U.S. NUCLEAR REGULATORY COMMISSION
REGION I

REPORT NOS.	50-220/92-26 50-410/92-30
DOCKET NOS.	50-220 50-410
LICENSE NOS.	DPR-63 NPF-69
LICENSEE:.	Niagara Mohawk Power Corporation
FACILITY NAME:	Nine Mile Point Nuclear Station Units 1&2
INSPECTION AT:	Scriba and Syracuse, New York
INSPECTION DATES:	October 19-29, 1992

INSPECTOR: Roy K Mathew — 11-25-92
R. K. Mathew, Reactor Engineer, Electrical Section, EB, DRS — Date

APPROVED BY: W. H. Ruland — 11/25/92
W. Ruland, Chief, Electrical Section, Engineering Branch, DRS — Date

Areas Inspected: Design, design changes and modifications, installation and testing of modifications, interface/communication, quality assurance and technical support for the deviation/event reports (DERs) and work requests (WRs). This inspection also reviewed one of the previously identified open items.

Results: Generally, modifications and design changes reviewed were of good quality and technically accurate. However, one issue remained unresolved regarding a simple design change (SDC NO. SC1-0217-91) that was issued for implementation in Unit 1 without performing a thorough analysis and documentation. The technical evaluations, reportability, operability, and corrective actions for DERs and WRs were found to be adequate. The quality assurance audits and surveillance program were found to be sufficient to identify issues in licensee's engineering and technical support area and to correct them in a timely manner. Good communication/interface exists at Nine Mile between engineering and plant staff. Backlog of DERs remained high and in need of continued management support. This inspection also closed an unresolved item (50-410/92-17-02) regarding Unit 2 Division III service water design deficiency.

1.0 SCOPE OF THE INSPECTION

The purpose of the inspection was to verify that the design changes, modifications and post-modification testing for the Nine Mile Point Units 1 and 2 were performed in accordance with plant procedures, requirements and commitments specified in the facilities Technical Specifications (TS), NRC rules and regulations, safety analysis report and the quality assurance (QA) program. Also included in the scope of the inspection was the assessment of the communication/interface between engineering and site organizations, and technical support for the resolution of DERs and WRs.

1.1 Administrative Controls for Design Changes and Modifications

Administrative procedures and engineering procedures were reviewed to determine whether the engineering activities were specified and controlled by approved procedures. The procedures reviewed included plant modifications, design change request, design input, design verification, safety evaluations, design document changes, the configuration management process, station operations review committee (SORC) reviews, deviation/event reports and the modification/simple design change program.

The review indicated that the licensee's procedures provided adequate administrative guidelines and controls to ensure that design, design changes and modifications performed did not involve an unreviewed safety question. Appropriate requirements and guidelines were provided for the 10 CFR 50.59 screening review and safety evaluations, design input, design calculations and design verifications. However, during the review of procedure NEP-DES-320, "Design Change Initiation," it was noted that the simple design change package check list did not require any supervisory approval for design changes that cost up to $25,000. The inspector was concerned that safety-related design changes could be generated and approved without proper management review. In response to this concern, the licensee stated that the design documents generated as part of the design change process would get proper approval and therefore, further approval was not needed. However, the licensee stated that they would review the existing procedure to determine whether there is any need for management approval on the simple design change package checklist. The licensee's engineering staff was interviewed to determine their understanding of the modifications and design process. Discussions showed that the engineering staff was knowledgeable of the procedural requirements and guidelines.

The inspector concluded that the licensee had established adequate procedures and programs to ensure that plant design changes, modifications and engineering activities were performed in a controlled manner.

1.2 Design Changes and Modification Program Implementation

Several simple design changes and modifications for Units 1 and 2, as shown in attachment 2, were randomly selected and reviewed to determine that they were performed in conformance with the requirements of the Technical Specifications (TS), 10 CFR, the Safety Analysis Report, the licensee's Quality Assurance Program and in accordance with licensee's procedures. Also, technical quality of modifications, thoroughness of design analysis, design input, technical review and safety evaluations, management involvement and review and resolution of problems from a safety standpoint were evaluated.

Generally, the modifications and design changes reviewed were found to be well organized, complete and in accordance with the applicable procedures. Materials, processes, parts and equipment were identified properly and were suitable for its application. Applicable design inputs were documented correctly into the design, except for one case which was not documented properly. This case is discussed further in this section. A review of the preliminary and final safety evaluation for the modifications and SDCs indicated that generally, the safety evaluations were adequate and provided supporting conclusions. All the pertinent information was provided in the design package but the information provided in the 50.59 review was not always sufficiently descriptive to support the conclusions. The inspector noted that one would have to review the whole package to understand the full scope of the modification.

The design considerations included evaluations such as the licensing basis, equipment qualifications, fire protection, fuel analysis, control room habitability, ALARA, ISI/IST and seismic qualifications. The design drawings were observed to have been marked-up or revised to reflect the as-built configuration. The post-modification tests were properly identified and successfully completed before operations acceptance. Partially completed modifications for multi-system modifications were controlled adequately. No configuration control problems were identified during this review. Installation packages reviewed were found to be adequate. Constructability walkdown for the "design complete" modifications and as-built verification for the completed modifications indicated that the design and modification implementation were adequate. Design verifications reviewed were found to be in accordance with applicable procedures.

During the review of design changes and modifications, the inspector interviewed several system, site and corporate engineers and project staff at both units to determine their understanding of the modification process and knowledge of their assigned systems. The engineers and project team members were very knowledgeable of their modifications and design changes. Also, good communication existed between the different engineering groups and the plant staff.

During the review of simple design change SC1-0217-91, Revision 0, several discrepancies were identified. This design change was issued to the field to fabricate and install a monorail and trolley to rig a containment isolation valve (33-04) located on the 281 ft elevation of the Unit 1 reactor building. This design change would reduce radiation exposure to the maintenance staff during work on this valve. The licensee considered this design change a nonsafety-related modification since it did not perform any safety function and it was used only during maintenance.

The licensee's 10 CFR 50.59 preliminary safety evaluations did not document the seismic impact of this nonsafety modification. Furthermore, the structural design calculation (S6-RX261-R1G01) generated for this SDC did not consider the effect of seismic loads. However, the licensee stated that during the design process, the impact due to seismic load was considered and an engineering judgement was made that this change did not impact the loading limits of the attached safety-related structural beam. The inspector noted that the licensee's procedures (NEP-DES-320 and NEP-DES-142) did not provide clear guidelines regarding seismic evaluations for permanent loads added to the safety-related structures. However, the licensee had adequate procedures to address seismic evaluations for electrical and mechanical component replacements inside the safety-related buildings.

Subsequent to the inspection, the licensee performed a re-evaluation of the structural loading calculation (S6-RX261-R1G01) for the seismic loads. An additional page was added to the calculation to document the effects of seismic loads. The calculation concluded that the addition of seismic loads has no impact on the original calculation conclusions. A review of the licensee's evaluation showed that the licensee had neither considered the weight of the valve nor the full design capability of the rigging unit (1000 lbs) for calculating the seismic forces. The licensee stated that they did not consider the seismic loads generated from the valve since the valve rigging was performed only during maintenance and the valve was inoperable during that period. The licensee did not consider that plant maintenance could be performed during plant operation and the design should account for the bounding conditions. This item is unresolved pending the licensee establishing the following: 1) Calculations to show that seismic loads for the rigging unit (for the maximum design rating of 1000 lbs) would be within the design capacity of the attached structure; 2) 10 CFR 50.59 review to document this evaluation, and 3) Developing appropriate procedures and guidelines to evaluate and document seismic impact for loads attached to the safety-related structures (50-220/92-26-01).

2.0 ENGINEERING SUPPORT FOR DERs AND WRs

The inspector reviewed several randomly selected Deviation/Event Reports (DERs) to determine whether the engineering dispositions were technically accurate and based on established requirements, and to determine the degree of engineering support in their resolution.

The review indicated that there is a large backlog of DERs (749) for the nuclear engineering department. However, the licensee has made very little progress in reducing the existing backlog since the last SALP period. The engineering dispositions including the preliminary screening, reporting and operability reviews were found to be adequate.

During the review of Unit 1 DER 1-91-Q-1543 regarding main steam isolation valve (MSIV) motor failure due to the degradation of the magnesium rotor, the inspector noted that the licensee had addressed the specific problem with the MSIV motor. They have not yet established a program in Unit 1 to address potential problems with other motors. Unit 2 had already implemented a program to monitor the possible degradation of these motors by performing borescope inspections and periodic locked-rotor current monitoring. This problem was originally identified in the industry in 1986. To address this issue at Unit 1, a task team was developed by the licensee to review and provide appropriate recommendations. The team recommended that a monitoring and/or replacement program should be developed and implemented to address this issue. The licensee stated that the motors with magnesium rotors inside the containment at Unit 1 are scheduled to be replaced in the 1993 refueling outage and further inspections and actions are to be implemented based on the outcome of the inspections. It was noted that the affected motors (10 magnesium rotor motors) were relatively new and were tested on a quarterly basis. The licensee determined that the degradation of these motors, until the program is implemented, is very low. The operability and environmental qualification aspects of these motors were reviewed. No unacceptable conditions were noted during this review.

Several maintenance work requests (WRs) were reviewed to determine whether any modifications or design changes were performed through the routine maintenance work request process. The review indicated that maintenance work did not involve any design changes or modification to the plant.

3.0 COMMUNICATION AND INTERFACE CONTROL

Good communication exists between the plant and engineering personnel. This was evidenced by the staffing of site and system engineering groups for each plant to support the engineering/technical needs of the plant. Furthermore, the communications between operations, maintenance and the engineering group were established through morning meetings at each unit and also a combined meeting for both units. The inspector attended the combined plant staff meeting and noted that representatives from site and system engineering, corporate engineering, plant operations and maintenance staff attended the meeting. During this meeting, representatives from different sections of each unit discussed plant status, daily action items, problem areas and any technical issues. The inspector determined that this meeting provided an effective communication channel between different divisions of both plants. The active participation of management representatives from different organizations at these meetings complimented the effective communication at Nine Mile.

4.0 TEMPORARY MODIFICATIONS (T-MODS)

The inspectors reviewed the licensee's temporary modification program to assure that temporary installations were performed and controlled in accordance with licensee's procedure AP 6.1. A sample review of temporary modifications for each unit indicated that the licensee was performing adequate 10 CFR 50.59 evaluations and technical review. A plant walkdown verification of some of the open T-mods indicated that they were tagged in accordance with the applicable procedure. A review of the temporary modification log kept in the control rooms for Units 1 and 2 indicated that the licensee had made some progress in reducing the number of open temporary modifications. The licensee stated that they were in the process of making some of the modifications permanent and were assigning priorities to eliminate long standing temporary modifications.

5.0 QUALITY ASSURANCE (QA) INVOLVEMENT IN TECHNICAL SUPPORT

The quality assurance area was reviewed to evaluate QA involvement in assessing the quality of engineering services. Audit reports and surveillance reports for the period April through September 1992, were selected for review to determine the above.

Results of the audits and surveillance provided for a self-assessment of engineering performance including identification of strengths and opportunities for improvement. Also, the audits revealed weaknesses in the implementation of modification programs. No significant deficiencies were identified. Discussions with the QA staff indicated that the weaknesses identified in the engineering programs were being addressed by the engineering management in a timely manner. The inspector concluded that quality assurance had adequate involvement to monitor and implement appropriate corrective actions in a timely manner.

6.0 CONCLUSIONS

In summary, modifications and design changes reviewed were generally of good quality and technically accurate, except for one case, where the licensee failed to perform thorough evaluations and documentation to verify the seismic loads for a non-safety design change in the reactor building. Engineers and project team members were very knowledgeable of their modification and design changes. The 10 CFR 50.59 screening process, safety evaluation, design input, SORC and technical review, post-modification testing were found to be thorough and in accordance with procedures and applicable regulatory requirements. Participation of system engineers and site engineers were noted during the modification review. Quality Assurance had adequate involvement to monitor and implement corrective actions in a timely manner. Good communication exists at Nine Mile between engineering and plant staff. The backlog of DERs remained high and in need of continued management support.

7.0 STATUS OF PREVIOUSLY IDENTIFIED ITEMS

7.1 (Closed) Unresolved Item (50-410/92-17-02) Unit 2 Division III Emergency Diesel Generator (EDG) service water design deficiency:

This item pertains to the design deficiency in the logic for the Division III EDG low service water pressure isolation during the Unit 2, March 23, 1992, loss of off-site power (LOOP) event. The design caused the diesel cooling water flow to isolate if the diesel had been running for more than one minute before the low pressure condition occurred. The licensee addressed this issue by implementing a modification which installed new time delay relays for the supply valves and increasing the time delay for the division 1/2 discharge valves. This modification and licensee's actions were reviewed by the NRC previously and were documented in inspection report No. 92-17. During that review, the inspector determined that one issue needed to be addressed by the licensee to determine the safety significance. Specifically, once a Division I or II EDG was running for more than the 160 second time delay, it would immediately isolate on low service water pressure for a Division III service water pipe rupture before the Division III service water isolates. The licensee was asked to determine if Division III service water pipe break was a credible scenario and if so, what the effect would be on Division 1 and 2 EDGs.

During this inspection, the inspector reviewed the licensee's evaluations to address the above issue. The review indicated that the Division III service water lines are designed and installed as safety-related and to the seismic Category 1 standards. The scenario explained above would happen only if there is a passive failure in the Division III service water piping. The service water lines are designed to withstand any operational and design basis event. The licensee's design basis and licensing basis do not postulate any passive failures. Also, no credible active failures exist to cause the above scenario. Based on this, the inspector determined that the existing design did not cause any unacceptable conditions. This item is closed.

8.0 UNRESOLVED ITEMS

Unresolved items are matters about which more information is required to ascertain whether they are acceptable items or violations. One unresolved item identified during this inspection is discussed in detail in Section 1.1 of this report. A previously identified unresolved item is discussed in Section 7.0.

9.0 EXIT INTERVIEW

At the conclusion of the inspection on October 29, 1992, the inspector met with licensee representatives denoted in Attachment 1. The inspector summarized the scope and results of the inspection at that time.

ATTACHMENT 1

Persons Contacted

Niagara Mohawk Power Corporation (NMPC)

R. Abott, Manager, Unit 2 Engineering
V. Atnasov, Electrical Design
J. Chamberline, Project Engineer
* J. Conway, Manager, Technical Support, Unit 2
D. Cummins, Modifications, Technical Support, Unit 2
W. Crandall, Systems Engineer
J. Darweesh, ISEG Engineer
* A. DeGracia, OPS Manager/Plant Manager, Acting
J. Driscoll, Systems Engineer
G. Elridge, EQ program Manager
D. Goodney, Electrical Design, Unit 1
A. Julka, Design Supervisor, Unit 2
S. Kim, Electrical Design Engineer
* J. Kroehler, Manager, QA Engineering
* R. Magnant, Site Licensing
* P. Mangano, Supervisor, Site Engineering, Unit 2
M. McCormick Jr, Plant Manager, Unit 2
R. Morey, Systems Engineer
G. Pace, Supervisor, Procurement
T. Picciot, Technical Support, Unit 1
* A. Pinter, Site Licensing Group
D. Sandwick, Supervisor, NMP1, Project Management
* J. Spadafore, Program Director, ISEG
J. Sullivan, Supervisor, Unit 2, Project Management
* K. Sweet, Manager, Unit 1,Technical Support
C. Terry, Vice President, Nuclear Engineering
* G. Thompson, General Supervisor, Unit 2.
* J. Vinquist, President, MATS Inc./NMPC
B. Walker, Supervisor, Site Engineering, Unit 1
D. Weaver, Supervisor, Procurement Engineers
* B. Wolken, Supervisor, Site Engineering, Unit 1
W. Weaver, Procurement Engineer
W. Yaeger, Manager, Unit 1, Engineering
* A. Zallnier, Supervisor, Site Licensing

<u>U.S. Nuclear Regulatory Commission (NRC)</u>

* W. Schmidt, Sr. Resident Inspector
* J. Yerokun, Project Engineer

* Denotes those attending the exit meeting conducted on October 29, 1992.

ATTACHMENT 2

DESIGN CHANGES AND MODIFICATIONS REVIEWED

1. SC2-0301-91- Diesel jacket circulating water pump pressure set point changes
2. SC2-0423-91- Correct as-built wiring for 2OFGA0103
3. SC2-0194-92- Correct low dc bus alarm wiring
4. SC2-0148-91- Relocate UPS for main stack
5. PN2Y86MX085- Delete nuisance alarms in the control room
6. PN2Y89MX138- Install remote terminal unit and connect with energy management system
7. PN2Y90MX021- Replacement of Riley temperature switches
8. PN2Y92MX006- Revise logic for service water valves MOV 95 A/B and MOV 66A/B
9. N1-91-016- EDG performance monitoring
10. N1-90-020- Torus airspace pressure redundant instrumentation
11. SC1-0217-91- Installing rigging for valve 33-04
12. SC1-0112-92- Change setpoint for pressure switches for diesel fire pump
13. SC1-0028-91- Install pressure taps between each ESW check valve and blocking valve
14. SC1-099-92- Revise torque switch settings
15. N1-91-008- RBEV non-coincidence logic changes
16. N1-90-0192- Hydrogen/Oxygen monitoring system upgrade
17. N1-89-115- G panel ground separation
18. N1-90-143- Replace 345 kV capacitor voltage transformer