U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket/Report Nos:	50-220/97-09 50-410/97-09
Licensee:	Niagara Mohawk Power Corporation P.O. Box 63 Lycoming, NY 13093
Facility:	Nine Mile Point, Units 1 and 2
Location:	Scriba, New York
Dates:	August 4-15, 1997 and August 25-29, 1997
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EXECUTIVE SUMMARY

This engineering inspection was conducted to review: (1) engineering activities involving plant design changes, environmental qualification of electrical and mechanical equipment, Generic Letter 89-10 motor-operated valve (MOV) program, engineering staffing and training; and 2) licensee actions in response to the identification that 40 isolation valves in the reactor water cleanup system (RWCS) were susceptible to fire-induced hot shorts. This report covered the result of a two-week onsite inspection by three regional-based inspectors and an NRC contractor.

<u>Engineering</u>

- Immediate corrective actions following the identification that 40 isolation valves in the RWCS were susceptible to fire-induced hot shots were appropriate. However, an apparent violation of Unit 2 operating license No. NPF-69, Item 2.G was identified. (Section E1.1)

- Overall, environmental qualification (EQ) procedures met the EQ and EQ maintenance program requirements. However, a violation was identified involving the use of an unapproved procedure to conduct software testing. An additional violation (with two examples) involving a lack of written instructions or procedures for the calculations of the qualified life of mechanical EQ equipment and for the environmental qualification environmental design criteria (EQEDC) database also was identified. (Section E1.2)

- The scope of the Generic Letter 89-10 motor-operated valve programs at NMP were acceptable for program closure. The licensee made a reasonable attempt to test under dynamic conditions as many valves as practical. (Section E1.3.1)

- At Unit 1, several of the valve factor justifications for nontestable MOVs were weak and nonconservative. However, no immediate operability concerns were identified. The licensee committed to bolster its valve factor assumptions with additional industry information and/or analyses (e.g., EPRI PPM). In most cases, the licensee demonstrated design-basis valve capability margins that were adequate for GL 89-10 program closure. Closure of program at Units 1 and 2 is based upon the commitments discussed in Section E1.3 of this report. (Sections E1.3.2 and E1.3.3)

- While no invalid or unreasonable design inputs were identified during review of the licensee's design-basis MOV calculations, greater attention to detail and more rigorous review of design inputs are warranted. (Section E1.3.5)

- The design change for the replacement of recirculation system sample containment isolation valves met NRC and licensee design control requirements and appropriately considered applicable codes, standards, and specifications. However, the modification was not effective in resolving the problems for which it was implemented, and introduced an additional problem (pressure boundary leakage) as well. (Section E1.4)

- Design modification packages and associated safety evaluations were acceptable. Pertinent design inputs were specified and documented and required design reviews were performed. Adequate post-modification tests verified proper implementation of the changes. Compliance with engineering procedures was evident. (Section E1.5)

- Unit 1 Service Water System (SWS) Design Basis Document (DBD) appeared comprehensive and well prepared and was readily available to design personnel. (Section E1.6)

- Adequate mechanisms were provided for Unit 2 to maintain plant configuration through computerized design change document controls. Documents affecting the configuration of Unit 2 are available and retrievable in a user-friendly manner. (Section E1.6)

- The inspectors concluded that the Unit 2 design engineers were cognizant of the technical issues supporting the preparations for the next refueling outage. Unit 1 design engineers were actively involved in the resolution of NRC Bulletin 96-03. (Section E2.1)

- The licensee has critically self-assessed the quality of engineering dispositions of DERs, and is seeking ways to improve the engineering performance in this area. The inspectors concluded that initiatives such as this self-assessment were appropriate. (Section E2.1)

- The EQ group experienced manpower difficulties due to a loss of three (out of four) experienced personnel. Niagara Mohawk engineering training program provided abundant technical courses to both new-hired and existing technical staff. (Section E5.1)

- Nine previously identified inspection items were closed. The closure of one item (URI 50-220/96-07-07) resulted in a noncited violation of 10 CFR 50, Appendix B, Criterion VI, in that 133 updated critical drawings were not distributed for the use by operations and other personnel. (Section E8)

Report Details

Summary of Plant Status

This was the final routine engineering inspection during this SALP period (June 1996 to November 1997). Unit 1 was at 100% power during this inspection. Unit 2 was manually scrammed on August 4, 1997, to repair a flex hose in the containment (see Section E1.5), and was restarted on August 9, 1997. Unit 2 power was increased to 95% on August 13, 1997, and remained at this level (due to a moisture separator reheater problem) for the rest of the inspection period.

E1 Conduct of Engineering

E1.1 Reactor Water Cleanup System (RWCS) Isolation Valves - Unit 2

a. Inspection Scope (92700)

On August 26, 1997, the licensee identified that the control circuits of twenty pairs of isolation valves in the RWCS were susceptible to a fire-induced hot short problem. The inspectors reviewed this issue to determine the regulatory and safety consequences. The inspectors also reviewed the licensee's immediate corrective actions.

b. Observations and Findings

Background

On April 7, 1997, the licensee identified a deficient condition within the control circuitry of the Unit 2 emergency diesel generator service water valves. Subsequently, the licensee issued Licensee Event Report (LER) 97-02 to report the deficient condition to the NRC. As part of the corrective actions for LER 97-02, the licensee committed to perform a confirmatory evaluation of the Unit 2 plant design to verify that the systems required to achieve safe shutdown during a control room exposure fire were in accordance with the requirements of General Design Criterion 3 (Fire Protection) of 10 CFR 50, Appendix A, by December 31, 1997.

In August 1997, the licensee hired five auditors from Duke Engineering in Charlotte, North Carolina, to perform the confirmatory evaluation (an audit) as mentioned above. The audit started on August 4, 1997, and was ongoing at the time of this inspection.

The Issue

On August 26, 1997, the Duke Engineering team identified a deficient condition involving the RWCS isolation valves. There were twenty pairs of air-operated valves (AOV) in the RWCS that were used to isolate high pressure reactor coolant from low pressure piping and tanks. Each pair of valves was installed in series, and all valves were normally closed and fail closed. If any one or more pairs of these AOVs were opened inadvertently during plant operation, the reactor coolant pressure could rupture the low-pressure system, creating a small break loss-of-

coolant (LOCA) condition. According to the licensee, this scenario would be limited to a maximum flow of 150 gpm. Above this flow rate, a flow mismatch condition would cause the RWCS system containment isolation valves to close within 45 seconds.

The control functions of these valves were nonsafety-related. Therefore, electrical separation was not provided for the control circuits of these valves. The inspectors' review of the electrical diagrams indicated that the control circuits of these valves were located in two electrical panels (P187 and P188) mounted side-by-side at elevation 328 feet of the reactor building.

The Duke Engineering team identified that a postulated fire in the local area could cause the control circuits to be bypassed. This condition could energize the associated solenoid valves and open the AOVs, resulting in a small-break LOCA scenario as described above.

Immediate Corrective Actions

The licensee issued a Deviation/Event Report (DER 2-97-2519) on August 26, 1997, to document and track the resolution of this condition. The licensee also issued a night order on August 26, 1997, to provide a continuous fire watch in the area where P187 and P188 were located, and changed the operating procedure to isolate the RWCS system upon confirmation of fire in that area.

The inspectors conducted a walkdown of reactor building at elevation 328 feet on August 27, 1997, confirmed the location of panels P187 and P188, and verified the presence of fire watches in the area.

c. Conclusion

The inspectors concluded that the licensee's immediate corrective actions following the identification of the deficient condition were appropriate. However, this deficient condition constituted an apparent violation of Nine Mile Point Unit 2 Operating License No. NPF-69, Item 2.G, which requires the licensee to implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR). Table 9B.5-1 of the FSAR, which is part of the approved fire protection program, identified the twenty pairs of isolation valves in the RWCS as high/low pressure interface valves. A note associated with these valves stated that no single fire could cause sufficient spurious operations to violate the high/low pressure interface in this flow path. However, a single fire in the vicinity of control panels P187 and P188 could cause the high/low pressure isolation to fail and create a small-break LOCA condition. (EEI 50-410/97-09-01)

E1.2 Mechanical and Electrical Environmental Qualification Program - Unit 2

a. Inspection Scope (37550)

The inspectors reviewed the Unit 2 mechanical and electrical environmental qualification (EQ) program and procedures, and the computer databases developed to calculate the qualified lives of the mechanical and electrical EQ equipment, to determine whether these procedures and databases met the regulatory requirements.

b. Observations and Findings

Review of EQ Program and Procedures

Implementation of the Unit 2 EQ program is discussed in Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," of the Unit 2 FSAR. The EQ program was developed to fulfill the requirements of 10 CFR 50.49, General Design Criteria 1,4,23 and 50 of 10 CFR 50, Appendix A, and Criterion III of 10 CFR 50, Appendix B. The licensee had developed Administrative Procedure NEP-DES-25, "Environmental Qualification Program," dated October 30, 1995, to provide administrative control of the EQ program.

The original EQ program was developed by Stone & Webster Engineering Corporation (S&W) and was prescribed in two S&W documents, (1) Environmental Qualification Document (EQD), dated April 9, 1987, with Addendum 13, dated April 23, 1993; and (2) Environmental Qualification Environment Design Criterion (EQEDC), Revision 4, with Addendum 2, dated March 29, 1993. The second document (EQEDC) established the environmental conditions, including temperature and pressure profiles and radiation doses at various zones where the equipment was located. Section 5.1 of EQD discussed the aging analysis of all EQ equipment. This analysis was a large document consisting of about thirteen binders, and was used as the basis for EQ maintenance requirements (EQMR). The inspectors did not identify any unacceptable conditions pertaining to these documents.

Development of EQ Databases

With the intention of making EQ documentation easier, the licensee issued an Information Service Request (ISR E-92-0532) on February 3, 1992, to develop an integrated database for the EQEDC. Subsequently, the licensee hired a contractor to develop the database. This database consisted of two parts, EQEDC and EQEDC II. The second part (EQEDC II) was also entitled, MEQ/SCEW. EQEDC was for environmental data input only. EQEDC II could provide calculations (such as aging analysis) and generate SCEW (system component evaluation work) sheets.

Development of these two databases generated many volumes of documents that required licensee approval and signoff, including: (1) software requirement specifications, (2) software design specifications, (3) an acceptance test plan (which served as software validation and verification), and (4) a software training package and implementation plan.

The inspectors reviewed these documents and found most of them properly approved and signed off. However, two documents, Acceptance Test Plan for EQEDC II Database, and Software User Manual for SCEW and MEQ Database Management System, were not approved and signed off. The licensee stated that the second document was not intended to apply to safety functions. However, the first document was used to validate and verify the EQEDC II database, and according to the licensee, this database had been used to calculate the qualified lives of electrical EQ equipment for the Unit 2 power uprate (a safety-related function). Unit 2 had updated its rated thermal power output for more than a year. The inspectors' review of this document indicated that the acceptance tests for EQEDC II had been conducted on March 30-31, 1994, and several test anomalies were noted. The inspectors informed the licensee of this condition (test with an unapproved test plan), and subsequently, the licensee issued DER C-97-2502 on August 22, 1997, to document this problem.

The licensee stated that they had completed a preliminary (informal) acceptance test during the week of August 25, 1997, to confirm the reliability of the software in question. Random data extractions were also made from the database tables to confirm that all of the information for various parts were present. A formal verification and validation test would be completed at a later date. The inspectors determined that the above condition (test with an unapproved test plan) constituted a violation 10 CFR 50, Appendix B, Criterion VI, Document Control, which requires that measures shall be established to control the issuance of documents, such as instructions and procedures, and that these measures shall assure that documents are reviewed for adequacy and approved for release by authorized personnel. **(VIO 50-410/97-09-02)**

Qualified Life Calculations of Electrical and Mechanical EQ Equipment

The licensee stated that in 1994, when the EQEDC II database was used to calculate the qualified lives of electrical EQ equipment to support the Unit 2 power uprate, the EQEDC II database was not ready for calculating the mechanical EQ equipment qualified lives (also for power uprate). The licensee also stated that a mechanical EQ engineer at that time used the R-base database, which the individual was familiar with, to perform the required calculations. The inspectors were provided with a list of calculation results (about 30 pages listing approximately 1,500 items) for review. The results indicated that about 80% of the items had qualified lives of 40 years. The inspectors asked the licensee how these calculations were performed, what formulas were used, whether the calculation process was documented, and whether the program used for the calculations had been validated and verified. The licensee could not provide satisfactory answers to these questions.

During the inspection, the licensee performed calculations and analyses for a sample of 53 items (all under 40 years qualified life). The results of these calculations and analyses confirmed the correctness of the original calculation results. However, the lack of documentation to support the calculation process constitutes a violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, which requires activities affecting quality to be prescribed by documented instructions or procedures, and accomplished in accordance with these instructions or procedures. **(VIO 50-410/97-09-03)**

Interface Software for EQEDC and MEL2 Databases

In 1996, the licensee completed the interface software for the EQEDC database and the MEL2 (Unit 2 master equipment list) database. The MEL2 database listed all safety- and nonsafety- related equipment at Unit 2. A dedicated engineer from the Unit 2 Project Management group was responsible for maintaining and updating the MEL2 database, while the EQ group was responsible for the EQEDC database. The Unit 2 EQ master list (EQML), consisting of electrical and mechanical EQ components, was a subset of MEL2 database.

Three documents were generated for the development of the interface software, including a software requirement specification, a software design specification, and a software test plan. The inspectors noted that all of these documents were properly approved and signed off. The personnel responsible for the interface software stated that, following the completion of the interface software, they downloaded the data from the MEL2 database to the EQEDC database. They also explained that this interface software was developed in such a way that only MEL2 data could be downloaded to EQEDC, not vice versa. This provision was to prevent the MEL2 database from being compromised by the EQEDC database. Following downloading, the EQML from the MEL2 database should match the EQML from EQEDC database. Any differences would be due to changes in the EQEDC database and should be explainable.

The inspectors conducted a test to determine the validity of the interface software. Because the EQML was a large subset of MEL2, containing more than several hundred thousand components, the inspectors avoided using this for the test. With the help of Niagara Mohawk software personnel, the inspectors first selected 18 mechanical EQ pumps from both databases. The pumps from both databases matched. However, when Limitorque motor-operated valves (MOV) were searched, there were 18 MOVs that were in the MEL2 database, but not in the EQEDC database. According to the licensee, these MOVs (2RHS*MOV1A, 1B, 1C, 2A, 2B, 8A, 8B, 9A, 9B, 12A, 12B, 112; 2SWP*MOV33A, 33B, 90A, 90B) should have been in the EQEDC database because they all required EQ. The licensee's investigation revealed that these MOVs had been deleted for unknown reasons by the former EQ Program manager, who had access to the EQEDC database. The inspectors were informed of this result on September 9, 1997, during a conference call with licensee personnel (one licensing engineer, one general supervisor, one EQ engineer, two software engineers, and the MEL2 engineer). The inspectors questioned the licensee's control of the EQEDC database, which was software that supported safety functions. The licensee explained that the EQEDC and EQEDC II databases had been used only for the Unit 2 power uprate, and had not been officially used thereafter for EQ activities, including EQ maintenance; therefore, any mishandling of the EQEDC database after completion of the Unit 2 power uprate would not cause an operational problem. The inspectors pointed out that there existed no provisions in the EQEDC and EQEDC II documents to restrict their use as stated. The licensee stated that they would not use the EQEDC and EQEDC II databases again until further training and tighter controls were provided, and that they might procure a PC-based EQ database, developed by GLS Enterprise, Incorporated, to perform the EQ documentation instead.

Other potential problems with the EQEDC and EQEDC II databases were also identified by the inspectors on August 25, 1997, and August 28, 1997. The inspectors observed that the search of components from the EQEDC II database resulted in components with negative or zero qualified lives. For example, the qualified life for 2CSH*V256 was -229.5 years, for 2GTYS*TIS9A&B was -15775 years; and for 2RSS*LT114 was -13.82 years. During the inspection, the licensee was able to identify that the wrong qualified life for 2CSH*V256 (and the qualified life for another component identified by themselves on August 26, 1997) was due to lack of input of radiation dosage. For the other incorrect calculation results, the licensee stated that the affected components were not within the scope of the Unit 2 power uprate and that the wrong calculation results had never been used. The inspectors pointed out that the EQEDC and EQEDC II database documents did not limit the use of these databases only to the Unit 2 power uprate calculations. The inspectors determined that the above conditions constituted a second example of a violation of 10 CFR 50, Appendix B, Criterion V. **(VIO 50-410/97-09/03)**

Review of Information Service Requests

The inspector reviewed the Information Service Reports (ISR) that were related to the EQEDC or MEQ/SCEW databases. According to the licensee, an ISR was a request for database service to be performed by the software group. The database service could be large in scope, such as the development of a database software, or small items for the resolution of certain software discrepancies.

There were 23 ISRs in this category issued between January 1992 and June 1997. Four ISRs were to request software development and funding. The remaining 19 ISRs involved problems with the EQEDC and EQEDC II (MEQ/SCEW) databases. All of the problems appeared to be minor in nature, and 15 were closed. The eight open ISRs were all issued this year; three on January 1, 1997, and five on June 10, 1997. The inspectors questioned the status of the open ISRs. The licensee stated that they were still working with those ISRs and still did not have a firm schedule for closure. However, all ISRs would be closed before EQEDC and EQEDC II could be used officially. The inspectors had no further questions.

c. Conclusions

The inspectors concluded that the licensee had provided sufficient EQ procedures in its original EQ program to meet EQ and EQ maintenance requirements. However, there were nonconformances in the licensee's processes for performing EQ activities. The inspectors identified one violation involving the use of an unapproved procedure to conduct software testing. Another violation was identified with two examples involving a lack of written instructions or procedures for the qualified life calculations of mechanical EQ equipment and for the EQEDC database.

E1.3 Generic Letter 89-10 Motor-Operated Valve Program Review (TI 2515/109)

Introduction and Purpose

On June 28, 1989, the NRC issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," requesting licensees to establish a program to ensure that switch settings for safety-related motor-operated valves (MOVs) were selected, set, and maintained properly. Seven supplements to the GL have been issued to provide additional guidance and clarification. NRC inspectors of licensee implementation of the provisions of the GL and its supplements have been conducted based on the guidance provided in NRC Temporary Instruction 2515/109, "Inspection Requirements for Generic Letter 89-10," which is divided into Part 1, "Program Review," Part 2, "Verification of Program Implementation," and Part 3, "Verification of Program Completion."

The NRC conducted the initial Part 1 inspection at Nine Mile Point (NMP) in October 1992, as documented in NRC Inspection Report (IR) 92-82. A second MOV inspection was conducted to follow-up the items identified during the Part 1 inspection, as documented in NRC IR 50-220/93-22 (Unit 1) and 50-410/93-21 (Unit 2). Part 2 inspections, which were conducted in May 1995 and October 1996, and documented in NRC IRs 95-11 and 96-15, respectively, included an update of the open items developed during the previous inspections. The purpose of this Part 3 inspection was to closeout the NRC staff's review of the GL 89-10 program at Nine Mile Point.

E1.3.1 GL 89-10 Program Implementation

a. Inspection Scope

The inspectors reviewed licensee reports NER-1M-041 and NER-2M-010, Generic Letter 89-10 Closure Summary for the Motor Operated Valve Program Implemented at Niagara Mohawk Nine Mile Point Nuclear Station Unit 1 (Unit 2), dated August 4, 1997, and November 22, 1996, respectively. These documents contained technical discussions regarding program assumptions for rate of loading, stem lubricant degradation, valve factor, and stem friction coefficient, and the results of third-party program assessments. For Unit 1, the inspectors also reviewed valve thrust calculations and tables that summarized the available valve factors (thrust margins) for the MOVs in the program. Using these documents, the inspectors selected for detailed review a sample of low margin MOVs. For GL 89-10 program closure at Unit 2, the inspection consisted of a review of outstanding followup items and commitments from previous inspections. These items are discussed in Sections E1.2.3 and E8 of this report.

b. Observations and Findings

The licensee's methods of demonstrating MOV design-basis capability included verification by: (1) valve-specific dynamic test at design-basis conditions, (2) valve-specific test linearly extrapolated to design-basis conditions, and (3) industry information obtained from tests of similar valves. The Unit 1 program consisted of 37 MOVs (22 gate valves, 11 globe valves, and 4 butterfly valves), ten of which were tested under dynamic conditions. The program at Unit 2 included 177 MOVs (78 gate valves, 56 globe valves, 37 butterfly valves, and 6 ball valves), and 43 dynamic tests were conducted.

The inspectors reviewed design calculations, test packages, and engineering evaluations for the following MOVs:

Unit 1

31-07	Main feedwater isolation
31-08	Main feedwater isolation
39-07	Emergency condenser steam isolation
33-01R	Reactor water cleanup return inboard isolation
33-02R	Reactor water cleanup supply inboard isolation
201-07	Torus air vent & purge isolation

Unit 2

2ICS*MOV124	Reactor core isolation cooling test flow control
2ICS*MOV121	Reactor core isolation cooling steam supply isolation
2CSH*MOV110	High pressure core spray test return
2MSS*MOV111	Main steam drain isolation

During the GL 89-10 inspection at Unit 2 in October 1996, the inspectors identified that high pressure core spray pump test return isolation valve 2CSH*MOV110 did not have adequate capability margin if a generic load sensitive behavior factor of 25% were assumed. The licensee committed to dynamically test the valve during the first quarter of 1997 to strengthen the load sensitive behavior assumption applied to the valve. The licensee performed the test and the inspectors verified through review of the data that the valve factor, load sensitive behavior, and stem friction coefficient values were within the program design-basis assumptions.

At Unit 1, containment spray to radwaste plug valves 80-114 and 80-115 were excluded from the GL 89-10 program. The valves are primary containment isolation valves with a safety function to close. Per Engineering Design Standard 1M-EDS-002, "Design Basis Review For Safety-Related Motor Operated Valves," the licensee removed the valves from the program on the basis that GL 89-10 specifically mentioned only globe, gate, and butterfly valves. The inspectors considered the justification to be contrary to Supplement 1 (Question 4) of the GL, which states that all MOVs in safety-related piping systems are to be considered within the scope of the program. Subsequently, the licensee stated that the

applicable technical specification limiting condition for operation is entered whenever the valves are opened. On this basis, the inspectors concluded that exclusion from the GL 89-10 program was justified. The inspectors also noted that the valves are operated routinely under close to design basis differential pressure conditions, and that the torque switch settings had been established on the basis of calculations performed under quality assurance design controls.

c. Conclusions

The scope of the GL 89-10 programs at NMP was acceptable for program closure. The licensee made a reasonable attempt to test under dynamic conditions as many valves as practical. At Unit 2, a commitment to dynamically test valve 2CSH*MOV110 was completed.

E1.3.2 Operator Sizing and Switch Settings

a. Inspection Scope

The inspectors reviewed valve packages that established the thrust requirements for the MOVs in the Unit 1 GL 89-10 program. These documents included thrust calculations and test evaluation summaries associated with the selected valves, and MOVs from other valve groups. The licensee's methods for determining minimum thrust requirements and the evaluations of load sensitive behavior and stem friction coefficient were summarized in the Unit 1 Closure Summary Report (NER-1M-041). The purpose of the review was to assess the licensee's justifications for the assumptions used in the thrust calculations which formed the basis for determining the MOVs' design-basis requirements.

b. Observations and Findings

The licensee used standard industry equations to determine the thrust and torque requirements of gate and non-rotating stem globe valves. The results of the calculations determined the allowable thrust and torque setting ranges (called "windows"). The windows incorporated valve and actuator structural (weak link) limits, motor capability under degraded voltage conditions using pullout efficiency in both the open and closed directions, and reduction in motor capability due to elevated ambient temperature. The thrust at control switch trip (CST) was adjusted for load sensitive behavior, torque switch repeatability, spring pack/coefficient of friction degradation, and diagnostic equipment uncertainties. The value of the uncertainties was based on the square root of the sum of the squares methodology with the exception of load sensitive behavior and degradation factor biases. The combination of the uncertainties was used to adjust the thrust and torque windows during setup of the torque switches in the field.

There were four butterfly valves in the Unit 1 program. The valves were 20-inch and 24-inch Streamseal valves manufactured by Allis Chalmers. The operating torques required for seating and unseating were determined by summing the manufacturer's predicted seating/unseating, bearing, and hydrostatic torque requirements. Similar to the gate valves, the torque requirements were adjusted to account for diagnostic equipment uncertainty and torque switch repeatability.

Valve Factor and Grouping

The program at Unit 1 separated the valves into 13 groups. Some of the groups contained valves with small variations in design-basis differential pressure and valve size. However, most groups contained MOVs that were identical concerning manufacturer, size, ANSI pressure class, disk/wedge type, and design-basis differential pressure.

Measured valve factors were compared with the values assumed in the design-basis thrust calculations and with valve factor data compiled from industry sources. Because of the limited number of dynamic tests at Unit 1, the licensee gathered industry test data to justify further the valve factors applied to those groups. The inspectors identified the following weakness in the licensee's valve factor justifications:

- Group A consisted of two 14-inch and four eight-inch Rockwell Equiwedge 900 psi class double-disk gate valves located in the main feedwater and emergency condenser systems, respectively. The valves may be required to close under blowdown conditions. The valves were not practical to test in-situ and had an applied valve factor of 0.47. The valve factor was based on a single Electric Power Research Institute (EPRI) Performance Prediction Program (PPP) test of valve number 43. After a review of available margins, the inspectors noted that the available valve factors for feedwater isolation valves 31-07 and 31-08 were less than the assumed design-basis value (0.42 and 0.40, respectively). The available valve factors were based on the thrust measured at CST reduced to account for torque switch repeatability, diagnostic equipment uncertainty, load sensitive behavior, and stem lubricant/spring pack degradation.

  The inspectors verified through review of static diagnostic test traces that the torque switches were bypassed for at least 95% of the closing stroke. Using motor capability adjusted for degraded voltage, elevated ambient temperature, and changes in stem friction coefficient, the available valve factors were 0.72 and 0.69, respectively. By comparing unadjusted required thrust against degraded voltage motor output capability, the inspectors estimated that the valves had approximately 17% margin to account for changes in stem friction coefficient under dynamic conditions, and concluded that there were no immediate operability concerns with the valves. However, a single industry data point was not considered to be sufficient long-term justification for the design-basis valve factor. The licensee contracted an independent engineering firm (MPR Associates) to calculate

required thrust using the EPRI Performance Prediction Model (PPM). The preliminary results of that analysis indicated a required thrust considerably greater than that calculated by the licensee using a 0.47 valve factor. The inspectors noted that Rockwell Equiwedge valves were not specifically modeled in the PPM, and that the analysis may be overly conservative. Nonetheless, the inspectors considered the preliminary analysis to contain potentially adverse information that required timely evaluation by the licensee to assure valve operability. This is an unresolved item. **(URI 50-220/97-09-04)**

The licensee planned to modify the feedwater isolation valves to increase output capability, and sizing calculations for new motor-actuators were underway. In a letter dated September 30, 1997, the licensee committed to perform the modifications during the next (Spring 1999) refueling outage.

- Group B contained two six-inch Anchor-Darling 900 psi class double-disk gate valves in the reactor water cleanup system (valves 33-01R and 33-02R). The valves may be required to close under blowdown conditions and were not practical to test dynamically. Although a program valve factor assumption of 0.39 applied to the group, a more conservative valve factor (0.5) was assumed to calculate required thrust. The assumption was based on a steam blowdown test performed by Anchor-Darling in 1991 and a single test performed by EPRI as part of the PPP. The inspectors noted that the same valve had been used in both tests, and had not been preconditioned before either test (resulting in a low valve factor). In addition, other industry tests typically have resulted in higher valve factors for this valve style. A single industry data point is insufficient justification for GL 89-10 program closure. During the inspection the licensee obtained a new (and higher) design-basis thrust requirement for valve 33-01R using the EPRI PPM. While reduced, the available capability margin of valve 33-01R, based on motor output capability, remained adequate to support current operability. Due to disk orientation, the thrust requirement of valve 33-02R was lower and the design-basis capability of the valve was acceptable.

The inspectors verified that the torque switches of both valves were bypassed at least the first 95% of the closing stroke, permitting output margin to be based on motor output capability rather than thrust at CST. In both cases, the diagnostic test traces indicated that the valves were wedged adequately at 95% closure. The inspectors noted that the stem friction coefficients derived from static tests (0.01) were considerably lower than the program's design-basis value of 0.2. (A value of 0.077 was used in the licensee's thrust calculation for valve 33-01R). The licensee was concerned that modifications to increase motor-actuator output capability to account for the higher stem friction coefficient would result in exceeding the structural (weak link) limits. The inspectors noted that use of a lower stem friction coefficient assumption could be justified programmatically by periodic confirmatory diagnostic tests. In its September 30, 1997, letter, the licensee committed to evaluate valve 33-01R for modification and provide the NRC a letter with the technical bases for its final decision.

- Group G contained two one-inch Edwards Hermavalve 1500 psi class globe valves and Group K contained two 1.5-inch Yarway 1500 psi class Y-pattern globe valves. The licensee applied a single data point to justify the valve factors for these groups. Although there were no operability concerns, the inspectors did not consider that a single test sufficiently justified the valve factors for GL 89-10 program closure. In its September 30, 1997, letter, the licensee committed to evaluate additional industry data or to perform analyses (e.g., EPRI PPM) to assure that the assumed valve factors were consistent with the best available information.

- Group H consisted of four 20-inch and 24-inch Allis Chalmers 125 psi class Streamseal butterfly valves. The MOVs were the torus and drywell vent and purge isolation valves that typically operate in a low differential pressure air or nitrogen environment. The licensee used the manufacturer's methodology to calculate required torque. The bearing coefficient used to calculate bearing torque was increased to account for the highest coefficient of friction observed during the EPRI PPP test program. However, the manufacturer's methodology was based on the test result from a six-inch valve, and justification for correlating this data to the larger Unit 1 valves was not provided. Based on the low design-basis differential pressure conditions, the inspectors did not have an operability concern regarding these valves. However, in its September 30, 1997, letter, the licensee committed to develop a plan to obtain confirmatory industry information to validate the manufacturer's methodology.

<u>Load Sensitive Behavior</u>

The licensee's Closure Summary Report documented its evaluation of load sensitive behavior at Unit 1. The data (seven data points) were limited due to the small number of dynamic tests the licensee was able to perform. Consequently, the licensee used information developed by the Boiling Water Reactor Owners' Group (BWROG) Valve Technical Resolution Group that correlated stem lubricants with load sensitive behavior. The study indicated that plants using Nebula EP-0 (the lubricant used at Unit 1) should use a bias margin of 4% and a random value of 25% to account for load sensitive behavior. The licensee utilized the recommended margins in its MOV calculations with the exception of rising, rotating stem globe valves. The licensee used a total bias margin of 32% for these valves based on in-situ testing of two valves and review of industry data. While the selected values were acceptable for GL 89-10 program closure, the inspectors noted that the licensee relied primarily on industry information rather than plant-specific test results. The licensee's periodic verification program includes monitoring of load sensitive behavior and on-going assessment of MOV program design assumptions.

Stem Friction Coefficient

The licensee performed a statistical analysis of stem friction coefficient that used 43 static test data points obtained during the MOV test program. The analysis indicated a mean value of 0.12 and a mean plus two standard deviations of 0.197. For most valves, the licensee's calculations utilized a value of 0.2 to determine minimum required actuator torque. The inspectors noted that stem friction coefficients tend to increase under dynamic conditions. The licensee stated that changes in stem friction coefficients had been accounted for by including in its calculations a margin for load sensitive behavior and a 5% margin for lubricant degradation. In addition, in the case of valves set up to operate primarily on the limit switches, the licensee increases the assumed stem friction coefficients by a factor of 0.046. The inspectors considered the licensee's approach to be adequate for GL 89-10 program closure.

Linear Extrapolation

The licensee used a method of linear extrapolation of dynamic test results based on the ratio of the design-basis differential pressure to the maximum differential pressure observed during the test. The licensee reviewed the NRC Safety Evaluation of the EPRI PPM Topical Report, which stated that valve factors derived from tests conducted under significantly less than design-basis differential pressure conditions must also consider whether the contact load is sufficiently large for an accurate valve factor determination. At Unit 1, only two dynamic tests were considerably below design-basis values (approximately 33 to 36% of design-basis differential pressure). The licensee reviewed the test results for the two outliers and bounded the data scatter. The inspectors considered the licensee's approach to be acceptable for GL 89-10 program closure.

c. Conclusions

Several of the valve factors were not thoroughly justified technically and were not conservative. However, no immediate operability concerns were identified. The licensee committed to bolster its valve factor assumptions with additional industry information and/or the EPRI PPM. The licensee's treatment of load sensitive behavior and stem friction coefficient assumptions and linear extrapolation of dynamic test results was acceptable for GL 89-10 program closure.

E1.3.3 Motor-Operated Valve Design-Basis Capability - Unit 1

a. Inspection Scope

The inspectors reviewed dynamic test data traces, performance evaluations and associated test reports for the selected MOVs. The purpose of the review was to assess the licensee's efforts to establish the design-basis capability of the MOVs in its GL 89-10 program.

b. Observations and Findings

Approximately one-third of the valves in the program were tested dynamically. The licensee calculated valve factors for the open and close direction tests of each of the selected MOVs. Inspectors found that the licensee properly reduced and evaluated diagnostic test results and correctly implemented its procedures for assuring capability prior to restoring valves to service. However, the plant data base was limited, and the licensee had to rely on industry information to support the assumptions applied to untested valves. As discussed in Section E1.3, the dynamic testing and/or acquired industry information did not in all cases fully validate all of the valve factor assumptions. However, no immediate operability concerns were identified, and the licensee committed to perform by the next refueling outage several actions to improve capability margins.

c. Conclusions

In most cases, the licensee demonstrated design-basis valve capability margins that were adequate for GL 89-10 program closure. However, closure of the Unit 1 program is based on the commitments discussed in Section E1.3.2 of this report.

E1.3.4 Motor-Operated Valve Design-Basis Capability - Unit 2

a. Inspection Scope

In IR 96-15, the NRC concluded that the licensee had insufficient plant-specific or industry data to justify the valve factor assumptions applied to certain valve groups for GL 89-10 program closure. The inspectors reviewed the licensee's progress in obtaining additional valve factor information for groups V03, V06, GL03a, GL06a, and butterfly valves.

b. Observations and Findings

The inspectors found that the licensee had planned and scheduled modifications to increase the capability margins of the valves in groups VO3 and GL06a. In its letter dated September 30, 1997, the licensee committed to change the gear sets of main steam drain valves 2MSS*MOV111 and 2MSS*MOV112 during the Spring 1998 refueling outage. In addition, the licensee committed to develop a plan to verify the butterfly valve manufacturer's torque predictions. The inspector noted that the licensee planned to perform dynamic tests of two butterfly valves during the next refueling outage.

The inspectors also found that little progress had been made in further supporting the valve factor assumptions for groups V03, V06, or GL03a either through obtaining more data from industry sources or using the EPRI PPM. The licensee committed to obtain additional data or perform analyses (e.g. EPRI PPM) for the valves in these groups by January 30, 1998. The licensee also agreed to notify the NRC in writing upon completion of the commitments. The inspectors considered this an inspection followup item. **(IFI 50-410/97-09-06)**

c. Conclusions

Based on the licensee's commitments, the inspectors concluded that NRC review of the GL 89-10 program at Unit 2 was closed.

E1.3.5 Motor-Operated Valve Design-Basis Calculations

a. Inspection Scope

The inspector reviewed a sample of Unit 1 motor-operated valve design-basis calculations to verify that design inputs were appropriate and properly traceable to valid sources.

b. Observations and Findings

The inspectors found that the great majority of design inputs used in the calculations reviewed were technically correct and properly referenced. However, several examples were identified in which design inputs either had not been updated to reflect the most current information or were not well documented.

Engineering Design Standard 1M-EDS-002, Design-Basis Review for Safety-Related Valves, Revision 7, dated June 5, 1997, is the source document for many of the assumptions and justifications used in the GL 89-10 program at Unit 1. Attachment 3 of the Standard, Justification for MOV Calculational Assumptions Used in the MOV Design Basis Component Level Reviews, contains the justifications for the assumed valve factors, stem friction coefficients (COFs), and rates of loading. The Attachment states that the values contained therein are to be used "...unless justification is provided for a specific MOV. That justification shall be provided in this EDS or the specific valve GL 89-10 calculation. The COF used in calculations shall be .20 or the measured COF from static testing plus at least a .02 margin." (The margin factor of .02 was based, in part, on in-situ test results.)

The inspector had the following specific observations regarding the calculations reviewed:

- Calculation S20.1-33V080, MOV 33-01R, Revision 3, dated April 7, 1997: (1) Per the EDS, the design-basis valve factor for this valve is 0.39. However, the calculation assumed a valve factor of 0.5. While the assumption is more conservative, the basis for the assumption is not documented. (2) The assumed stem COF is 0.077, referenced to the EDS and a note which states, "COF from field testing plus margin." The COF measured during the static test conducted on March 25, 1997, was 0.01. Using EDS guidance, the assumed COF would be 0.01 + 0.02 = 0.03. The cover sheet for revision 3 of the calculation states that the COF was revised per DER 1-97-0966. The DER states that, "Mechanical design will revise the MOV sizing calculation to incorporate a COF that will envelope the as-left minimum torque." The as-left minimum torque is a field setup "target" value that is calculated using a stem COF assumption. Thus, while the calculation

is consistent with measured valve performance and is acceptable, the justification for this design input is circular and has no basis in design. (3) The overall actuator gear ratio was referenced incorrectly back to the Limitorque Corporation actuator sizing and selection procedures.

- Calculation S20.1-40V050, MOV 40-05, Revision 4, dated July 28, 1997: (1) The assumed stem COF of 0.15 was referenced back to the EDS and Attachment C of the calculation. A static test of the valve conducted on April 1, 1997, yielded a COF of 0.14. Per the EDS, the assumed COF should have been 0.14+0.02 = 0.16: Attachment C concluded that there was "no adverse risk" in using a design COF of 0.15. The inspector agreed with the licensee's assessment, but noted that no justification was provided for reducing the degradation margin specified in the EDS. (2) The assumed valve factor of 0.6 was referenced to the EDS. However, the EDS specified a valve factor of 0.5. While conservative, the calculation did not provide a basis for the assumption. (3) The references in the calculation for core spray test line flow and torus water level elevation were reversed. (4) The value of 2200 gallons per minute assumed in the calculation of core spray pump head at minimum flow was not referenced. The licensee stated that the assumption was based on the minimum flow established by the operators in emergency operating procedure EO1-1, Attachment 4.

- Calculation S20.1-31V080, MOV 31-08, Revision 2, dated March 1, 1997: The COF of 0.18 was referenced back to the EDS. Per Attachment C of the calculation, the COF measured during a static test in 1995 (plus 0.02 for degradation) was 0.17; thus, use of a value of 0.18 was acceptable. The inspector noted, however, that the COF measured during a static test performed in 1997 was 0.17. Thus, the assumed COF was no longer conservative or consistent with the EDS.

c. Conclusions

No invalid or unreasonable design inputs were identified during review of the licensee's design-basis MOV calculations. However, the inspector concluded that greater attention to detail and more rigorous review of design inputs were warranted.

E1.4 Replacement of Recirculation System Sample Valves

a. Inspection Scope

The inspectors reviewed design change N2-96-029 for the replacement of recirculation system sample valves 2RCS*SOV104 and 2RCS*SOV105. Deficiency/Event Reports (DERs) pertaining to problems which arose following the valve replacements also were reviewed to evaluate the effectiveness of the modification, maintenance practices, and compliance with ASME Section XI Inservice Test (IST) and technical specification requirements for primary containment isolation.

b. Observations and Findings

During the Unit 2 refueling outage in 1996, the licensee replaced containment isolation valves 2RCS*SOV104 and 2RCS*SOV105 under design change N2-96-029. The valves were replaced to correct repetitive IST, 10 CFR 50, Appendix J local leak rate test, and valve position indication problems. The original valves were 3/4-inch globe valves with screw-in type, seal welded body-to-bonnet joint manufactured by Target Rock Corporation. The replacement valves were almost identical with the originals valves with the exception that the body-to-bonnet joints were sealed by gaskets and bolted. The licensee stated that bolted bonnet would allow access to the valve internals.

The inspectors reviewed the completed modification package and found that it met all applicable regulatory and licensee procedural requirements for ASME Code Section III and XI replacements and tests, seismic and environmental qualification reviews, safety classification evaluations, and post-modification testing. Affected drawings were updated and Final Safety Analysis Report sections were reviewed for revision. Design verification was performed in accordance with the requirements of 10 CFR 50, Appendix B, Criterion III, Design Control.

Following the refueling outage, the new valves displayed the same kinds of problems as had been experienced with the original valves, and extensive troubleshooting and repair efforts took place during a planned outage in June 1997. The problems documented in DERs 2-97-0479, 2-97-1742, and 2-97-2339 included local leak rate tests and IST stroke time failures, and valve position indication problems on both valves, and body-to-bonnet leakage of valve 2RCS*SOV105. At the time of the inspection, both valves were closed and de-activated for containment isolation purposes per Technical Specification 3.6.3, and downstream manual isolation valve 2RCS*V145 was shut. The inspector reviewed operator shift logs and IST surveillance records and verified that the licensee consistently had taken the appropriate steps to maintain the integrity of the containment penetration. Following the planned outage, a body-to-bonnet leak developed on valve 2RCS*SOV105. The leakage was stopped by retorquing the bonnet bolts. The inspector found that the bolts initially had been torqued to the appropriate value when the valve was relatively cool, and that the leakage occurred after the valve had heated up to normal operating temperature. The licensee's maintenance procedures did not contain guidance regarding rechecking fastener torque following heat-up of a component. The inspector concluded that because the licensee did not fully evaluate the cause of the leakage, an opportunity was missed to have identified this potential causal factor and to have enhanced current maintenance practices.

The inspector reviewed the root cause analysis performed under DER 2-97-1742. The licensee contacted other utilities and found that they had experience similar problems with this design solenoid-operated valve (SOV) in high temperature and pressure applications. The licensee concluded that tight clearances between the pilot and main disks (that were susceptible to buildup of corrosion products), a weak magnetic coil, and difficult-to-set limit switches made the valve design

inappropriate for the application. The licensee was preparing a new modification that specifies a valve that addresses these design shortcomings.

In reviewing DERs, the inspector noted that the licensee performed a formal evaluation that reconciled current plant operation with the valves closed against the FSAR Table 6.2-56, which describes the valves as normally open, and the technical specification requirement for continuous reactor coolant system conductivity monitoring. Per DER implementing procedure NIP-ECA-01, this Engineering Supporting Analysis is required to document the acceptability of an FSAR nonconformance that persists for longer than six months. The inspector reviewed the analysis and considered the conclusions to be acceptable.

c. Conclusions

Design change N2-96-029 met NRC and licensee design control requirements and appropriately considered applicable codes, standards, and specifications. However, the modification was not effective in resolving the problems for which it was implemented, and introduced an additional problem (pressure boundary leakage) as well. A more comprehensive evaluation of the original problems with valves 2RCS*SOV104 and 2RCS*SOV105, as was performed in 1997, could have identified the fundamental design deficiency earlier and prevented subsequent operational challenges. The licensee also missed an opportunity to have enhanced maintenance practices regarding torquing of fasteners.

E1.5 Review of Design Change Packages

a. Inspection Scope (37550)

The scope of this inspection was to review and assess engineering and technical support to plant operations by focusing on the design change process and its implementation. These activities were assessed to ensure compliance with NRC regulatory requirements, updated final safety analysis report (UFSAR), and the licensee's engineering procedures.

b. Observations and Findings

Elimination of Flex Hose (2RCS*HOSE40) from Valve 2RCS*HYV17B (Unit 2)

The inspector reviewed a 1997 operational event that led to design change (DC) No. N2-97-063, Revision 1. This design change eliminated flex hose 2RCS*HOSE40 on the reactor coolant system (RCS), permanently plugged the valve body side drain on valve 2RCS*HYV17B, and capped the associated drain pipe. As background, the inspector reviewed a similar 1991 event that led to design change PN2Y91MX011. The 1991 design change removed flexible hose 2RCS*HOSE44 and replaced it with a stainless steel pipe.

On March 30, 1991, a reactor coolant system pressure boundary leak was discovered. As part of the corrective action, the licensee implemented modification No. PN2Y91MX011. For this design change, the inspector reviewed the stress analysis to validate the as-built configuration of the stainless steel pipe, and noted that the results were within ASME Section III allowables that were documented in Section 3, Table 3.9A-2, Part II, of the Unit 2 FSAR.

For DC No. N2-97-063, the inspectors verified that the design bases and system description in the FSAR indicated that this drain line does not perform any function required for plant or valve operation, and that removal of this drain line did not affect the operation of the RCS as described in Section 5.4.1 of the FSAR. Further, the inspector verified that the plug material (ASME SA-479, type 316L stainless steel) used in the design change was compatible with the RCS valve body material. The selection of this material was performed using design parameter and material properties outlined in Section 5.2.3 (RCS materials) of the FSAR. A 10 CFR 50.59 Safety Evaluation was prepared in accordance with procedure NIP-SEV-01, Revision 3, dated June 17, 1997.

The new plugs were welded and inspected to ASME Code Section XI and Section III, Class 1 criteria and by approved Welding Procedures applicable to welding on the RCS pressure boundary. Further, the inspector verified that there was another drain outlet located at the bottom of the valve body which would enable maintenance personnel to drain the valve to open it for inspection. To assure leak tightness of the repair, the inspector verified that an inservice leakage test on all the fittings and associated welds was satisfactorily performed.

In summary, the DC No. N2-97-063, Revision 1 package addressing the corrective actions for flex hose leakage of 1997 was found to be acceptable. The appropriate requirements were established and documented in the design change package. The design change package was implemented in compliance with the licensee's design change process documented in procedure NEP-DES-01, Revision 2, dated January 20, 1997.

The inspector concluded that the evaluation of material compatibility was prepared using FSAR section 5.4.1 properties, and the installation and welding of the plug were performed in accordance with the sections XI and III of the ASME Code of record. DC No. N2-97-063 had a thorough 10 CFR 50.59 Safety Evaluation.

<u>Second Stage Feedwater Heater Nozzle Repair (Unit 1)</u>

The inspector reviewed Unit 1 design change No. N1-97-004. This design change replaced the nozzle of feedwater heat exchangers HTX 51-10 and HTX 51-12.

The licensee's analysis of the results of corrosion/erosion monitoring during previous refueling outage concluded that, at the current nozzle wall thinning rate, feedwater heat exchangers HTX-51-10 and HTX-51-12 may not have adequate wall thickness to operate to the next refueling outage without violating design minimum wall thickness requirements.

The inspectors reviewed key aspects of the modification to ensure the technical adequacy of the design change, with the following detail: Although the repairs to the shell side of the 2nd Stage Feedwater Heaters, including nozzles, shell, impingement plates and extraction steam piping are classified as nonsafety-related, the licensee proceeded with this design change as if it were a safety-related (pressure retainer component) design change using the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, "Unfired Pressure Vessel," 1965 Edition. This approach was conservative.

In terms of adherence to design control procedures, the package was found to be thorough, and the design input and design verification were prepared in accordance with procedures NEP-DES-05, Revision 2, dated July 17, 1997, and NEP-DES-07, Revision 1, dated July 2, 1997, respectively.

The inspector concluded that Unit 1 design change No. N1-97-004 that replaced the extraction steam nozzle on feedwater heat exchangers HTX 51-10 and HTX 51-12 was thorough and acceptable. The design change was consistent with the requirements of Section VIII of the ASME Boiler and Pressure Vessel Code of record, and the design change was prepared in accordance with approved procedures.

Polar Crane Modifications (Unit 2)

The licensee implemented Plant Change Request (PER) No. PC-0442-91 to modify the polar crane rails in the Unit 2 containment building. The PER documented the welding of ten rail splices. This welding task would minimize cracking of hold down clip studs resulting from excessive gaps between rail sections.

The inspectors reviewed the key components of the modification package and noted that the design activity to accomplish the specialized welding of the polar crane railing splices was performed in accordance with Bethlehem Steel Corporation procedure No. E.C. 2510561B, revised April 1972. The structural design inputs and the calculation supporting the welding of the splices were prepared following the guidelines and requirements of procedures NEP-DES-310-0, "Conceptual Design Input," Revision 6, dated November 22, 1991. Calculation SO53-7AB092, "Polar Crane Rail Welding Evaluation of Axial Thermal Loading" Revision 9, dated August 27, 1992, validated the as-built configuration of the polar crane rail splice joints. In this calculation the inspector verified that the thermal load at each connection is less than the material allowable and concluded that the welding of the crane rail splice joint was acceptable. 10 CFR 50.59 Safety Evaluation No. D92-251 evaluated the safety aspects for installation of complete splice welds for the circular rail of the reactor building polar crane. The safety evaluation was found to be thorough.

The inspectors concluded that Plant Change Request (PER) No. PC2-0442-91 documenting the welding of the polar crane rail splice joints was thorough. The supporting calculation justifying the as-built configuration was acceptable, procedure adherence was evident in the package, and the safety evaluation was of good quality.

Fibrous Insulation Replacement to Minimize Clogging of ECCS Strainers (Unit 1)

The inspector reviewed Design Document Change (DDCs) 1M00299 and 1M00060. These DDCs documented the replacement of fibrous insulation on several piping systems with reflective insulation.

Because of the potential for rapid clogging of Emergency Core Cooling System (ECCS) strainers from the combination of fibrous insulation with corrosion products, in Unit 1 interim response to NRC Bulletin 96-03, the licensee has a commitment to replace fibrous insulation installed within the containment drywell with reflective insulation. This corrective action was consistent with the Unit 1 design basis with respect to thermal insulation (Unit 1 Updated Final Safety Analysis Report, Chapter V, Revision 14)--i.e. use of a mix of fibrous and reflective insulation in these applications.

The inspectors verified that the safety significance of the DDCs was evaluated under licensee procedure NIP-SEV-01, "Applicability Reviews and Safety Evaluations," Revision 3, dated June 17, 1997, using an Applicability Review (AR) form. The AR was completed to determine whether the change affects the design or licensing basis of the plant, and concluded that return to the original design basis does not require a formal safety evaluation under 10 CFR 50.59.

In summary, DDCs 1M00299 and 1M00060 were issued as configuration changes, not as design changes. Because changing the type of insulation returned the unit to its original configuration, the licensee concluded that a safety evaluation was not needed. The inspectors concluded that the DDCs were acceptable.

Add Time Delay in RCIC Initiated Turbine Trip (Unit 2)

Unit 2 originally was designed such that the main turbine would trip immediately upon actuation of the reactor core isolation cooling (RCIC) system. If reactor power at the time of the actuation were greater than 35%, an automatic reactor trip also would occur. The licensee implemented design change SC2-0283-91 to preclude main turbine/reactor trips resulting from inadvertent or spurious RCIC initiations. The design change accomplished this function by replacing the turbine trip auxiliary relay with a time delay relay set for four minutes. The delay provides plant operators an opportunity to verify the cause of a RCIC actuation and take appropriate actions prior to receiving a main turbine trip.

The turbine trip was designed to protect the turbine from increased moisture level in the steam that resulted from RCIC injection into the reactor vessel head volume. An analysis performed for the licensee by General Electric Corporation stated that the increased moisture was of no long-term consequence provided that operation in this condition was limited to five minutes per occurrence and eight minutes per year of operation. The design change specified operating procedure changes to provide corrective actions for inadvertent RCIC system initiation and to monitor total accumulated main turbine run time under low steam quality conditions. The inspector noted that the post-modification test performed under implementing work order 94-05825-00 adequately verified the function of the new relay, and that the appropriate drawings were marked up for revision.

Neither the replaced relay nor the main turbine trip function are safety-related. However, since the modification affected the RCIC system logic diagram (Figure 7.4-1) in the Unit 2 FSAR, the licensee properly performed a safety evaluation (NSE 94-053) per 10 CFR 50.59, concluding that the change did not create an unreviewed safety question. The inspector found the safety evaluation to be thorough and acceptable.

Suppression Pool Air Space Temperature Increase (Unit 2)

Design change SC2-0109-94 increased the maximum allowable suppression chamber (wetwell) air space temperature from 110°F to 122°F under normal operating conditions, and revised associated alarm setpoints accordingly. The change was implemented as part of the corrective actions for DER 2-93-0914; because of leaking safety/relief valves, the previous temperature limit could not be maintained without use of suppression chamber sprays. The increased operating limit affected transient and accident analyses, emergency operating procedure calculations, station blackout evaluations, environmental qualification of equipment, structural analyses, and secondary containment drawdown and heating and ventilation system calculations. The inspector noted these analyses previously had been performed assuming a maximum wetwell air space temperature of 125°F. In all but one case, these analyses bounded the new 122°F limit.

The exception involved containment steam bypass capability. During a loss of coolant accident or main steam line break inside the drywell, steam released from the reactor coolant system is directed through vent pipes to the suppression pool, where it condenses. However, the potential exists for steam to bypass the suppression pool through drywell floor seams, downcomer and safety/relief valve pipe penetrations, vacuum breakers, floor and/or equipment drain piping and penetrations, and could increase containment pressure. The allowable bypass leakage is defined as the amount of steam that could bypass the suppression pool without exceeding primary containment design pressure (45 psig). Unit 2 Technical Specification 3.6.2.1.b limits bypass leakage in terms of a steam flow area of 0.054 square feet. The licensee calculated that 122°F was the highest air space temperature that could be permitted without requiring a change to the technical specification limit.

The inspectors reviewed safety evaluation 95-046 and observed that the evaluation properly considered the relevant design and licensing basis inputs, including; 10 CFR 50, Appendix A General Design Criteria, NUREG-0800 (NRC Standard Review Plan) for containment performance, 10 CFR 50.63 (Station Blackout Rule), 10 CFR 50.49 (Environmental Qualification), Emergency Operating Procedures, and FSAR Section 3.9 (design requirements for piping and supports). The impact of the design change on affected design calculations, procedures, and studies were evaluated thoroughly.

Install Fuse In-Line with TIS-201.9-60 (Unit 1)

Nitrogen supply temperature indicating switch TIS-201.9-60 had been treated as safety-related due solely to receiving power from safety-related power supply RPS11, Circuit No. 26, which also supplies power to the containment atmospheric monitoring system. To downgrade the switch to nonsafety-related, the licensee installed a safety-related fuse to provide electrical isolation. The inspectors found the design change package to be complete in that affected drawings were identified and properly revised, the fuse and fuse-holder were properly sourced, and an adequate 10 CFR 50.59 screen was performed.

c. Conclusions

The reviewed design change packages and associated 10 CFR 50.59 safety evaluations were acceptable. Pertinent design inputs from codes, standards, and NRC design criteria were specified and documented in the packages, and required design reviews were performed properly. Adequate post-modification tests verified proper implementation of the changes. Compliance with engineering administrative procedures was evident.

E1.6 Management of Plant Design Basis and Configuration Control

a. Inspection Scope (37550)

The scope of the inspection was to: (1) assess the degree to which the engineering organization maintained the design basis of the plant current; (2) verify that regulatory requirements and licensee commitments were properly implemented; and (3) ensure that the plant design conformed to the description documented in the FSAR.

b. Observations and Findings

Review of Configuration Management Procedures

"Nuclear Division Policy," Revision 10, dated July 17, 1997, sets forth the overall program for controlling configuration management at Nine Mile Point Units 1 and 2. This policy described the methodology that the licensee used to comply with regulatory requirements and guidelines, industry standards and practices, and commitments to regulatory agencies, as outlined in the operating licenses, the FSARs, and the technical specifications.

Nuclear Interface Procedure NIP-DES-01, "Determination of Design Control Applicability," Revision 0, dated November 2, 1995, applies to any proposed activities that may have potential design impact. This procedure covers critical aspects that are necessary to maintain an adequate plant configuration. For example, Sections 3.1 and 4.6 of the procedure screened the proposed activities to see if changes would be needed to drawings, specifications, data necessary to maintain plant configuration, system design basis documents, set point data sheets, and engineering programs and plans to ensure continued operation of the units within approved design limits.

The inspectors concluded that the licensee intends to maintain the configurations of both units with a series of administrative control procedures. These procedures would enable the licensee to identify plant configuration issues early and properly address these issues to ensure an accurate plant configuration.

Review of Configuration Management Implementation

At Unit 1, the licensee had completed a design basis reconstitution (DBR) program. The program included 21 safety-related systems. The inspectors selected the DBR of the Service Water System (SWS) (No. SDBD-502, Revision 1) for review. The Design Basis Document (DBD) contained a system overview; system requirements; a design description; and operation and maintenance considerations. The items were clearly and thoroughly described. The DBD also outlined the regulatory requirements applicable to the design modification process.

Through a review of a sample of as-built packages, the inspectors noted that the scope of the SWS walkdown included large and small bore piping, safety related supports for large bore piping and name plate information. The walkdown utilized isometric drawings, vendor drawings, piping and instrumentation diagrams (P&ID's) and piping and supports drawings.

The inspector noted that the SWS DBR was prepared in sufficient detail. The SWS walkdown identified 322 observations of which 321 have been closed. The one observation which remains open included a walkdown which was not performed because the area was inaccessible at the time closure of the final walkdown package.

At Unit 2, the inspectors verified that the licensee used deviation/event reports (DER) as an administrative tool to ensure that plant design changes are documented properly in drawings or procedures to maintain the plant design basis current. The verification was based on an observation of a demonstration by the licensee of a computerized database showing drawings and DERs affecting this drawings. The DER system also tracked configuration problems until they are resolved.

The licensee provided a historical explanation of the Unit 2 configuration management program. Prior to the initial testing and start up of the plant, the architect/engineer (A/E) controlled the design and configuration of the plant. After installation and testing were completed, the A/E provided flow diagrams to the licensee to document the as-built conditions. After turnover, the licensee verified the accuracy of the documentation through walkdowns, and converted the flow diagrams into piping and instrumentation diagrams (P&ID). The inspectors noted that the licensee has a computerized method of updating and maintaining the P&IDs.

c. Conclusions

Based on a limited review of the licensee configuration management described above, the inspectors concluded: 1) Unit 1 SWS DBD appeared comprehensive and well prepared and was readily available to design personnel; and 2) adequate mechanisms were provided for Unit 2 to maintain plant configuration through computerized design change document controls, documents affecting the configuration of Unit 2 were available and retrievable in a user-friendly manner.

E2 Engineering Support of Facilities and Equipment

E2.1 Resolution of Technical and Regulatory Issues and Engineering Backlogs

a. Inspection Scope (37550)

The scope of this inspection was to assess the extent and quality of engineering involvement in the resolution of technical and regulatory issues, including a review of licensee control of the engineering work backlog.

b. Findings and Observations

Engineering Involvement in the Resolution of Technical Issues

To assess engineering involvement in the resolution of technical issues, the inspector attended a Unit 2 pre-outage meeting to observe the licensee's discussions on the preparation of engineering activities scheduled for the next refueling outage. In this particular meeting, several engineering issues such as design changes for instrumentation and controls, reactor core stability, and the snubber reduction program were discussed in terms of status and schedules. The inspector noted that the design engineers present in the meeting were cognizant of the issues discussed. For example, the snubber reduction program was discussed in technical detail and in terms of additional activities scheduled for the next Unit 2 outage. The inspector noted that prior to embarking on the snubber reduction program (SRP), there were 802 Technical Specification Snubbers at Unit 2. As part of the SRP, engineering is proposing to remove 168 snubbers. One hundred and twenty (120) snubbers will be removed during the outage with the balance to be removed after the outage. The licensee added that when the program is completed the population of Technical Specification Snubbers would be reduced to 634.

Engineering Involvement in the Resolution of Regulatory Issues Affecting the Plant

Engineering is actively involved in the resolution of several regulatory issues. The inspectors selected one issue for a detailed review. On May 6, 1996, the NRC issued Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction (ECCS) Strainers by Debris in Boiling Water Reactors." The Bulletin requested licensees to implement appropriate procedural measures and plant modification to minimize the potential of clogging of ECCS suppression pool suction strainers by debris generated during Loss of Coolant Accident (LOCA). The bulletin requested all licensees to implement these actions by the end of the first refueling outage starting after January 1, 1997.

The first Unit 1 refueling outage following January 1, 1997, started on March 3, 1997. This short time span did not allow the licensee sufficient time to fully implement the Bulletin's requested actions. Therefore, the licensee requested a deferral from the NRC on October 4, 1996, that subsequently was approved on June 19, 1997.

The licensee's deferral request also provided its initial response to Bulletin 96-03. The required 180-day Bulletin response was provided on November 4, 1996. The licensee response detailed a series of compensatory actions that would be implemented until modifications could be installed during the 1999 refueling outage. These compensatory actions included torus dislodge, vent header and torus down comer visual inspections, forward flow exercise on the raw water inter-tie check valves, a torus agitation test, thorough cleaning of the drywell, reduction of fibrous insulation, and containment coating inspections.

Through interviews with the cognizant personnel the inspectors noted that in addition to these site specific actions, the licensee remains aware of the current industry trends and initiatives through direct involvement with the BWR owners group. Further, the licensee plans to install high capacity suction strainers no later than the 1999 refueling outage.

Key Aspects of Backlogged Engineering Work

One of the key aspects of licensee controls of engineering backlogs was through DER disposition and DER implementation. To determine the backlog of engineering work, the inspectors interviewed the engineering personnel and reviewed the latest engineering performance indicators on the implementation of DERs in the Mechanical, Electrical and Structural Engineering Departments, including the Project Management and Plant Support Departments. The inspectors noted in the performance indicators for both Units 1 and 2, that the backlog for DERs to be dispositioned and to be implemented had been increasing gradually, with the exception of structural engineering discipline, which had not been increasing. The design change backlog for Units 1 and 2, which included temporary modifications, simple design changes, and major design changes, also was discussed with the engineering managers who determined that the design change backlog remained high, but was manageable at the present.

In terms of qualitative assessment of the DERs, the inspectors reviewed a licensee self-assessment which indicated that DER trend data showed that no improvement in error rates occurred during last year. This data, coupled with the data for the current year, indicates that corrective actions taken to date have not been effective in reducing the number of errors attributable to engineering. The licensee initiated DER C-97-2027 to address this issue and to find solutions to reduce engineering errors in the DER process.

c. Conclusions

Based on observations made during a Unit 2 pre-outage meeting with follow-up interviews with engineering personnel, the inspectors concluded that the Unit 2 design engineers in the meeting were cognizant of the technical issues supporting the preparations for the next refueling outage of Unit 2.

Based on a review of Unit 1 licensee response to NRC Bulletin, 96-03, "Potential Plugging of Emergency Core Cooling Suction (ECCS) Strainers by Debris in Boiling Water Reactor," the inspectors concluded the Unit 1 design engineers were actively involved in the resolution of NRC Bulletin 96-03.

While manageable, the engineering backlog of DERs at Units 1 and 2, with the exception of the structural engineering discipline, is gradually increasing. The licensee has critically self-assessed the quality of engineering dispositions of DERs, and is seeking ways to improve the engineering performance in this area. The inspectors concluded that initiatives such as this self-assessment appear appropriate.

E2.2 Deficiency/Event Reports

a. Inspection Scope

The inspectors reviewed a sample of recent deficiency/event reports (DERs) to assess the quality of engineering support of operations at Nine Mile Point.

b. Observations and Findings

Increased Control Rod Drive Speed - Unit 1

DER 1-97-1433 was initiated when the licensee determined during pre-startup testing that the withdrawal speed of control rod drive (CRD) 06-19 was not within the nominal limit of three inches per second (± 20%) specified in FSAR Sections IV.B-6.2, Reactor Design, and X.C-2.0, Control Rod Drive Hydraulic System, and General Electric Specification GEK-724, Control Rod Drive System for Nine Mile Point Nuclear Station. The slowest speed attainable was four inches per second. The licensee declared the CRD inoperable and maintained it fully inserted pending completion of a safety evaluation for the higher speed and to operated the rod at reduced drive water pressure. Through troubleshooting, the licensee determined the cause to be leakage past a degraded CRD piston seals to the reactor vessel. The inspector noted that CRD seal degradation is normal at boiling water reactors, that the condition did not affect the scram time of the rod, and that the licensee scheduled the CRD for refurbishment during the next refueling outage (RFO15).

General Electric Corporation (GE) provided an analyses that concluded that operation during startup with a single CRD withdrawal speed up to five inches per second would not exceed the peak fuel enthalpy design criterion of 170 calories/gram. Thus, the margin of safety of 280 calories/gram evaluated in the bounding rod drop accident analysis would not be reduced. During power operation, GE concluded that the condition's effect on the minimum critical power ratio safety limit was negligible, and bounded by existing analyses. The licensee revised operating procedure N1-OP-5, Control Rod Drive System, to provide administrative controls for the operation of CRD 06-19, consisting of reducing drive water pressure to 200 psid (vice normal differential pressure of 250 to 270 psid).

The inspector reviewed Safety Analysis Section XV of the Unit 1 FSAR, and the Unit 1 technical specifications and verified that the licensee had considered the transients and license requirements relevant to the condition. The licensee's 10 CFR 50.59 safety evaluation for the operating procedure change was performed properly. The inspector found the licensee's disposition of the DER to be acceptable.

Conflicting Technical Specification Requirements - Unit 2

DER 2-97-0312 documented the licensee's discovery that: (1) the turbine stop valve (TSV) position setpoints for the reactor protection (RPS) and main condenser low vacuum trip bypass functions are incompatible, and (2) automatic action to fully enable the low condenser vacuum isolation logic does not occur in operating modes two and three when only one TSV is opened or a single low vacuum bypass key lock switch is placed in the "normal" position.

Technical Specification (TS) Table 3.3.2-1.1.e defines the channel operability requirements for the low condenser vacuum isolation signal for the main steam isolation and main steam line drain valves. The logic is required to be operable in operating modes one, two , and three when any one TSV is greater than 90% open and/or when the key locked condenser low vacuum bypass switch is open (in the "normal" position). TS Table 2.2.1-1.9 specifies the reactor protection system TSV trip setting to be less than or equal to five percent closed (95% open). However, since both functions are initiated by the same TSV limit switch, a conflict exists.

The licensee determined that during normal operation, the four TSVs are opened as a group, and that procedures require that all four of the low vacuum bypass switches be in the "normal" position prior to opening the TSVs. Consequently, the low vacuum trip logic is enabled when needed to assure that the main condenser and turbine exhaust hood are protected from overpressurization. Operators were informed of the limits of the automatic control logic and that these manual actions were needed to ensure that the protection is always in effect as required by the system design basis. A design change also was initiated to change the low condenser vacuum isolation trip bypass setpoint.

The inspector concluded that the licensee's corrective actions were appropriate. The inspector also considered that the licensee's identification of this subtle inconsistency indicated a good questioning attitude regarding the plant licensing basis.

Deficiency/Event Report Dispositions - Unit 2

Unit 2 engineering personnel performed a self-assessment of DERs to assess the quality of dispositions provided by the technical support organization. Twenty-one DERs initiated during the latter half of 1996 were reviewed to assess corrective actions against the causes identified in the DERs. The licensee concluded that approximately half of the DER dispositions were deficient (i.e., did not meet department expectations) in that the proposed corrective actions were too narrowly focused to prevent recurrence of the conditions. DER 2-97-0225 was initiated as an adverse trend to evaluate the root cause of the condition.

The licensee's root cause evaluators determined that management expectations were not well defined or understood causing the DER validation process to be performed and documented inconsistently. This resulted in subjective and/or incorrect causal determinations. The results of the DER disposition were disseminated to the Unit 2 Technical Support staff.

The inspector concluded that the licensee's assessment of DER quality was self-critical, and that initiation of a DER and performance of a root cause evaluation was appropriate.

c. Conclusions

A control rod drive speed problem was resolved safely and appropriately by Unit 1 Engineering. Identification of a subtle design and technical specification discrepancy at Unit 2 indicated a good questioning attitude. Unit 2 Engineering performed a self-critical assessment of DER disposition quality.

E5 Engineering Staffing and Training

E5.1 Engineering Staffing and Training Overview

a. Inspection Scope (37550)

The inspectors reviewed the engineering staffing and the engineering training program at Nine Mile Point to determine whether the licensee had adequate engineering staff to support the operations at both units and whether a quality training program was provided to the engineering staff.

b. Observations and Findings

Staffing

The inspectors interviewed the four managers of Units 1 and 2 design engineering and technical support regarding their staffing situations. Each group had several unfilled positions. They continued to use contractors and consultants to perform job functions as needed. They stated that their groups would be fully staffed by the end of this year, and that their goal was to minimize or eliminate the use of contractors.

The fuel and analyses department, which included the EQ group, was headed by a new general supervisor, who took over the position only several weeks ago. The department had a staff of about 25, including five contractors, with six unfilled positions. The general supervisor stated that all vacant positions would be filled by the end of this year.

The EQ group, headed by a new EQ program manager, usually had a staff of four EQ engineers (including the program manager). Recently, the former EQ manager and a mechanical EQ engineer left Niagara Mohawk, and a third EQ engineer was on-loan to the FSAR review group since April 1997. An engineer (contractor) with some EQ background was transferred to the EQ group recently from the Quality Assurance group. Because of this loss of three (out of four) experienced EQ personnel and the new EQ program manager was still new in his position, compounded with fact that certain EQ activities performed by the former EQ personnel were not properly documented (see Section E1.2), the licensee did experience difficulties during this transition period. The licensee hired two experienced EQ consultants from GLS Enterprises, Incorporated, in Huntsville, Alabama, to assist the EQ program manager to handle routine EQ activities and to resolve EQ issues. Two vacant positions in the EQ group still needed to be filled. The licensee stated that the offer for one of the positions had already been accepted and that they expected the EQ group to be fully staffed by the end of this year.

During the inspection, the vice president of engineering told the inspectors that they were in the process of establishing a pool of junior engineers to supplement existing staff. He stated that this effort would ease the problem of rising turnover in experienced engineering staff. The licensee expected to hire 15 entry-level engineers in the areas of EQ, probabilistic risk assessment (PRA), fire protection and fuel, and system and design engineering. The "Job Requisition" for this effort had already been approved by senior Niagara Mohawk management, and the licensee expected to start this effort quickly.

Training

The engineering training program at Nine Mile Point was described in Nuclear Training Procedure NTP-TQS-404, "Training for Engineering Support personnel," Revision 7, dated March 18, 1997. The inspectors' review of this procedure found it similar to the previous revision. Three types of training were provided to engineering and technical staff (including design engineering and system engineering personnel): (1) orientation training, (2) position-specific training, and (3) elective training. The orientation training, about six weeks in all, covered basic engineering courses and regulatory requirements, such as reactor theory and 10 CFR 50.59 safety evaluations. This training was required of all engineering personnel at the time of initial employment. The position-specific training was tailored to specific need by the employee's supervisor. The elective training was chosen by the staff and approved by the manager.

The engineering training department recently, on June 10, 1997, provided an EQ training (an eight-hour course) to 14 technical staff. The instructor was James Gleason from GLS Enterprises, Incorporated. The inspectors reviewed the lesson plan and found the course well organized and containing good EQ topics.

The inspectors reviewed the list of the core course and other technical course offered and found that there were abundant courses for both new-hired and existing staff. Niagara Mohawk also shared their training facility with FitzPatrick and Ginna to provide more flexible training for their staff. At the time of the inspection, there were two full-time engineering instructors, providing the training needs of about 250 technical and managerial staff.

c. Conclusions

Vacant positions existed in the design engineering, including the Fuel and Analyses groups, and the technical support at both units. Contractors were being used to offset the impact of unfilled engineering positions. The EQ group did experience manpower difficulties due to a loss of three (out of four) experienced EQ personnel. However, the licensee was able to hire two experienced consultants to ease the difficulties during the transition period. The licensee expected to have all vacant positions filled by the end of this year.

The inspectors also concluded that Niagara Mohawk engineering training program provided abundant technical courses to both new-hired and existing technical staff.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-410/97-05-07: This item pertained to the calculations of Unit 2 normally-energized (NE) Agastat relays at Unit 2. During an April 1997 inspection (IR 97-05) the inspector found that the licensee had used a calculation method that was not based on appropriate engineering judgement.

During this inspection, the licensee twice met with the inspector to discuss and clarify the findings documented in IR 97-05. The first discussion was held on August 12, 1997, between the inspector and a Niagara Mohawk electrical engineer and a Niagara Mohawk engineering supervisor. The inspectors stated that the negative assessment in the report was based on the review of a licensee document, Nuclear Engineering Report NER-2E-007, "Agastat Relay Service Life," Revision 1, dated March 10, 1997, that was provided to the inspector's review during the April inspection. The second discussion was held on August 28, 1997, between the inspector and three Niagara Mohawk personnel (the vice president of engineering, Unit 2 engineering manager, and a Unit 2 engineering supervisor). Another item was also discussed in this meeting as discussed in Section X1 of this report.

During the first meeting (discussion), the licensee engineer also wanted to clarify two items. The first item was that the NE safety-related Agastat relays, should have been NE safety-related Agastat relays with active safety functions. The reason for this clarification was that the licensee had divided its Agastat relays into four categories, (1) normally energized safety-related relays with active safety functions, (2) normally energized safety-related relays with passive functions, (3) normally de-energized safety-related relays, and (4) nonsafety-related relays. The second item was that the seven relays mentioned in the NRC report with 40 years service life as an example were normally de-energized or intermittently energized. These were listed as normally energized for conservatism, but a note explaining this fact was not added to the list. Four of these relays would be moved (in the new revision of NER-2E-007) from the normally energized list to the normally de-energized list. The inspectors acknowledged the clarifications, but noted that they did not change the general conclusion of the report.

As mentioned in the April 1997 inspection report, following that inspection, the licensee contracted PECo laboratories to perform more tests and to establish a more reliable methodology to determine relay coil temperatures based on measured ambient temperatures.

The PECo test results and the new calculation method were documented in Unit 2 Nuclear Engineering Report NER-2E-007, "Service Life of Agastat Relays Located in Mild Environments," Revision 3, dated August 12, 1997. Attachment 3A listed the NE safety-related Agastat relays with active safety functions. The inspectors' review of this list confirmed that: (1) all NE relays in this category that had not been previously replaced were scheduled for replacement by the next refueling outage (June 30, 1998), and (2) for the second round replacement, the relays in this category would be replaced every 10 years.

The inspectors considered the new calculations and new replacement schedule acceptable. Therefore, this item is closed.

E8.2 <u>(Closed) Followup Item 50-220 & 410/95-11-05</u>: This item pertained to the revision of MOV switch setting and capability calculations based on industry and site-specific test data. This item was opened pending NRC review of the licensee's evaluation of load sensitive behavior and stem friction coefficient data. As discussed in Section E1.3, the licensee's analyses of these factors was acceptable for GL 89-10 program closure. Therefore, this item is closed.

E8.3 (Closed) Followup Item 50-220 & 410/95-11-06: This item pertained to the analysis of dynamic test data for all valves in the GL 89-10 program. This item was opened pending verification that the licensee had accounted for additional diagnostic equipment uncertainty when open direction thrust measurements exceeded the sensor calibration range, and review of the licensee's method of extrapolating dynamic test date to design-basis conditions. Liberty Technologies' Customer Service Bulletin (CSB) 31 detailed steps needed to correct thrust measurements that are not within the diagnostic equipment sensor's calibration range. The inspectors reviewed the licensee's diagnostic test procedure and dynamic test traces and verified that CSB 31 had been implemented correctly. As discussed in Section E1.3 of this report, the licensee adequately justified its linear extrapolation of dynamic test data. Therefore, this item is closed.

E8.4 (Closed) Followup Item 50-220/95-11-07: This item pertained to the implementation of a MOV performance trending program at Units 1 and 2. This item was opened pending the licensee's completion of an MOV performance trending report at Unit 2 pursuant to procedure N2-MAP-MAI-0302, Trending of MOV Performance and Review of MOV Diagnostic Test Data. The procedures requires the ongoing collection of performance data and generation of a comprehensive report of findings every two years. The inspector reviewed Deviation/Event Report (DER) 2-96-2685, dated July 1997. The DER contained the first MOV trending report written for the Unit 2 MOV program. The report discussed: (1) problems identified during static testing, such as valve/disk guide deterioration, actuator-to-valve misalignment, actuator component degradation, and loose stem nut lock nuts and worn stem nuts; and (2) general performance trends identified during diagnostic testing, such as the effectiveness of past stem lubricant replacement, an apparent increase in thrust output over time for the same torque output (indicating an improved stem friction coefficient), higher wedge pullout forces for non-vertically mounted MOVs, and better than predicted HBC efficiencies. In addition, the report contained a review of corrective maintenance findings and corrective actions from 1985 to 1997, and specific potential problem areas highlighted by a review of diagnostic test data. The licensee identified, for example, that a resurgence of incidents had occurred involving damaging actuator electrical components when removing and reinstalling tight fitting covers on small actuators. Worn or broken gears and bearings, cyclic loading and "problematic" coefficients of friction were found to be more prevalent on SMB-000 actuators. The licensee also initiated DERs on the adverse trends identified during the review.

A trend report had not been prepared at Unit 1 at the time of the inspection. The Unit 1 trending procedure is identical to that implemented at Unit 2, and Unit 1 personnel were familiar with the Unit 2 report. The inspector concluded that the implementing procedures and the comprehensive Unit 2 trend report fully satisfied the intent of GL 89-10. Therefore, this item is closed.

E8.5 (Closed) Followup Item 50-220/95-11-02: This item pertained to Unit 1 GL 89-10 program scope. Justification for removing two plug-type containment isolation valves from the Unit 1 MOV program is discussed in Section E1.2 of this report. Attachment 2 of Engineering Design Standard Number 1M-EDS-002, Design Basis

Review For Safety-Related Motor Operated Valves, contained the licensee's justifications for removing other safety-related valves from the scope of the program. The inspector reviewed the Attachment and found the positions stated therein to be consistent with the guidance provided in GL 89-10 and its supplements. Therefore, this item is closed.

E8.6 (Closed) Followup Item 50-220/95-11-03: This item pertained to the review documentation of MOV structural "weak link" limits. This item was opened to review the translation of MOV structural limits into the design-basis thrust calculations. The inspectors reviewed the weak link and thrust calculations for several core spray system valves and verified that the thrust "windows" used to set up the valves were less than the lowest weak link limit. The inspector found that the calculations were complete, had been reviewed and verified in accordance with the licensee's design control procedures, and were consistent with each other. The inspector concluded that structural limits were being considered properly in establishing valve torque switch settings. Therefore, this item is closed.

E8.7 (Closed) Followup Item 50-220/95-11-04: This item pertained to the establishment of a program to periodically verify the design-basis capability of safety-related MOVs. The licensee had not finalized its plans for periodic verification of MOV capability. The current Unit 1 MOV Program Plan Description stated that each MOV will be diagnostically tested under static conditions every five years or three refueling outages unless longer intervals can be justified and that dynamic testing would only be performed as required by post-maintenance test procedures. The licensee plans to implement the program being developed by the BWROG. The plan would include a mix of static and dynamic diagnostic tests conducted at a frequency determined through consideration of MOV design capability and risk significance. Further NRC review of this matter will be conducted under GL 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves. Therefore, this item is closed.

E8.8 (Closed) Unresolved Item 50-220/93-22-01: This item pertained to pressure locking and thermal binding of safety-related gate valves. This item was opened because initial MOV design reviews had not considered the effects of pressure locking in establishing the operating requirements of susceptible valves. Subsequently, the licensee performed susceptibility evaluations and capability reviews of the safety-related MOVs at Unit 1 pursuant to NRC Information Notice 92-26, Pressure Locking of Motor-Operated Flexible Wedge Gate Valves, and GL 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves. In its response to the GL, dated February 13, 1996, the licensee identified six susceptible valves in the core spray system. The valves were modified to preclude the condition. In May 1997, the licensee identified two normally open core spray system valves (40-02 and 40-12) and four normally open emergency condenser system valves (39-07R, 39-08R, 39-09R, and 39-10R) that could become pressure locked if a loss of coolant accident were to occur while the valves were closed for periodic surveillance testing. The licensee notified the NRC of the condition regarding the core spray valves per 10 CFR 50.72, and documented the condition in Licensee Event Report 50-220/97-05. Regarding the emergency condenser valves,

the licensee initially determined based on 1994 test data that sufficient motor-actuator capability existed to overcome postulated pressure locking forces. However, the conclusion was changed as a result of new data obtained during testing in 1997. Procedures have been revised to declare the associated emergency condensers inoperable whenever the valves are closed, and the licensee is considering modifications to eliminate the constraint. The licensee modified the core spray valves during the 1997 refueling outage. The inspector concluded that the licensee had addressed the issue of pressure locking adequately for GL 89-10 program closure. The NRC will complete its evaluation of pressure locking and thermal binding at Unit 1 under GL 95-07. Therefore, this item is closed.

E8.9 (Closed) Unresolved Item URI 50-220/96-07-07: NRC Integrated Performance Assessment Process (IPAP) Inspection Report (IR) 50-220/96-201 documented a Unit 1 licensee review of Deviation/Event Reports (DER) 1-95-2051 and 1-95-1075 that identified configuration control concerns in electrical drawings, because design change requests (DCR) affecting drawings initiated years ago had not been entered into the configuration control database. Subsequently, this configuration control concern was documented in this unresolved item.

DERs 1-95-1075 and 1-95-2051 identified a number of DCRs affecting drawings that were not reviewed and issued by engineering to incorporate the changes in the corresponding drawings. Specifically, a total of 1,708 electrical drawings either did not incorporate the DCRs or did not have the DCRs posted against them, out of this population, 133 electrical drawings were critical drawings that plant operations and other plant personnel used in plant activities. To address this issue, the licensee developed an action plan to review, set priorities, and resolve the discrepancies covered by the two DERs. Under this action plan, each proposed drawing change was evaluated for safety significance and plant impact and then resolved by incorporating the changes into the corresponding drawings.

The inspectors verified that the DCRs identified in DERs 1-95-2051 and 1-95-1075 were assessed for safety significance and resolved accordingly. The inspectors verified that DERs 1-95-2051 and 1-95-1075 were closed.

The inspectors also verified, through interviews with the originator of DERs 1-95-2051 and 1-95-1075, the Unit 1 engineering manager, and reviews of the licensee internal correspondence on this issue, that the following additional corrective and preventive actions were completed by the licensee.

- DERs 1-95-2051 and 1-95-1075 associated with this configuration control issue were reviewed by members of the licensee's Senior Management Team, Engineering and Operations to verify that there were no nuclear safety issues.

- The licensee established a prioritization scheme to address the backlog of drawing updates and design resources accordingly. The establishment of priorities was evident in the closure of DERs 1-95-2051 and 1-95-1075.

- Procedures for reviewing DERs affecting drawings and drawing updates were in place to assure that configuration discrepancies were evaluated for plant impact at the time they are identified.

The inspector verified that each document listed in the attachments to DERs 1-95-2051 and 1-95-1075 were addressed individually and the affected drawings were properly updated, thereby, closing the DERs in their entirety. Further, the inspectors verified that the licensee had procedures with emphasis on maintaining the plant configuration by keeping track of the DERs affecting the configuration of the plant. Therefore, this unresolved item is closed. However, the failure to distribute 133 updated critical drawings for the use by operations and other personnel constituted a violation of 10 CFR 50, Appendix B, Criterion VI, "Document Control," which required documents affecting quality to be delivered to and used at the location where the prescribed activity is performed. However, this licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy."
(NCV 50-220/97-09-05)

E9 FSAR Reviews

A recent discovery of a licensee operating their facility in a manner contrary to the updated final safety analysis (UFSAR) description highlighted the need for a special, focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions.

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSARs that related to the areas inspected, including Unit 2 FSAR section 3.11, that pertained to environmental design of mechanical and electrical equipment. The inspector verified that other reviewed sections of the FSAR wording were consistent with the observed plant practices, procedures and/or parameters.

X1 Management Meetings

A management meeting was held on August 27, 1997, at the request of the Vice-President, Nuclear Engineering, to discuss the fire-induced hot shot issues documented in NRC Inspection Report 97-05. The attendees of this meeting was list in the "Partial List of Persons Contacted" section. The licensee stated that they reported their hot-shot-issue findings to the NRC even though they were not certain whether a regulatory noncompliance was involved. The licensee stated their opinion that the NRC's treatment of these findings as apparent violations was counterproductive. NRC management informed the licensee that hot-short issues remained under review by the Office of Nuclear Reactor Regulation.

Another management meeting was held on August 29, 1997, also at the request of the Vice-President, Nuclear Engineering, to discuss potential errors in IR 97-05. The attendees included the Vice-President, Nuclear Engineering, Unit 2 Engineering Manager, Unit 2 Electrical Engineering Supervisor, and the inspector involved in IR 97-05 inspection. Two items were discussed. The first item was that the thermal overload bypasses discussed in Section E1.2b should refer to Unit 1 rather than Unit 2. The inspector agreed with this

observation. However, the error did not affect the general conclusion of the report, no further actions was required. The second item pertained to the NRC's assessment of Agastat relay service life calculations at Unit 2. The Vice-President, Nuclear Engineering, asked whether the NRC conclusion regarding unsound engineering judgement applied to the calculations or the calculation method. The inspector stated that the assessment pertained to the calculation method.

X2 Exit Meeting

The inspectors met with the licensee personnel at the conclusion of the site inspection on August 29, 1997, and summarized the scope of the inspection and the inspection results. No proprietary materials were reviewed during this inspection. The licensee acknowledged the inspection findings at that meeting.

The inspectors amended the exit meeting in two telephone calls on September 11, 1997 and October 17, 1997, to Mr. G. Gresock of Niagara Mohawk. The inspectors stated that: (1) after reviewing additional documents provided by the licensee, one more example of violation of 10 CFR 50, Appendix B, Criterion V, was identified as discussed in Section 1.2b of the report; and (2) a noncited violation of 10 CFR 50, Appendix B, Criterion VI, was identified as the result of the closure of URI 50-220/96-07-07, as discussed in Section E8.9 of the report.

The inspectors amended the exit meeting again in another telephone call on October 21, 1997, to Mr. D. Baker of Niagara Mohawk. The inspectors stated that: (1) an inspection followup item (IFI) was added to track the licensee's commitment for Unit 2 MOV valve factor justification as discussed in Section E1.3.4 of the report; and (2) the inspection followup item for the feedwater isolation valve thrust requirements was changed to an unresolved item as discussed in Section E1.3.2.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Yaeger, Acting General Supervisor Fuels and Analysis
T. Page, Licensing Engineer
G. Gresock, Licensing Engineer
D. Wolniak, Licensing Manager
D. Baker, Licensing Supervisor
C. Burge, Information Management
L. Dick, QA Supervisor
M. Shanbhag, MATS
K. Vara, Technical Support Manager
C. Ware, Unit 2 Chemistry Manager
R. Dean, Unit 2 Manager Engineering
K. Dahlberg, Plant Manager, Unit 2
R. Sylvia, Executive Vice President and Chief Nuclear Officer
P. Konu, EQ Program Manager
R. Tessier, Manager, Training
B. Burtch, Manager, Nuclear Command
R. Hall, Director, HRD
C. Beckman, Manager, QA
S. Duty, Maintenance Manager, Unit 1

* C. Terry, NMPC, Vice President, NSAS
* G. Gresock, NMPC, Licensing
* J. Conway, NMPC, Vice President, Nuclear Engineering

NRC

* W. Ruland, Chief, Electrical Engineering Branch, DRS
* L. Cheung, DRS
* B. Norris, Sr. Resident Inspector, Nine Mile Point
* D. Hood, NRR/PD1-1
* J. Wiggins, Division Director, DRS
R. Skokowski, Resident Inspector

* Denotes attendees in the August 27, 1997 management meeting.

INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 92700	On-Site Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92903	Followup - Engineering
TI 2515/109	Inspection Requirements for Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-410/97-09-01	EEI	RWCS isolation valves susceptible to fire-induced hot short
50-410/97-09-02	VIO	Use of unapproved procedure for EQ program activities
50-410/97-09-03	VIO	No written instructions or procedures for EQ program activities
50-220/97-09-04	URI	Evaluation of feedwater isolation valve thrust requirements
50-220/97-09-05	NCV	Electrical drawing updates
50-410/97-09-06	IFI	Valve factor justification

Closed

50-220&410/95-11-05	IFI	Revise switch setting calculations
50-220&410/95-11-06	IFI	Analyze dynamic test data
50-220/95-11-07	IFI	Develop MOV failure analysis/trending program
50-220/95-11-02	IFI	GL 89-10 program scope
50-220/95-11-03	IFI	MOV weak link analysis
50-220/95-11-04	IFI	Periodic verification plan for GL 89-10
50-220/93-22-01	URI	Pressure locking and thermal binding evaluations
50-410/97-05-07	URI	Agastat relay service life calculation
50-220/96-07-07	URI	Electrical drawing updates
50-220/97-09-05	NCV	Electrical drawing updates

LIST OF ACRONYMS USED

ANSI	American National Standards Institute
AOV	Air-operated valve
ASME	American Society of Mechanical Engineers
BWROG	Boiling Water Reactors Owners Group
CFR	Code of Federal Regulations
COF(s)	Coefficient(s) of friction
CSB	Customer Service Bulletin
DBD	Design basis document
DBR	Design basis reconstitution
DCR(s)	Design change request(s)
DDC(s)	Design document change(s)
DER	Deficiency/Event Report
ECCS	Emergency core cooling system
EPRI	Electric Power Research Institute
EQ	Environmental qualification
EQD	Environmental qualification document
EQEDC	Environmental qualification environment design criterion
EQML	Environmental qualification master list
EQMR	Environmental qualification maintenance requirements
FSAR	Final Safety Analysis Report
GL	Generic Letter
gpm	gallons per minute
IPAP	Integrated Performance Assessment Process
IR(s)	Inspection report(s)
ISR	Information service request
IST	Inservice test(ing)
LER(s)	Licensee Event Report(s)
LOCA	Loss of coolant accident
MEL	Master equipment list
MOV(s)	Motor-operated valve(s)
NMP	Nine Mile Point
P&ID(s)	Piping and instrumentation diagram(s)
PPM	Performance prediction model
PPP	Performance prediction program
psid	pounds per square inch differential
PRA	Probablistic risk assessment
RCS	Reactor coolant system
RWCS	Reactor water cleanup system
SQEW	System component evaluation work
SRP	Snubber reduction program
SWS	Service water system
UFSAR	Updated Final Safety Analysis Report