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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/301 PLAINFIELD ROAD, SYRACUSE, N.Y. 13212/TELEPHONE (315) 474-1511

December 2, 1988
NMP1L 0330

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re:	Nine Mile Point Unit 1 Docket No. 50-220 DPR-63	Nine Mile Point Unit 2 Docket No. 50-410 NPF-69

Gentlemen:

Your letter dated September 15, 1988, transmitted Inspection Report No. 50-220/88-17 and 50-410/88-17, which required Niagara Mohawk to respond to a Notice of Violation and requested Niagara Mohawk to provide a summary of our planned corrective actions regarding several items. Our letter dated October 17, 1988, responded to the Notice of Violation and indicated that our summary of planned corrective actions regarding the other items would be provided by November 30, 1988. This letter provides that information.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

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Niagara Mohawk Power Corporation

Nine Mile Point Unit 1	Nine Mile Point Unit 2
Docket No. 50-220	Docket No. 50-410
DPR-63	NPF-69

Response to Request for Planned Corrective Actions Contained in Inspection Report No. 50-220/88-17 and 50-410/88-17

1.0 Inservice Inspection Program Stop Work Order at Nine Mile Point Unit 1.

1.1 NRC Finding (Sections 1.1.f and 16 of the subject NRC Inspection Report)

On August 16, the licensee Quality Assurance Department issued Stop Work Order 88-004, placing a hold on Inservice Inspection (ISI) examinations being performed by the ISI program contractor, Nuclear Energy Services (NES). The Stop Work Order cites a breakdown of the required 10 CFR 50, Appendix B, quality controls necessary to properly complete the ISI program. NES is currently developing an action plan for the licensee's review, prior to lifting of the Stop Work Order. (Section 1.1.f)

The Unit 1 ISI Stop Work Order issued by QA shows that organization can be critical of work that is ongoing. However, the oversight provided by Engineering that led to the April 1988 violation and civil penalty regarding ISI does not appear to have improved, as evidenced by the Stop Work Order. (Section 16)

1.2 NMPC Response

Engineering management oversight of ISI contractor activities has improved. The ISI organization, which was part of the Nuclear Generation Department, was transferred to the Nuclear Engineering & Licensing Department. As part of Engineering's overall control, the QA organization assists by providing independent data review and surveillance of the ISI contractor. This approach was outlined in our April 13, 1988 response to the March 14, 1988 Notice of Violation. Further explanation of our approach was given to the NRC in Region I Headquarters on October 27, 1988. The Stop Work Order (SWO) 88-004 is evidence of the effectiveness of our approach.

The quality-related deficiencies which led to SWO 88-004 and the October 5, 1988 misidentification of a weld were identified by our QA NDE oversight efforts. This oversight by QA is consistent with planned corrective actions in response to the Notice of Violation (March 14, 1988) and the NMPC QA Program. The deficiencies were identified as a result of effective program controls, and the timely and appropriate corrective actions taken are indicative of an improvement in our management oversight of contractor activities. The stop work actions taken in response to these detailed deficiencies indicate that Niagara Mohawk is closely monitoring NES and has a high concern for quality relative to production.

Nuclear Energy Services (NES) has been contracted by Niagara Mohawk Power Corporation (NMPC) to perform inservice inspection (ISI) examinations at Nine Mile Point Unit 1. On August 16, 1988, the NMPC Nuclear Quality Assurance Operations Organization issued SWO 88-004 directed to NES. The SWO required NES to stop all ASME Section XI Program Plan required examinations for Nine Mile Point Unit 1. On the same day, the NMPC Contract Administrator assigned to the NES contract informed the NES Project Manager that work would not be allowed to resume until directed by NMPC Engineering. The SWO was initiated after NMPC QA identified an increasingly negative trend in NES deficiencies based upon their review of NES documentation.

In response to the SWO, NES was required to develop a corrective action plan for NMPC Engineering and QA approval. NES submitted a report on August 26, 1988, outlining their action plan and concluding that the activities leading to the SWO were a result of the failure of NES to implement their approved Quality Assurance Program. The report included a description of the corrective actions, actions taken to identify other deficient items, a root cause determination, and actions to prevent recurrence.

NMPC accepted the NES action plan and initiated reviews to verify the implementation of the corrective actions. On September 14, 1988, when the critical corrective actions necessary for work to resume were completed, the SWO was lifted. Certain corrective actions required that work commence for implementation, such as re-examinations.

The following actions were carried out and verified as part of the Stop Work Order:

1. NES hired an independent consultant to review their Quality Assurance Program. This review concluded that NES has a comprehensive quality program that adequately addresses regulatory and NMPC requirements. In addition, the independent review concluded that NES QA program has been generally implemented at Nine Mile Point Unit 1. A weakness was noted in formalized job specific training. This weakness was corrected through retraining as part of the Stop Work Order recovery, and the additional assignment of an NES Site QA representative will assure documented training to future document revisions.

2. The deficiency documents referenced in the Stop Work Order were reviewed to assure that each was adequately responded to and that the critical corrective actions required prior to lifting of the Stop Work Order were implemented.

   Corrective actions required NES to re-examine all welds inspected by the examiners involved and to re-examine a sample of welds inspected by other examiners. The NES root cause analysis determined that there may have been a problem with NES calibration techniques. Therefore, NES revisited approximately 125 calibration data sheets to assure the adequacy of their calibration program.

3. NES appropriately responded to Niagara Mohawk document review sheets.

4. NES provided a full-time QA Site representative.

5. The NES QA Manager provided a QA indoctrination training for NES Site personnel.

6. The NES QA representative followed a training matrix and verified required training before work assignments were made.

7. Training and procedural compliance has been completed.

8. NES has established a commitment list to track open items.

9. NES has initiated a weekly surveillance program to monitor examination performance. A procedure delineating this surveillance program has been developed by NES and approved by NMPC.

10. NES has assigned a data controller to assist in processing the examination data reports. NES has also developed a data control procedure which has been approved by Niagara Mohawk.

Other Niagara Mohawk actions included the assignment of an Engineering Manager to oversee the Stop Work Order Corrective Action process to assure that the appropriate corrective actions were initiated and implemented prior to Engineering's acceptance of future work.

On October 5, 1988, our QA NDE oversight efforts identified a second case where a different NES examiner had inspected the wrong weld. Engineering directed NES to stop examinations until the issue was reviewed and resolved. NES determined that the root cause was human error coupled with the lack of a formalized system for component identification. Corrective action included the following:

1. NES verified that the examinations performed by the examiner involved were correct.

2. An independent verification of component identifications is now required.

3. The examiner involved was again instructed in weld/component identification.

On October 15, 1988, after verification of the above corrective actions, NES was directed to resume work.

2.0 General Condition in Unit 1 Including Poor Housekeeping and Cracking of the Reactor Building Wall

2.1 NRC Finding (Portion of Section 3.1 of the subject NRC Inspection Report regarding housekeeping.)

- The inspectors noted poor housekeeping practices inside the drywell. Large amounts of material such as scaffolding, grinding tools, and drills, were piled up in areas of the drywell. Also, empty cans of WD-40, tie wraps, and other small items were simply left lying around the perimeter of the drywell. The licensee has been informed that further efforts are needed to clean up this area. Graffiti was written on most walls in the drywell. This generates two ALARA concerns. One, the people are taking time to write graffiti, and secondly, people will have to take time to clean it up; all of this in a high radiation area. This concern will be closely monitored as the licensee proceeds toward plant restart...

- From their tours of the refueling floor area, the inspectors have noted that the spent fuel pit has quite a few items stored in it other than spent fuel. Items stored range from metal boxes with pipe sections in them and other objects sitting directly on top of the spent fuel racks, to a large number of metal objects suspended around the sides of the spent fuel pit from the spent fuel pool's railing. As the spent fuel pit is a seismic structure, the inspectors questioned the licensee as to the advisability of imposing additional loads on the spent fuel racks (loads being those directly imposed on the racks as well as those imposed, potentially, from suspended objects falling on top of the racks). The licensee responded that this has been examined by them within the last year and that their analysis would be presented to the inspectors. This matter will be updated in a subsequent report pending review of the licensee provided information.

2.2 Niagara Mohawk Response to Housekeeping Concerns

We concur with the observations noted during your inspection tours. Although we believe there is still need for improvement, the overall general housekeeping at NMP Unit 1 is improving. The Station Superintendent has made this item one of his priorities to resolve.

The lack of a comprehensive housekeeping plan has been identified as a contributing root cause of this problem. Therefore, a comprehensive housekeeping plan to address all aspects of this issue is being developed. The comprehensive plan will include standards and criteria for plant cleanliness and storage, areas of responsibility, details of inspection tours, and actions to be taken when areas need improvement. Also, the plan will include increased first level supervision involvement and cross disciplinary inspections. Further, prior to restart, a special procedure for system and area walkdowns will be implemented.

The general problem of graffiti has been addressed with an aggressive cleanup and painting campaign. A plan to address the graffiti in the drywell is being developed and will address ALARA and special drywell painting concerns. The root causes of the graffiti are difficult to determine at this time. Graffiti will be monitored during routine and special plant tours. At present, it does not appear to be a wide spread problem because there has not been a reoccurrence of graffiti in areas which have been cleaned and painted. If further incidents are discovered, every effort will be made to identify the responsible individuals and appropriate corrective actions will be taken.

The inspector was also specifically concerned with the large amount of excess materials stored in the spent fuel pool with respect to any potential impact this excess loading may have on the seismic qualification of the spent fuel storage racks. As discussed in a more recent Inspection Report (88-18), a 10 CFR 21 evaluation concluded that the condition is not reportable.

Further plans are being developed to remove most of the excess materials from the spent fuel pool. Inspection Report 88-18 indicated that the NRC has no further question on this matter, but will continue to monitor action related to the storage and removal of items from the spent fuel pool.

2.3 NRC Finding (Portion of Section 3.1 of the subject NRC Inspection Report regarding Cracking of Reactor building Wall.)

...from touring the Reactor Building at the 237' level, near the hydraulic control units, it is evident that more attention needs to be paid to housekeeping in this area. Also, on exiting the Southwest Corner Room at the 237' level, large cracks (up to five feet in length) were observed in the concrete near several large piping penetrations. The licensee was requested to determine if the wall is load bearing (or for biological shielding only) and to evaluate the condition of the observed cracks. This item will be reviewed in a subsequent report.

2.4 Niagara Mohawk Response to Cracking of Reactor Building Wall Concern

INTRODUCTION

Niagara Mohawk Engineering has observed the noted cracks in the Reactor/Turbine Building Walls, made assessments based on these observations and construction details, and developed recommended actions for resolution of these concerns. A complete report of the assessments, observations and recommended actions is available for review in Nuclear Engineering and Licensing. A summary of the report is provided below.

PERTINENT CONSTRUCTION DETAILS

In the vicinity of these penetrations the wall actually consists of the three feet (3'-0") thick concrete wall of the Reactor Building and the one foot-five inch (1'-5") thick concrete wall of the Turbine Building.

The bottom of these walls is located at elevation 237'-0" on concrete constructed around the torus. There is a one inch (1") wide space between the Reactor Building and Turbine Building walls for the entire height and length at the turbine building wall. This space is filled with resilient material.

The piping penetrations noted are associated with twelve inch (12") diameter core spray piping (System #81) and fourteen inch (14") (nominal) penetrations; numbers 20 and 21.

These walls are not used as biological shields, they are load bearing only.

OBSERVATION/DESCRIPTION OF CONCRETE CRACKS

Reactor Building Wall

The Reactor Building Wall can only be observed from the north side since the south side is bounded by the resilient material and the Turbine Building Wall.

There were four (4) cracks visible on this wall. These cracks were less than one-sixteenth of an inch (1/16") wide. Two cracks were approximately fifteen to eighteen inches (15"-18") long and started at the lower east side of the west penetration and were orientated downward and to the east. The other two cracks were approximately twenty to twenty-eight inches (20"-28") long and started at the upper west side of the west penetration. The trend of these cracks is upward and horizontally to the west.

Turbine Building Wall

The Turbine Building Wall can only be observed from the south side since the north side is bounded by the resilient material and the Reactor Building Wall.

There were approximately twelve (12) independent cracks visible on this wall, the widest of which was approximately three-sixteenths of an inch (3/16"). The lengths of the cracks ranged from approximately seventeen to sixty-four inches (17"-64"). Four (4) cracks appeared to start at the core spray piping wall penetrations. All cracks had an upward-westerly orientation at approximately 45 degrees. For the most part, the cracks are located below a diagonal of the wall oriented between the upper-west and lower-east corners of this wall.

In addition, it was observed that two reinforcing bars were damaged in the west penetration.

ASSESSMENT OF CONCRETE CRACKS

Reactor Building Wall

The bottom of the wall is structurally connected to the large mass of concrete surrounding the torus while the top of the wall is structurally connected to the floor at elevation 249'-0" in the Reactor Building. Restraining forces at the top and bottom of the wall are believed to be the cause of the cracks in the wall. These forces could have resulted from shrinkage of the concrete wall after placement. Restraining forces also could have developed as a result of differing strain rates occurring at the top and bottom of the wall; the strain rates being due to thermal fluctuations of plant operation and the different rates at which the torus concrete and elevation 249'-0" concrete expands/contracts with such fluctuations.

Turbine Building Wall

The bottom of this wall is also connected to the large mass of concrete surrounding the torus. However, the top of the wall is connected to floor elevation 250'-0" in the Turbine Building. It is believed that the causes of the cracks in this wall are the same as those outlined for the Reactor Building Wall. The cracks in this wall are believed to be more pronounced for the following reasons:

a) The Reactor Building wall is more than twice as thick as the Turbine Building wall and has more than two and one-half times as much reinforcing steel as the Turbine Building Wall. Therefore, the Reactor Building Wall can accommodate restraining forces with less evidence of cracking.

b) The difference in strain rates between the bottom and top of the Turbine Building Wall is probably greater than that of the Reactor Building Wall. The basis for this is that the top of the Turbine Building Wall is not connected to any part of the Reactor Building and thus often may not reach thermal equilibrium with the Reactor Building (i.e. the Torus Concrete). Therefore, the Turbine Building Wall is subjected to a greater differential in strain rates/sustained strains and restraining forces between the top and the bottom.

c) If the cracks developed shortly after construction due to shrinkage, drilling and chipping to widen the penetrations in the Turbine Building Wall to achieve alignment with the Reactor Building wall penetrations may have aggravated the cracks of the Turbine Building Wall.

PROPOSED ACTIONS

1. Reactor Building Wall

The cracks in this wall are of nominal length, width and quantity. In relation to the wall's design and intrinsic capacity/strength, the cracks are judged to be insignificant and do not diminish the structural capacity or design function of the wall. A calculation addressing stress and seismic capacity was performed to substantiate this conclusion.

2. Turbine Building Wall

The total length of the Turbine Building Wall on Column Row J is approximately one hundred-sixty feet (160'). In addition, there are reinforced concrete columns along the wall at twenty foot (20') intervals. Therefore, it is unlikely that the cracks observed in this wall between Columns 4 and 5 will affect the overall performance and capacity of the wall. Therefore, the overall performance of this wall does not need to be re-assessed.

However, an assessment has been made of the effect of the cracks on the local capacity/performance of the wall between Columns 4 and 5. The analysis conservatively reflected the cracks and the damaged reinforcing and indicated that the local capacity of the wall is sufficient to accommodate design loads.

3.0 Testing Overcurrent Devices

3.1 NRC Finding (Section 5.1.b of the subject NRC Inspection Report)

The inspector reviewed completed copies of procedure N1-EMP-GEN-R151 discussed in Section 2.1.d above [Electrical Preventative Maintenance Procedure for testing overcurrent devices for GE AK type circuit breakers]. Nine procedures were reviewed for safety-related breakers. Out of these nine, there were numerous cases where the "as-found" overcurrent device settings were not within the given ranges prior to breaker rework. In five of the nine cases, "as-left" overcurrent device settings were not in the prescribed ranges. The data sheets reviewed have spaces for maximum and minimum allowable settings for the long, short and instantaneous time delay trip amperage setpoints. In all cases, these were not filled out. The only information given for these settings was the percent over rated current and the specific current value. The data for each breaker is obtained from a data base, which is based on the selective tripping design of the specific load center.

The inspector requested that the licensee review the program for determining the required setpoints and review the previously performed procedures to determine if any corrective action might need to be taken to ensure that these breakers are able to function in their designed load capacities. This item is unresolved (50-220/88-17-03).

3.2 Niagara Mohawk Response

The procedure will be revised to require that if the "As Found" conditions do not fall within the tolerances, the Electrical Maintenance Supervisor will be informed. This supervisor will determine what actions are to be taken.

As indicated in five cases of the nine reviewed, the "As Left" timing points did not fall within the required data values. When the timing points were found outside the required values, separate Work Requests were generated to remove the breakers from service and replace the trip devices.

The final concern dealt with the fact that on the Data Sheet there are columns for maximum and minimum pick up for Long, Short, and Instantaneous time delay set points and no data was entered there. This information was not available on the set point data base at the time of the inspection. Since the inspection, Engineering has provided the necessary information to complete the test in accordance with the procedure.

4.0 Designated Alternatives for the Station Managers without knowledge of a Senior Reactor Operator

4.1 NRC Finding (Section 12 of the subject NRC Inspection Report)

...The inspector reviewed the appropriate standards for selection and training of nuclear power plant personnel, ANSI N18.1-1971 (Unit 1) and ANSI 3.1-1978 (Unit 2), to which the licensee is committed to verify that the persons selected for new positions met the minimum qualification requirements. The inspector noted that Mr. J. Willis, the Station Superintendent, and Mr. Dahlberg, Unit 1 Superintendent, do not meet the requirements and operating experience equivalent to that normally required to be eligible for a Senior Reactor Operator's (SRO) license.

Designation of a principal alternate who meets the SRO or equivalent requirement is allowed by the ANSI Standards for both individuals. The individuals designated for Mr. Willis are Mr. R. Abbott, Unit 2 Superintendent, and Mr. R. Randall, Unit 1 Operations Superintendent. Mr. Dahlberg's principal alternate is also Mr. R. Randall.

On August 23, Mr. C. Mangan, Senior Vice President, committed to formalize the chain of command at both units. He also committed to develop a plan such that Mr. Willis and Mr. Dahlberg will receive additional training. This item will be monitored by the inspectors in future inspection periods.

4.2 Niagara Mohawk Response

Niagara Mohawk disagrees with the finding stated above for the following reasons:

A. ANSI N18.1-1971 and ANSI 3.1-1978, Paragraph 4.2.1 (of each) requires that the Plant Manager (Plant Superintendent) "...shall have ten years of power plant experience, of which three years shall be nuclear power plant experience".

B. The Plant Manager (Plant Superintendent) at Nine Mile One is Mr. K.A. Dahlberg. Mr. Dahlberg has 16 years of power plant experience of which 15 years are nuclear power plant experience. Mr. Dahlberg has not received the formal training normally required for a SRO License. The principal alternate for Mr. Dahlberg is Mr. R. Randall. Mr. Randall has fifteen years of power plant experience, all of which is nuclear power plant experience, and he holds a SRO License for Unit 1. Thus, the requirements of the aforementioned ANSI documents are met in all respects.

C. Mr. J.L. Willis is assigned as the General Superintendent of Nuclear Generation (Site Director). The requirements of Paragraph 4.2.1 do not apply to this position, because the position of General Superintendent is senior to the Plant Manager. Nevertheless, Mr. Willis has 33 years of power plant experience of which 30 are nuclear power plant experience. Mr. Willis has previously received training normally required for a SRO License at a PWR facility.

It is planned that both Mr. Willis and Mr. Dahlberg will receive training in the future on plant specific items, including simulator familiarization. It is not intended that either receive all of the training necessary for a SRO License and such is not required.

5.0 Safety Evaluation Reviews made pursuant to 10 CFR 50.59

5.1 NRC Findings (Section 14 of the subject NRC Inspection Report)

10 CFR 50.59 permits the licensee to (a) make changes to a facility as described in the safety analysis report, (b) make changes to a procedure as described in the safety analysis report, and (c) conduct tests or experiments not described in the safety analysis report without prior Commission approval, unless the proposed change, test or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question. The inspector reviewed ten evaluations performed by the licensee pursuant to 10 CFR 50.59. The evaluations covered a period from October 31, 1986 through June 24, 1988. The inspector also reviewed the licensee's procedure for performing evaluations pursuant to 10 CFR 50.59, NT-100.B, Rev. 5, "Preparing and Control of Safety Evaluations." The inspector noted during the review of

the evaluations that the quality of the evaluations performed appeared to get progressively better with time (i.e., the later packages were better than the earlier ones). The inspector had the following specific comments as a result of the review:

a. The technical justification for SE 87-053 was incomplete. This change involved a revision to the service water (SW) pump trip setpoint for low discharge flow from 2500 gpm to 1000 gpm with a 10-second time delay. The vendor recommended a minimum allowable flow of 2300 gpm to protect the pump. The change to the 1000 gpm setpoint was made to prevent the pump from tripping during flow transients during the startup period.

   The safety evaluation referenced calculation 12177-CS-SWP*01 for verification that a setpoint of 1000 gpm was satisfactory. The inspector reviewed the referenced calculation. The calculation evaluated the allowable setpoint for the trip considering such things as instrument drift and uncertainties. It did not evaluate the effect of a 1000 gpm flow on the SW pump. The calculation contained a statement that the only way the pump would experience a reduction of flow to less than 2600 gpm would be if the discharge valve downstream of the pump were to be closed. In that event, the flow would be below 1000 gpm and a trip would be initiated. The discharge valve is a motor-operated valve which is designed to be either fully opened or fully closed. However, the inlet valve upstream of the pump is a manually operated butterfly valve. This valve could be inadvertently left in a partially open position permitting a flow of between 1000 and 2300 gpm. The licensee has committed to review the change to the trip setpoint and provide a complete justification for the change pursuant to 10 CFR 50.59.

b. The licensee's procedure for the review of changes, tests, and procedures pursuant to 10 CFR 50.59, NT-100.B, "Preparation and Control of Safety Evaluations," requires an environmental evaluation to be performed to determine whether an unreviewed environmental question exists. The environmental review is not required to meet the requirements of 10 CFR 50.59; however, the review is specified in the Environmental Protection Plan (EPP), Appendix B, of the facility operating license. A number of the packages reviewed did not provide evidence that an environmental review was performed. The inspector obtained a commitment from the licensee that a review of all safety evaluations performed under 10 CFR 50.59 would be conducted to ensure that environmental impact reviews are performed and documented.

c. The licensee is required under 10 CFR 50.59 to submit annually a report of any changes, tests, and experiments. The licensee submitted its report on January 21, 1988, to cover the period from October 31, 1986 to October 31, 1987. The report listed six modifications and did not indicate any other changes for this period. A review of the evaluations for this period indicated that the list was substantially incomplete, even

allowing for the modifications that had been evaluated and which were still not completed. The licensee indicated this was an oversight and committed to update the list through the end of August 1988 and submit it by October 31, 1988.

5.2 Niagara Mohawk's Response

5.2.1 Item (a)

Nuclear Engineering and Licensing has revised Nine Mile Point Unit 2 Safety Evaluation Report (SER) No. 87-053 to include additional justification for the service water pump trip setpoint change. This justification is contained in Revision 1 to SER 87-053 and concludes that the change does not constitute an unreviewed safety question. Revision 1 to SER 87-053 was approved by the Site Operations Review Committee.

Regarding the inspector's concern that a valve could be inadvertently left in a partially open position permitting a flow between 1000 and 2300 gpm, operating procedure N2-OP-11 contains adequate precautionary measures to prevent this from occurring as follows:

- Section D.9 Pg. #3 indicates, "Do not allow service water pump to operate at less than 2500 gpm flow for longer than 10 seconds or pump damage may result."

- Section E.1.a Pg. #4 indicates, "Verify system valve line-up is in accordance with Table I for all portions of the system to be operated." Also, Table I shows valve line-up positions to be "LOCKED OPEN" for each suction valve described.

- Section E.2.e Pg. #8 indicates, "Ensure that sufficient system flow exists to allow at least 2500 gpm for each pump that will be running."

5.2.2 Item (b)

Shortly after the inspection, Niagara Mohawk began reviewing all safety evaluations performed under 10 CFR 50.59 to ensure that environmental impact reviews were performed and documented. To date, of the approximately 700 evaluations identified as requiring environmental impact evaluations, approximately 350 environmental impact evaluations have been performed and documented. The remaining environmental impact evaluation reviews are scheduled to be completed by February 28, 1989.

5.2.3 Item (c)

As committed to during the inspection, Niagara Mohawk has updated the list of changes effecting the FSAR made to the plant between October 31, 1986 and August 31, 1988. This list was submitted on October 26, 1988.