31-01R and 31-02R. HPCI valves are not safety related and therefore, are 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.

Findings: Note 14 applies to traversing incore probe (TIP) valves TIP-1, TIP-2, TIP-3, and TIP-4. (These valves are also identified as 36-147, 36-148, 36-149, and 36-150, respectively.) License Amendment No. 140 (TS page 148a) includes these valves and indicates that they automatically close on low-low water level or high drywell pressure with a maximum acceptable stroke time of 60 seconds. The licensee committed in its January 29, 1993, submittal to update the UFSAR. NOTE 1 on pages III-12-1 and -2 of the Second Ten-Year Interval IST Program dated October 12, 1992, states that these valves are not shown on P&ID C-18014-C, Sheet 2. This is acceptable, as the valves are adequately tracked in the IST program.

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

Findings: Note 15 applies to reactor building closed-loop cooling water valves 70-92, 70-93, 70-94, and 70-95. Valves 70-92 and 70-94 are remote manual valves and valves 70-93 and 70-95 are self-actuated check valves; all are in closed loops inside the containment. License Amendment No. 140 (TS page 148a) includes these four valves and indicates that the two remote manual valves (70-92 and 70-94) have maximum permissible stroke times of 60 seconds. The UFSAR lists a 30-second stroke time. The licensee committed in its January 29, 1993, submittal to update the UFSAR.

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.

Findings: Note 16 applies to reactor building closed-loop cooling water valves 70-92, 70-93, 70-94, and 70-95. License Amendment No. 140 (TS page 148b, Notation 5) states that the valves do not require leak testing as they do not meet the requirements of Section II.H of 10 CFR Part 50, Appendix J. The valves do not require Appendix J leak rate testing since they provide isolation for a closed-loop inside containment. Although the GDC do not apply to

NMP-1, as stated in the Director's Decision to which this is attached, the NRC staff verified that these valves were being properly tested in accordance with applicable requirements.

Note 17: TS Table 3.3.4 identified these valves as Criterion 56, however, are not being tested according. FSAR Table VI-3b shows these valves as lines entering free space of containment yet are not being tested according.

Findings: Note 17 applies to containment spray valves 80-01, 80-02, 80-15, 80-16, 80-17, 80-18, 80-21, 80-22, 80-35, 80-36, 80-37, 80-38, 80-65, 80-66, 80-67, and 80-68. License Amendment No. 140 (TS page 148) includes all of these valves. Valves 80-17, 80-18, 80-37, 80-38, 80-65, 80-66, 80-67, and 80-68 are self-actuating check valves and, therefore, do not require stroke time testing. Valves 80-15, 80-16, 80-35, and 80-36 are normally open remote manual valves with a maximum allowable stroke time of 60 seconds. Valves 80-01, 80-02, 80-21, and 80-22 are normally open remote manual valves with a maximum allowable stroke time of 70 seconds. Although the GDC do not apply to NMP-1, as stated in the Director's Decision to which this is attached, the NRC staff verified that these valves were being properly tested in accordance with applicable requirements. The licensee committed in its January 29, 1993, submittal to update the UFSAR.

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.

Findings: Note 18 applies to the core spray system pump suction valves 81-01, 81-02, 81-21, and 81-22. License Amendment No. 140 (TS page 148) includes these valves and shows that they are remote manual valves with a maximum stroke time of 90 seconds, which is consistent with the UFSAR. Note 4 on TS page 148b of License Amendment No. 140 states that these valves are provided with a water seal and will be tested during each refueling outage not to exceed 2 years, consistent with Appendix J water-seal requirements. Leakage rates are not to exceed 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm. Although the GDC do not apply to NMP-1, as stated in the Director's Decision to which this is attached, the NRC staff verified that these valves were being properly tested in accordance with applicable requirements.