U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION I

Report No. 50-220/80-18

Docket No. 50-220

License No. DPR-63 Priority - Category C

Licensee: Niagara Mohawk Power Company

Facility Name: Nine Mile Point Nuclear Station, Unit One

Inspection At: Scriba, New York

Inspection Conducted: October 8, 10, 20-31, and November 12, 18, 1980

Inspectors: Edward J. Greenman /jm — 11/26/80
S. D. Hudson, Resident Inspector — date

J. R. White, Chief, Radiation Specialist — 11/26/80 — date

Edward J. Greenman /jm — 11/26/80
H. B. Kister, Chief, Reactor Project Section #4, RO&NS Branch — date

Approved by: Edward J. Greenman /jm — 11/26/80
E. J. Brunner, Chief, Reactor Operations and Nuclear Support Branch — date

Inspection Summary:
Inspection on October 8, 10, 20-31, and November 12, 18, 1980 (Inspection Report 50-220/80-18)
Areas Inspected: Special, unannounced inspection (79 hours) by the resident and regional office inspectors in the area of conformance with requirements of and commitments to NUREG-0578, TMI Short Term Lessons Learned, Category A items.
Results: One item of noncompliance was identified, and several items require further discussion and clarification to determine acceptability.

DETAILS

1. Persons Contacted

R. Abbott, Operations Supervisor
J. Aldrich, Superintendent of Training
J. Duell, Supervisor, Chemistry and Radiation Protection
M. Jones, Planning Coordinator
E. Leach, Superintendent, Chemistry and Radiation Management
T. Perkins, General Superintendent, Nuclear Generation
*T. Roman, Station Superintendent
M. Silliman, Technical Superintendent

2. Purpose of Inspection

The purpose of this inspection was to review the licensee's completion status regarding certain TMI Short Term Lessons Learned (NUREG 0578) Category A requirements. This is in response to an apparent failure of the licensee to complete the interim requirements for NUREG 0578 Item 2.1.8.b as determined by a Health Physics Appraisal Inspection, the results of which are also contained herein.

As a result of the Health Physics Appraisal Inspection finding, an Immediate Action Letter was issued (IAL 80-40) on October 17, 1980 confirming the licensee's commitment to (1) complete the interim item 2.1.8.b requirements and (2) report the results of their audit pertaining to the status of completion of all other Category A requirements to the Director, NRC Region I no later than the close of business October 24, 1980. An inspection conducted by a Region I Radiation Specialist on October 23, 1980 verified that the corrective actions by the licensee for Item 2.1.8.b had been completed. The results of the audit were provided to Region I on October 23, 1980.

3. Licensee Action on Previous Inspection Findings

As stated below.

4. Inspection Findings

In order to determine that the requirements of the NUREG 0578 Category A Items had been satisfied the inspectors reviewed the following known related documents and correspondence:

References

(1) September 13, 1979 letter from Denton to licensee forwarding NUREG 0578 and the requirements for implementation.

(2) Niagara Mohawk Power Corporation letter, Rhode to Eisenhut dated October 18, 1979, which provided information and commitments relating to NUREG 0578.

(3) October 30, 1979 letter from Denton to licensee which provided additiona clarification of NUREG 0578 requirements.

(4) Niagara Mohawk Power Corporation (NMPC) letter, Bartlett to Denton dated November 26, 1979 which provided several commitments to NUREG 0578.

(5) NMPC letter, Dise to Denton dated December 19, 1979, providing additional information regarding NUREG 0578 commitments.

(6) NMPC letter, Dise to Denton dated December 31, 1979, providing additional commitments and information to NUREG 0578.

(7) NRC Order to Show Cause to NMPC dated January 2, 1980.

(8) NMPC Response to the NRC Order to Show Cause dated January 22, 1980.

(9) NRR letter, Ippolito to NMPC dated March 21, 1980, results of NRR staff evaluation of NMPC's actions taken to satisfy Category A requirements.

The results of the document and correspondence review and inspections performed are provided below:

a. Item 2.1.3.a - DIRECT INDICATION OF SAFETY RELIEF AND SAFETY VALVE POSITION

Requirements of Denton Letter of Clarification dated 10-30-79

(1) Provide operator with unambiguous indication of valve position (open/closed).

(2) Valve position should be indicated in the Control Room--alarm should also be provided in conjunction with the indication.

(3) Valve position indication may be safety grade, if not a reliable single channel direct indication powered from a vital instrument bus may be provided if back up methods of determining valve position are available and discussed in the Emergency Procedures.

(4) Should be seismically qualified--if not by 1-1-80, provide justification and a schedule for upgrading.

(5) Environmentally qualified by 1-1-80, if not provide a proposed schedule.

Licensee Commitments

NMPC Letter- Bartlett to Denton dated November 26, 1979.

NMPC Letter- Dise to Denton dated December 19, 1979.

NMPC agreed to install Acoustic Monitoring System within 30 days after receipt of equipment.

Inspection Findings

The licensee appears to have satisfied the requirements of this item except for paragraph 2 above.

The licensee installed the acoustic monitoring system to give positive position indication of each safety and relief valve during the March 1980 outage.

Individual valve position, however, is only available (new system) in the auxiliary control room (relay room) below the control room. If any valve is opened, a common annunciator alarms in the control room. The licensee stated that he does not plan to provide individual valve position from the new system in the control room. The licensee further stated that the new system will not be inputed to the plant process and event recorder as was previously indicated by the NRR evaluation dated March 21, 1980.

For information, the originally installed system also provides individual valve position indication through the use of SRV pilot valve indication and temperature indicators located on the tailpipe downstream of the valve discharge point. They read out and alarm on the process computer.

The licensee letter, Dise to Denton dated May 12, 1980, stated that seismic and environmental qualification of the acoustic monitoring system will not be completed until July 1981.

This item is considered unresolved pending further clarification regarding the acceptability of the licensee's installation relating to the requirement for individual valve position indication in the control room. (50-220/80-18-01)

b. Item 2.1.4 - CONTAINMENT ISOLATION

Denton Letter ofClarification dated 10-30-79

(1) Provide diverse containment isolation signals that satisfy safety grade requirements.

(2) Identify essential and non-essential systems and provide results to NRC.

(3) Non-essential systems should be automatically isolated by containment isolation signals.

(4) Resetting of containment isolation signals shall not result in the automatic loss of containment isolation.

Licensee Commitments

References:

(a) NMPC letter-Dise to Ippolito dated November 28, 1979
(b) NMPC letter-Dise to Denton dated December 31, 1979
(c) NRR letter-Ippolito to Dise dated March 21, 1980
(d) NMPC letter-Bartlett to Denton dated November 26, 1979

Commitments

(1) Plant design meets NRC position (licensee statement)

(2) Licensee will identify systems

(3) Any required modifications (Ref. (d)) resulting from a design review to ensure that: non-essential systems are automatically isolated or are normally isolated by manual valves will be performed by 1-1-80.

All non-essential systems that penetrate primary containment are automatically isolated except as noted in response (Ref. (b)) to position 2.

(4) Design of control system for auto isolation meets NRC position. (licensee statement)

(5) Two systems identified (Ref. (b)) had been previously identified by IE Bulletin 79-08. In their response; the licensee committed to install containment isolation valves by March 1981 in their response to the Bulletin.

The licensee's apparent rationale for not meeting the 1-1-80 date for item 5 above is that the existing commitment has been accepted by the NRC as a result of previous correspondence and remains valid.

Inspection Findings

The licensee appears to have satisfied the requirements of this item except for paragraphs 1 and 3 of the Denton clarification letter. This conclusion is based on the inspector's review of the licensee's response dated 12-31-79. The inspector noted that the Main Steam System (including warm-up and emergency cooling vents) the Reactor Water Cleanup System and the Shutdown Cooling System isolate on low-low water level only and questions whether or not it meets the requirement for diverse containment isolation signals recognizing that closing signals other than containment isolation (e.g., RPS) provide valve closure.

The inspector also noted that the Reactor Head Spray, Suppression Chamber water make-up, and the atmosphere to suppression chamber vacuum relief system are normally isolated by a normally shut air-operated or motor-operated valve and a check valve; however, the air-operated or motor-operated valves in these systems do not receive containment isolation signals.

The recirculation loop sample line and suppression chamber to waste system line are each normally isolated by two normally shut manual valves. The licensee has committed to installing automatic containment isolation valves in these lines during the March 1981 refueling outage in accordance with a previous commitment to IEB 79-08. However, this does not necessarily comply with paragraph 3 of Denton's clarification which requires compliance by 1-1-80. Unless, that is, the use of should is meant to imply that this is a recommendation only. If so, clarification is required how the Order to Show Cause applies to recommendations.

The inspector did, however, verify that the licensee has modified his procedure No. N1-SP-11, Reactor Water Sampling with Cleanup system Isolated, Revision 1 dated 12-3-79 and procedure No. N1-OP-14, Containment Spray System, Revision 8 dated 12-26-79 to require that an operator be stationed to isolate these systems if necessary, when they are in use. This is in accordance with their commitment in their 12-31-79 submittal, and apparently agreed upon by NRR in their March 21, 1980 evaluation..

On a sampling basis, the inspector also independently verified that the Main Steam isolation valves and the drywell floor drain isolation valves will not re-open as a result of resetting or clearing of the containment isolation signal.

This item is considered unresolved pending further discussion and clarification of the acceptability of the licensee's position relating to diverse isolation signals and automatic isolation of non-essential systems. (50-220/80-18-02)

c. Item 2.1.6.a - SYSTEMS INTEGRITY
Requirements of Denton Letter of Clarification dated 10-30-79

(1) Provide a summary description of the program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a erious transient or accident by January 1, 1980.

(2) Include a list of systems which are excluded from this program.

(3) Testing of gaseous systems should include helium leak detection or equivalent methods.

(4) Program to consider leakage reduction for potential release paths due to design and operator deficiencies.

Licensee Commitments

(1) Program

(a) Core Spray, Containment Spray

(1) Visually inspect for leakage and perform necessary maintenance once per quarter.

(2) Perform leakage test with helium once per cycle.

(b) Emergency Condenser

Leak test once per cycle using a helium tracer.

(c) Vacuum Relief System

Leak test once each operating cycle using soap bubble test.

(d) Primary Containment H_2-O_2 Monitor System

Leak test once each operating cycle using a helium tracer.

(e) Shutdown Cooling

Physical inspection of system while operating each operating cycle. If required, use helium tracer.

(f) Cleanup System

Physical inspection of system while operating in letdown mode once each cycle. Use helium tracer if required.

(g) Reactor Water Sampling

Leak test with sample flowing using helium tracer once each cycle.

(2) Provided list of systems included plus a description of types of systems excluded with basis for exclusion. This was the reverse of what Item 2 required.

(3) Committed to using helium, if required.

(4) Release pathways due to design and operator deficiencies considered but determined by NMPC not to apply.

Inspection Findings

The licensee appears to have satisfied the requirements of this item based on the inspectors review of the licensee's submittal of 12-31-79.

The inspector noted that the leakage testing program had not yet been formally established. Also, formal procedures for conducting the test of the fluid systems and ensuring that systems are re-tested once each cycle have not yet been prepared. However, as stated in the licensee's letter, Lempges to Denton, dated July 30, 1980, six of the original eight systems included in the program have been leak tested. An additional one of the original eight was tested during the September 1980 outage. (Shutdown Cooling)

As a result of further discussions with the resident inspector, the licensee has agreed to add the Containment Atmosphere Dilution (CAD) System and the Drywell Continuous Air Monitoring (CAM) System to the leakage testing program.

d. Item 2.1.8.a - POST ACCIDENT SAMPLING

The Requirements of the Denton Clarification Letter of 10-30-79

Minor Plant modifications for taking samples, design review, and procedural modifications (if necessary) shall be completed by 1-1-80.

Licensee Commitments

No commitments were made in the licensee's 12-31-79 Dise to Denton letter to the above clarification, however their letter Rhode to Eisenhut dated October 18, 1979 stated that existing procedures would be reviewed and revised as necessary by January 1, 1980.

Inspection Findings

The licensee procedures presently appear to satisfy only the requirements for reactor coolant sampling during the more probable event conditions. The design review and the results of the shield survey indicated that under present conditions, with the sample sink in its present location, and assuming the radioactive source term given in RG 1.3 and 1.4, a reactor water sample could probably not be obtained for up to 30 days with the present installation without the potential for exceeding NRC exposure limits.

The inspector reviewed Procedure No. N1-PSP-10 Reactor Water Sampling-Suspected High Activity, Revision 0, dated 3-12-80, and noted that valve manipulation and sampling is accomplished in the Reactor Building. The licensee stated that a reactor water sample could be obtained in 2 to 3 hours using the above procedure for the more probable event conditions.

The inspector also reviewed Procedure No. N1-PSP-11, High Activity D/W Atmosphere Sampling and Analysis, Revision 0, dated 3-11-80. The procedure requires that containment atmosphere sampling be conducted at the H_2-O_2 Analyzer in the Turbine Building. The licensee stated that a sample could be obtained in about one hour in accordance with the above procedure.

The inspector further noted that no formal, approved procedures for obtaining a post accident reactor water sample or containment atmosphere sample existed prior to March 1980. (Note the procedure issue dates above.) This may be contrary to the Denton clarification requirement, however the licensee apparently made no commitment to issue interim procedures.

In subsequent discussions on November 18, 1980, the licensee stated that, assuming the reactor building was accessible the following

existing procedures would have been adequate to accomplish the necessary sampling:

NI-SP-11, Reactor Water Sampling with Reactor Cleanup System Isolated, Revision 1, dated December 3, 1979.

NI-SP-12, Procedure for Determining Drywell Equipment and Floor Drain Activities After Containment Auto Isolation, Revision 0, dated August 17, 1979.

NI-SP-8, Off Gas Sampling, Rev. 0, dated April 4, 1978.

The inspectors reviewed the above procedures and determined that the water sampling procedures SP-11 and 12, although cumbersome and time consuming, may have been acceptable for lower level events with the use of extensive planning efforts and surveys.

The gas sampling procedure, SP-8, was set up for taking condenser off gas samples with a portable sampling rig. For taking samples of containment atmosphere if the Drywell Continuous Air Monitor was inoperable or off scale. Extensive planning and work would probably have been required to set up the portable sampler for taking samples at the H^2-O^2 Sample Station. In summary, since NUREG 0578 or the clarification of the October 30, 1979 Denton letter did not discuss Reactor Water and Containment Air Sampling regarding the more probable accidents when access to the reactor building was feasible, and the licensee did not specifically commit to preparation of interim procedures, it is unclear what the "Category A" Item 2.1.8.a requirements were.

The inspector also verified that design review and a description of the major modification to install a reactor water sample sink in the Turbine Building were included in the licensee's submittal to NRR on 12-31-79.

This item is considered unresolved pending further discussion and clarification regarding the acceptability and timeliness of the licensee's action relating to the requirement for interim post accident sampling procedures by 1-1-80. (50-244/80-18-03)

e. Item 2.1.8.b - INTERIM METHOD TO QUANTIFY NOBLE GAS RELEASE DURING ACCIDENT CONDITIONS

Post Accident Gaseous Effluent Monitoring Capability

In the course of the inspection, the inspector reviewed the licensee's efforts to implement certain post-accident actions that were identified in NUREG-0578, TMI Lessons Learned Task Force Status Report and Short Term Recommendations. In the course of this review, the documents listed in Paragraph 4 of this report were examined.

The particular item reviewed was NUREG-0578, Section 2.1.8.b "Increased Range of Radiation Monitors", which discusses the necessity for nuclear power plants to have the capability to monitor and quantify high level releases of noble gas in the post-accident situation, and recommends the acquisition of equipment to establish an installed capability to monitor noble gases up to 10^5 uCi/cc (Xe-133).

Reference 4(1), directed all operating reactor licensee's to implement the actions contained in NUREG-0578 as soon as possible in accordance with the implementation schedule attached. The schedule defined two Implementation Categories as follows:

Category A: Implementation complete by January 1, 1980.

Category B: Implementation complete by January 1, 1981.

In the letter, Reference 4(1), Item 2.1.8.b was designated as Category "B" and referred to the requirement to complete installation of extended range monitors with an upper range capacity of 10^5 u/Ci/cc (Xe-133).

Reference 4(2) provided further clarification of the requirements, and in the case Item 2.1.8.b expanded the specification to include a Category "A" interim requirement ("provisional fix") to quantify noble gas releases as high as 10,000 Ci/sec until final installation of the extended range monitors as follows:

"CLARIFICATION

1. Radiological Noble Gas Effluent Monitors

   A. January 1, 1980 Requirements

      Until final implementation in January 1, 1981, all operating reactors must provide, by January 1, 1980, an interim method for quantifying high level releases which meets the requirements of Table 2.1.8.b.1. This method is to serve only as a provisional fix with the more detailed exact methods to follow. Methods are to be developed to quantify release rates of up to 10,000 Ci/sec for noble gases from all potential release points..."

The following was specified in Table 2.1.8.b.1:

"INTERIM PROCEDURES FOR QUANTIFYING HIGH LEVEL ACCIDENTAL RADIOACTIVITY RELEASES

- Licensees are to implement procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

- Examples of major elements of a highly radioactive effluent release special procedures (noble gas).

  - Preselected location to measure radiation from the exhaust air, e.g., exhaust duct or sample line.

  - Provide shielding to minimize background interference.

  - Use of an installed monitor (preferable) or dedicated portable monitor (acceptable) to measure the radiation.

  - Predetermined calculational method to convert the radiation level to radioactive effluent release rate."

Additionally, the 2.1.8.b Category "A" specification of Reference 4(2) required the licensee to describe and document the method the licensee intended to utilize as follows:

"The licensee shall provide the following information on his methods to quantify releases of radioactivity from the plant during an accident.

1. Noble Gas Effluents

   a. System/Method description including:

      i) Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique,

      ii) Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction,

iii) A description of method to be employed to facilitate access to radiation readings. For January 1, 1980, Control Room read-out is preferred; however, if impractical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

iv) Capability to obtain radiation readings at least every 15 minutes during an accident.

v) Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b. Procedures for conducting all aspects of the measurement/ analysis including:

i) Procedures for minimizing occupational exposures

ii) Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii) Procedures for dissemination of information.

iv) Procedures for calibration."

The licensee's response to the expanded requirement, Reference 4(5), dated December 31, 1979, stated the following:

"By January 1, 1980 the following provisional steps will be taken:

The existing in-line stack monitors are capable of detecting 50 Ci/sec. or approximately 0.55 uCi/cc (Xe-133) with normal ventilation flow of 180,000 ft.^{3}/minute. These monitors have read out and alarm capability in the main control room. Quantification of higher level noble gas releases will be provided by means of a portable gamma survey instrument. This instrument will be installed such that it will monitor a portion of the sample line to the existing stack monitors. This line comes from an isokinetic probe in the main stack.

Background radiation will be shielded by means of a lead cave built around the detector. The instrument has an upper limit of

at least 1000 R/hr. It will be calibrated with a Xe-133 source such that the reading can be related from R/hr. to uCi/sec stack release rate. Since all station effluents are discharged via the stack, the effluents monitored in this line are representative of the stack discharge. Until the Xe-133 calibration can be accomplished, the existing stack monitor calibration dependence data will be utilized to establish a calibration factor.

Readings on the interim monitor will be taken locally and the results verbally communicated to the main control room. This method would be used only in a case where the existing monitors were off-scale (high). Communication will be by means of a headset and will be taken approximately every fifteen minutes, when required.

The in-line monitors are powered from redundant AC power sources. These monitors are not presently powered from emergency sources. Power to the interim monitors will be from a DC battery source, capable of eight consecutive days of continuous readout."

Further correspondence from the licensee to the Director, Office of Nuclear Regulation on October 18, November 26, and December 19, 1979 indicating that some Category "A" requirements would not be implemented until after January 31, 1980, resulted in an Order to Show Cause being issued to Niagara Mohawk Power Corporation, Reference 4(6), on January 2, 1980, which discussed the basis of Category "A" implementation and stated:

"Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT the Licensee show cause, in the manner hereinafter provided, why it should not:

By January 31, 1980, implement all "Category A" requirements (except the requirement of 2.1.7.a of NUREG-0578) referred to in Part II of the Order, except those for which necessary equipment is shown, by appropriate and timely documentary justification to the Director, Office of NRR, to be unavailable, or in the alternative, place and maintain its facilities in a cold shutdown or refueling mode of operation. "Category A" requirements not implemented by January 31, 1980, owing to the unavailability of necessary equipment shall be implemented within 30 days of the date such equipment becomes available but no later than June 1, 1980."

Reference 4(7), dated January 22, 1980, provided the licensee's answer to the Order to Show Cause, and identified the status of Item 2.1.8.b in the attached Exhibit "A" as,

"REQUIREMENT	TITLE	IMPLEMENTATION CATEGORY (1)	DATE IMPLEMENTATION COMPLETE
2.1.8.b	High Range Radiation Monitors - Effluents - Procedures	A	December 31, 1979

(1) Category A: Implementation Complete by January 1, 1980"

On October 8, 1980, during review of this area the inspector noted that the only action that had been performed pertaining Item 2.1.8.b was the installation of a single lead brick (approximately 7" x 4" x 2") on the main stack sampling line; with a small hole, about 1" in diameter and 1" in depth, bored through the face of the brick to the stack sampling line.

Discussions with the Superintendent, Radiochemistry and Radiation Protection revealed that the lead brick was in actuality the licensee's only action taken to quantify high level noble gas releases. The individual indicated that it was the licensee's intention to dispatch a person with a teletector (an instrument having a range of 1000 R/hr) to the stack area, and insert the probe into the bored hole provided for this purpose in the event that normal stack monitoring equipment went off-scale in the emergency situation; and thereby obtain a radiation reading to relate to noble gas release.

To this end, the Superintendent, Radiochemistry and Radiation Protection produced data by which the licensee determined that the conversion factor (for a teletector calibrated with Co-60) that relates noble gas concentration to dose rate was 0.5 uCi/cc/mR/hr. Upon examination of the data it was found that the method employed by the licensee was subject to many errors and was in actuality not based on increasing concentration of noble gas versus dose rate, but rather on a questionable technique involving the decay of normal gaseous activity versus declining dose rate.

In further discussions, the Superintendent, Radiochemistry and Radiation Protection stated that it was the licensee's intention that this activity would be performed in accordance with EPP-6, "Implant Emergency Surveys," Rev. 2, dated October 2, 1980, which the licensee considered to meet the procedural requirements specified for Category "A" Item 2.1.8.b. Subsequently, the inspector reviewed this procedure to determine if it provided adequate instruction for performing the activities required for this item. The following was noted:

a) The purpose of the above procedure was only to outline actions to be taken by plant personnel when making investigatory surveys during a radiological emergency.

b) The procedure did not adhere to the specifications of Category "A" Item 2.1.8.b in that,

- it did not provide for the capability to take radiation readings for this item every 15 minutes;
- it did not provide instructions for minimizing occupational exposure, other than instructions to exit the area if levels exceeding 10 R/hour (2.5 R/hour on offshifts), or the method to be used to facilitate access to the specific monitoring location;
- it did not provide calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration the radionuclide spectrum distribution as a function of time after shutdown;
- it did not provide instructions detailing the calibration technique for the instrument specific to the function it served;
- it did not describe the location of the monitoring point;
- it did not provide for installed or dedicated portable instruments to be used specifically for this item, and
- it did not provide for methods to be used to minimize background interference.

In summary, examination of this item revealed the following pertaining to the requirements of Item 2.1.8.b as specified in Reference 4(2), and as represented by the licensee in Reference 4(5):

1. No portable gamma survey instrument was installed on the indicated sampling line nor was there an instrument dedicated specifically to this particular monitoring activity;

2. No lead cave or other shielding sufficient to minimize background interference existed on the sampling line, however a small piece of lead brick was attached to the line that was used by the licensee to correlate concentration to dose rate as provided by a teletector (normal survey instrument) calibrated to Co-60. At the time of this finding there had been no indication of action initiated to construct a lead cave around the detector;

3. While teletectors do have a range capability of 1000 R/hr, the licensee'sfacilities are unable to provide any instrument calibration beyond ~27 R/hr (Co-60);

4. A calibration of the teletector with Xe-133 was never performed, however, the licensee did attempt to correlate dose rate to concentration by a questionable method that was later found to under estimate Xenon concentration by a factor of 40. Also, at the time of this finding there had been no action ever initiated to perform the calibration with a Xenon-133 source;

5. Although verbal communication to the Control Room was stated to be via a headset, there was no system in place or planned to provide this ability;

6. The teletector was correctly described as being powered by a DC source, and could be capable the eight days of continuous operation;

7. Instrument sensitivity energy dependence and calibration information was not provided;

8. A method to facilitate access to the location where the radiation readings were to be taken was never developed;

9. Procedures of any sort relating to the implementation of this requirement were never developed;

10. A predetermined calculational method to convert the radiation level to radioactive effluent release rate was not developed.

In addition, it was further determined that personnel had not received any training regarding the performance of activities relating to Item 2.1.8.b.

In order to effect immediate resolution of this item, the licensee committed to perform corrective actions in this area as described in a letter from B. H. Grier, Director, NRC - Region I, to T. E. Lempges, Vice President, Niagara Mohawk, dated October 17, 1980 (IAL 80-40).

Based on the above findings it was found that the licensee had failed to comply with the Order to Show Cause of January 2, 1980, as it related to the completion of Category "A" Item 2.1.8.b described in the October 30, 1979, letter from Director, Nuclear Reactor Regulation, Reference 4 (2), in that the stated requirements were not completed by January 31, 1980. (50-220/80-04)

Other aspects of this particular item are further discussed in IE Reports 50-220/80-11 and 50-220/80-17.

f. Item 2.1.8.c - IMPROVED IN PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

Requirements of the Denton Clarification Letter of October 30, 1979

Monitoring must include portable instruments

Capability to accurately detect presence of iodine in region of interest following an accident.

To be accomplished with a Single Channel Analyzer calibrated to the 365 kev of I 131.

Care to be taken to prevent saturation of counting system.

Licensee Commitments

NMPC Letter-Dise to Denton dated December 31, 1979

Portable instruments will be used

Samples will be purged to remove entrapped noble gases and analyzed with GELI Spectrometer.

Inspection Findings

This licensee appears to have completed the requirements of this item. However, the licensee stated that he intends to sample for iodine using a portable Staplex High Volume Air Sampler, in accordance with Radiation Protection Procedure No. S-RTP-70, "Operation and Calibration of Staplex High Volume Air Sampler," Revision 1 dated 8-10-78. The Health Physics Appraisal Team Inspection, conducted September 29-October 10, 1980, has indicated that the use of a high volume sampler in conjuction with the type of charcoal cartridge utilized by the licensee is technically inadequate to determine radioiodine concentration. The high linear velocity through the charcoal cartridge does not provide sufficient residence time in the charcoal region to assure radioiodine absorption. The licensee's procedure assumes 100% collection efficiency without any justification, bases or evaluation to support this capability.

The GELI Spectrometer is calibrated with a mixed isotope source that does not contain iodine. The mixed isotope source consists of Cd-109, Co-57, Ce-139, Hg-203, Sn-113, Cs-137, Y-88 and Co-60.

The licensee does not have formal procedures for purging samples to remove noble gases.

This item is considered unresolved pending further discussions and clarification of the acceptability of the licensee's action relating to the use of Hi Volume Samplers, GELI Spectrometer, and lack of found procedures for purging samples. (50-220/80-18-05)

g. Item 2.2.1.A SHIFT SUPERVISOR RESPONSIBILITY

The Requirements of the Denton Clarification of October 30, 1979
Clarification of responsible individuals defined

Licensee Commitments

NMPC Letter- Bartlett to Denton dated November 26, 1979

NMPC agreed to comply with this item

Inspection Findings

The licensee appears to have satisfied this requirement based on the inspector's review of Administrative Procedure No. APN-2A, Conduct of Operations and Composition and Responsibilities of Station or Unit Organization, Revision 2, dated 12-18-79, and an internal memo from the Vice President of Electric Production to the General Superintendent of Nuclear (site manager) dated 10-9-79 emphasizing the responsibilities and authority of the Shift Supervisor.

A review of training records show that the shift supervisors received training in this area during February 1980. Staff members, who hold senior reactor operator licenses were required to review APN-2A as "routed reading."

With regard to reduction of Shift Supervisor administrative duties, the licensee stated that the STA has been granted the authority to approve radiation work permits, welding and burning permits, meal tickets, and overtime slips, which relieves the shift supervisor of these administrative duties.

h. Item 2.2.1.b - SHIFT TECHNICAL ADVISOR

Requirements of the Denton Clarification Letter of October 30, 1979

(1) Provide an STA on shift and available within ten minutes.

(2) The STA shall have a BA Degree or Equivalent.

Licensee Commitments

NMPC Letter- Bartlett to Denton dated November 26, 1979

NMPC Letter- Rhode to Denton dated December 20, 1979

The licensee agreed to augment the present staff with an Ass't. Shift Supervisor in accordance with NUREG 0578 and the Denton Clarification letter of 10-30-79.

Inspection Findings

The licensee appears to have satisfied this requirement and his commitments, based on the inspector's review of the shift manning schedule and review of the resumes of the Shift Technical Advisors.

All Shift Technical Advisors have at least a BA Degree and plant specific training is currently in progress. Schedules indicate that they are available within ten minutes.

i. Item 2.2.1.c SHIFT RELIEF AND TURNOVER PROCEDURES

Requirements of the Denton Clarification Letter of October 30, 1979

(1) Provide a checklist for oncoming and offcoming control room operators and oncoming shift supervisor to complete and sign. The following items, as a minimum shall be included:

(a) Assurance that critical plant parameters are within allowable limits.

(b) Assurance of availability and proper alignment of all essential systems through a check of the control console.

(c) Identification of systems and components in a degraded mode of operation permitted by the technical specifications.

(2) Checklists or logs provided for completion by offgoing and oncoming Auxiliary Operators and technicians.

(3) Establish a system to evaluate the effectiveness of the shift and relief turnover procedure.

Licensee Commitments

NMPC Letter- Bartlett to Denton dated November 26, 1979

NMPC agreed to comply with this item.

Inspection Findings

The licensee appears to have satisfied the requirements of this item.

The inspector verified that the shift turnover checklist meets the requirements of paragraphs 1 and 2 above. The licensee stated that his method for satisfying paragraph 3 consisted of the Operations Supervisor review of the completed checklist, QA audits of the implementation of the checklist, and a review of station occurrence report records for problems due to misaligned safety systems.

j. Item 2.2.2.A - CONTROL ROOM ACCESS

Requirements of the Denton Clarification of October 30, 1979

None

Licensee Commitment

NMPC Letter- Bartlett to Denton dated November 26, 1979

NMPC committed to comply with this item

Inspection Findings

The licensee appears to have satisfied the requirements of NUREG 0578 and his commitments of 11-26-79.

The inspector verified that Administrative Procedure No. APN 2A, Conduct of Operations and Composition and Responsibilities of Station or Unit Organization Rev. 2 dated 12-18-79, includes procedures for limiting control room access and established the control of all station operations under the Shift Supervisor on duty.

k. Item 2.2.2.b TECHNICAL SUPPORT CENTER

Requirements of Denton Clarification of October 30, 1979

(1) Establish a Technical Support Center and provide a complete description.

(2) Provide plans and procedures for engineering/management support and staffing of TSC.

(3) Install dedicated communications between the TSC and the Control Room, Near Site Emergency Center, and the NRC.

(4) Provide monitoring (portable or permanent) for both direct radiation and airborne contaminants. Designate action levels for protective measures.

(5) Assimilate or ensure access to technical data--licensee best effort to display plant parameters

(6) Develop procedures for performing accident assessment from the Control Room if TSC should become uninhabitable.

(7) Submit to NRC longer range plan for upgrading TSC.

Licensee Commitments

NMPC Letter- Bartlett to Denton dated November 26, 1979

NMPC Letter- Dise to Denton dated December 31, 1979

NMPC agreed to comply with this item

Inspection Findings

The licensee appears to satisfy the requirements of NUREG 0578 and his commitment of 11-26-79.

This conclusion is based on the inspectors review of Administrative Procedure No. APN-2B, Special or Emergency Operations, Revision 0, dated 12-19-79, the licensee's submittal to NRR on 12-31-79, and an inspection of the facilities in the Technical Support Center.

1. Item 2.2.2.c ONSITE OPERATIONAL SUPPORT CENTER
Requirements of the Denton Clarification of October 30, 1979
None provided

Licensee Commitments

NMPC Letter- Bartlett to Denton dated November 26, 1979

NMPC committed to comply with this item

Inspection Findings

The licensee appears to satisfy the requirements of NUREG 0578 and his commitments of 11-26-79.

This conclusion is based on the inspector's review of Administrative Procedure APN-2B, Special or Emergency Operations, Revision 0, dated 12-19-79 and inspection of the Onsite Operational Support Center.

5. Unresolved Items

An item about which more information is required to determine acceptability is considered unresolved. Five paragraphs contain unresolved items (4.a, 4.b, 4.d, 4.f.

6. Exit Interview

An exit interview was conducted with licensee representative (denoted in Paragraph 1) on November 12, 1980. The findings of the inspection were presented.