U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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Report No:	50-369/96-11, 50-370/96-11
Licensee:	Duke Power Company
Facility:	McGuire Generating Station, Units 1 & 2
Location:	12700 Hagers Ferry Rd. Huntersville, NC 28078
Dates:	December 1, 1996 - January 11, 1997
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ENCLOSURE 2

EXECUTIVE SUMMARY

McGuire Generating Station, Units 1 & 2
NRC Inspection Report 50-369/96-11, 50-370/96-11

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection.

Operations

- Operator performance in recognizing a feedwater valve hydraulic fluid leak at the valve controller was good. Operators placement of the unit in a favorable operating condition to preclude a more serious transient was timely. The subsequent return to rated power was conducted with good attention to plant parameters and personnel safety (paragraph O2.1).

- Review of the monitoring, root cause determination, and prevention of component mispositionings concluded that the licensee's program was properly focused and receiving good management attention. However, while the number of significant mispositioning events (as defined by the licensee's program) has decreased, the overall number of mispositioned components was considered high indicating further improvements could be made with continued management focus (paragraph O7.1).

Maintenance

- Initial use of automatic welding equipment to weld secondary pipe welds was unsuccessful and the licensee stopped work until the problem was identified and corrected. Welding of dissimilar metal welds manually was satisfactory. Radiography, material control, and personnel training within the areas inspected was adequate (paragraph M1.2).

- Licensee response to the failure of 1RN171 during testing was good. Proper compensatory actions were taken to maintain system operability until an appropriate temporary modification was developed, evaluated, and implemented (paragraph M2.2).

- Preparation and coordination of Boraflex testing was good. Previous test results coupled with current testing indicated no gross boraflex degradation had occurred.

Engineering

- The licensee met the intent of GL 89-10 in verifying the design basis capabilities of their MOVs. Several weaknesses were identified. Examples included the limited quantity or quality of data that was used to establish the capabilities of several groups of MOVs and the licensee's application of PRA in establishing MOV operability. The licensee initiated actions to correct the more significant weaknesses and IFI 50-369, 370/96-11-01 was identified to track their completion.

ENCLOSURE 2

Strengths were identified which included strong management and personnel support, application of state of the art technology, leadership in addressing industry problems, and the detailed packages of data and evaluations that were developed for each valve group (paragraph E1.1). A related NCV was identified involving improper deferrals of periodic MOV lubrications (NCV 50-369, 370/96-11-04, Section E8.3).

Based on the NRC staff's review of the McGuire GL 89-10 program and its implementation, and the actions established by the licensee in PIP 0-M96-3542, the NRC staff is closing its review of the GL 89-10 program at McGuire. The completion of these licensee actions will be assessed as part of the NRC staff's monitoring of the licensee's long-term MOV program (paragraph E1.1).

- A Violation was identified (VIO 50-369/96-11-02) concerning the installation of a temporary security fence to restrict access to the Unit 1 exterior valve vault. Installation of the security fence was not conducted in accordance with established station procedures (paragraph E2.1).

- The commercial grade dedication (CGD) process was effectively implemented at McGuire by the Procurement Engineering organization and was consistent with applicable regulatory requirements. Resolution of identified procurement problems was adequate. Self assessments provided adequate monitoring of station performance in CGD activities. An URI was identified (URI 50-369, 370/96-11-03) for further NRC review of the environmental qualification of Grinell hydraulic pipe supports (paragraph E4.1).

Plant Support

- The inspectors determined that the licensee effectively implemented a program for shipping radioactive materials required by the NRC and Department Of Transportation regulations (paragraph R1.1).

- Radiological facility conditions and housekeeping in radioactive waste storage areas were good. Material was labeled appropriately, and areas were properly posted. All exposures were below regulatory limits and the licensee was continuing to maintain exposures As Low As Reasonably Achievable (paragraph R1.2).

- A Violation was identified (VIO 50-369/96-11-05) for the failure to conduct a 10 CFR 50.59 written safety evaluation to provide the bases for the determination that a test not described in the FSAR did not involve an unreviewed safety question. This test involved the lowering of hydrazine levels in Unit 1 secondary systems which could have potentially impacted reactor power from the effects of the change on feedwater venturi fouling (paragraph R1.3).

- The licensee had maintained an overall high level of operability for radiation monitors in 1996 and was effectively tracking monitor performance. An Inspector Followup Item was identified (IFI 50-369/96-11-06) to track the closeout actions on the Problem Investigative Process associated with the solubilization of the Cobalt-58 (paragraph R2.1).

- The licensee had continued to maintain effective capabilities to perform environmental samples (paragraph R2.2).

- Radiation Protection technicians and Chemistry technicians were receiving an appropriate level of refresher training to enhance work activities (paragraph R4.0).

- The licensee was effectively conducting formal RP and Chemistry audits as required by Technical Specifications and completing corrective actions in a timely manner (paragraph R7.0).

- Review of an annual fire drill conducted in December 1996 concluded that the drill was conducted in a realistic and professional manner. The licensee's critique of the drill was candid and identified several issues, which, once resolved, should improve the plant and offsite agency response to a fire emergency (paragraph F5).

Report Details

Summary of Plant Status

Unit 1 began the inspection period at approximately 28% power while the licensee repaired a hydraulic leak at feedwater isolation valve 1CF26. Following repairs, the unit was returned to 100 percent power on December 3, and operated at power throughout the remainder of the reporting period.

Unit 2 operated at 100 percent power throughout the reporting period.

Review of UFSAR Commitments

While performing inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that were related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures, and/or parameters.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below. Operator monitoring of plant parameters for abnormal conditions was good, TS LCO compliance was maintained, and special attention was given to specific challenge areas such as freeze protection monitoring.

O2 Operational Status of Facilities and Equipment (71707)

O2.1 Hydraulic Fluid Leak at Feedwater Isolation Valve 1CF26

a. Inspection Scope

At the beginning of the inspection period, Unit 1 operated at approximately 25 percent power. Operators had reduced Unit 1 power in order to realign main feedwater flow from the D steam generator main feedwater nozzle to the upper feedwater (auxiliary feedwater) nozzle. The licensee had previously performed a rapid downpower in accordance with station abnormal procedures and had realigned feedwater flow to reduce the effects of an inadvertent closure of feedwater isolation Valve 1CF26 due to a previously identified hydraulic fluid leak at the valve controller.

Valve 1CF26 is located in the Feedwater System flowpath to the D steam generator main nozzle. Maintenance technicians were dispatched and identified a failed O-ring at the oil side accumulator fill valve.

Valve 1CF26 is a safety-related hydraulic isolation valve. The valve receives a signal to close on a Safety Injection, Low Tavg coincident with Reactor Trip, HI-HI doghouse water level, or HI-HI steam generator level.

b. Observations and Findings

With the unit stabilized at approximately 25% power, the licensee began repairs to correct the failed O-ring and re-establish normal feedwater flow to the D steam generator. During the troubleshooting and repair efforts, maintenance technicians found the accumulator fill valve loose enough to remove without the use of a wrench indicating insufficient torque. The licensee could not determine whether the insufficient torque was due to maintenance on the fill valve or a failure to verify torque following installation of the manifold block assembly during 1EOC10.

During the repair effort the licensee repaired or replaced several other components to enhance valve performance. During functional testing, the licensee determined that the associated hydraulic pump performance did not meet station acceptance criteria. The pump was replaced and tested satisfactorily. The licensee also contacted the equipment vendor prior to completion of the repairs to ensure that the repairs were adequate to prevent recurrence. The valve was stroke tested satisfactorily and returned to service. Feedwater flow was realigned to the main feedwater nozzle and the Unit 1 was returned to 100 percent power.

c. Conclusion

Although the licensee identified evidence of a loss of torque at the fill valve, no root cause was identified. The inspectors also concluded that Maintenance and Engineering response following identification of the hydraulic fluid leak was prompt and detailed. Operator performance in recognizing the controller malfunction and placing the unit in a favorable operating condition was also good. The subsequent return to rated power was conducted with good attention to plant and personnel safety.

O7 Quality Assurance in Operations (40500)

O7.1 Review and Control of Component Mispositionings

a. Inspection Scope

During the inspection period, the inspector reviewed the licensee's process for identifying, tracking, and improving the number of component mispositioning events.

b. Observations and Findings

The inspector reviewed implementation of the licensee's 5M process (McGuire Monthly Managers Meeting on Mispositionings) used to identify, track, and improve the stations' performance in the area of component mispostionings. The process has included a monthly meeting with multi-disciplined managers, to discuss the most recent mispositioning problems and events. The issues are then classified by component, type of activity, responsible group, and potential causes. From the monthly meetings, annual assessment reviews were performed, which provided recommendations from the 5M members to incorporate changes to improve the plant performance in the configuration control area. The inspector attended portions of 5M meetings, reviewed annual assessments of mispositioned components, and discussed the process with involved personnel.

At the December 1996 5M review, recent mispositioning problems were presented to the members by the responsible managers. The inspector considered that each individual problem was discussed in good detail and allowed members to appropriately classify the severity of the problem and agree or disagree on the assigned root cause. Managers presenting issues were well prepared and the 5M members exhibited a good questioning attitude. The inspector also reviewed several meeting minutes from other 5M meetings. The documents were detailed and contained specific action items incorporating process improvements made by the 5M members. The licensee's process incorporated detailed trending of the identified mispositionings, such that corrective actions could be taken for adverse trends in specific areas.

Mispositioning issues at McGuire are classified as mispositioning problems or events. Mispositioning problems which have impacted plant operations prior to discovery are classified as "events". Based on the available data for several years, the number of "events" has decreased from 11 in 1994 to 0 for 1996. Over this time, the number of mispositioning problems being reviewed has increased due to a lower threshold for reporting misposition type problems. This further indicated to the inspector that the overall "events" have been reduced. However, based on the overall number of mispositioning problems (68 for 1996), the number of mispositioned components was considered high.

c. Conclusions

Review of the monitoring, root cause determination, and prevention of component mispositionings concluded that the licensee's program was properly focused and receiving good management attention. However, while the number of significant mispositioning events (as defined by the licensee's program) has decreased, the overall number of mispositioned components was considered high indicating further improvements could be made.

O8 Miscellaneous Operations Issues (92700)

O8.1 (CLOSED) LER 50-369/96-02: Inadvertent Manual Initiation of a Unit 1 Feedwater Isolation due to an Inappropriate Action. The event occurred when an operator was preparing to close the reactor trip breakers, the operator inadvertently depressed the train B feedwater isolation initiate push button. Unit 1 operators promptly recognized the inadvertent CF isolation condition and initiated actions to reset the function. No plant transient resulted from the event. Subsequent corrective actions were taken emphasizing the use of self-checking. The inspectors reviewed additional corrective actions taken to eliminate human factor contributors to the issue including the addition of covers to prevent inadvertent actuation of these push buttons. This LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments (61726 and 62707)

The inspectors witnessed selected surveillance tests to verify that approved procedures were available and in use, test equipment in use was calibrated, test prerequisites were met, system restoration was completed, and acceptance criteria were met. In addition, resident inspectors reviewed and/or witnessed routine maintenance activities to verify, where applicable, that approved procedures were available and in use, prerequisites were met, equipment restoration was completed, and maintenance results were adequate.

a. Inspection Scope

The inspectors observed all or portions of the following work activities:

Procedure/Work Order	Title
• IP/0/A/3090/30	Installation and Removal of Temporary Modifications
• PT/0/A/4550/36	Controlling Procedure for SFP Storage Rack Boraflex Examination
• WO 96100592	Installation of Temporary Modification

b. Observation and Findings

The inspectors found the work performed under these activities to be professional and thorough. All work observed was performed with the work package present and in active use. Technicians were experienced

and knowledgeable of their assigned tasks. The inspectors frequently observed supervisors and system engineers monitoring job progress, and quality control personnel were present whenever required by procedure. When applicable, appropriate radiation control measures were in place.

In addition, see the specific discussions of maintenance observed under M1.2, M2.1, and M2.3, below.

M1.2 Steam Generator Replacement (50001)

a. Inspection Scope

To evaluate the licensee's Steam Generator Replacement Program (SGRP) for McGuire Unit 1, by observation of selected work activities including welding, material storage and handling, nondestructive testing, machining and welder training.

b. Observations and Findings

Applicable Codes and Standards

By review of the applicable sections of the McGuire FSAR, Steam Generator (SG) Replacement Manual and various scope documents, the inspectors ascertained that the following ASME Code Sections and Editions were applicable to the SGRP.

- SG replacement, ASME Code Section XI, 1989 Edition, Article IWA-400

- Weld process, inspection and leak testing, Corporate Manual Section NSD-400 and Code Case N-416-1

- Welding procedures for secondary pipe butt welds, were qualified to the latest ASME Code Section IX in effect at the time of the qualification. The original construction code of record for McGuire Units 1 and 2 is the ASME Code Section II, 1971 Edition.

Main Feedwater Pipe Welding

The Main Feedwater (CF) piping is being rerouted to accommodate the location of the CF nozzle on the replacement Steam Generators. The replacement piping is made from SA-335 (P11) material which was produced from chrome-moly steel. This material was selected on the basis of its demonstrated good resistance to flow assisted erosion corrosion attack. The licensee is replacing all the feedwater piping from the CF nozzle back to the crane wall including the bypass lines around the check valve

adjacent to the CF nozzle. This rerouting added approximately 40 feet of piping from the crane wall, rising up to the 788 foot elevation. The scope of work on the CF system was addressed in modification package MG 19420, "CF Piping Reroute Due to S/G Nozzle Relocation," February 9, 1996.

In an effort to minimize weld fabrication and pipe assembly inside containment, the licensee is prefabricating CF pipe spools on-site at the fab-shop. The method of weld fabrication has been changed from manual to machine welding. This change was in part due to lessons learned from the Catawba SGRP and for improvement in weld quality, ease of welding and a corresponding increase in production.

The automatic welding machines used were Dimetric's Gold Track IV/DSP models. To gain the necessary proficiency in their operation, the licensee selected 32 welders. These welders were divided into four teams and sent to the vendor's facility for training. Each team trained for a period of three weeks which consisted of classroom instruction, hands-on machine familiarization and testing on pipe coupons. On January 7, 1997, the inspector accompanied the SGRP Weld Lead Engineer, on a visit to the vendor's facility located in Davidson, North Carolina. The inspector toured the training facility, observed the hands-on training activities and discussed the program with the weld lab manager. The training provided appeared adequate and should achieve its objectives.

In the fab-shop, the inspectors observed fabrication of CF pipe spools using the gas tungsten arc (TIG) welding machines discussed earlier in the previous paragraph. By review of Work Process Control Sheets (WPCS) and Field Welds Data Sheets (FWDS) the inspector ascertained the following:

CF pipe groove welds were being fabricated using a TIG welding machine procedure qualified for welding chrome-moly material. The parameters of the qualification were documented on Procedure Qualification Record (PQR) L-140D, Rev. 0, dated November 31, 1996. Production welds were fabricated using FWDS L-222A, Rev. 1, generated to provide detailed information of the PQR above, for use on production welds. As such, the inspectors verified that machine settings were consistent with the qualification parameters and the essential variables of ASME Code, Section IX.

Welding in progress at the time of this inspection involved the following weld joints.

Welding in Progress

Weld #	Size	Description	Condition
CF1FW27-15	16"x.844"	Elbow to Pipe S/G-1D	Welding Out
CF1FW27-23	16"x.844"	Elbow to Pipe S/G-1D	Welding Out

The inspectors also checked completed welds for weld reinforcement, workmanship, weld and welder identification, cleanliness and component identification for traceability. This work effort was performed on the following weldments.

Completed Welds

Weld #	Size	Description	Condition
CF1FW24-27	16"x.844"	Pipe to Elbow S/G-1A	Completed, RT Rejected
CF1FW24-27	16"x.844"	Pipe to Elbow S/G-1A	Completed, RT Rejected
CF1FW27-6	18"x.938"	S/G-1D	Completed, RT Accepted
CF1FW27-20	16"x.844"	Pipe to Elbow S/G-1D	Completed, RT Rejected

As a followup to this field inspection, the inspector reviewed selected Weld Process Control Sheets for completeness and accuracy including signoffs for cleanliness, fitup, code inspector's hold points, preheat, interpass temperature checks and final visual inspections as applicable. In addition, the inspectors selected for review, qualification records of welders who participated in the fabrication of these welds. A total of seven welders were selected for a check of performance qualifications and updates. These qualification records were found to be in order.

Material Traceability and Control

Feedwater Piping - Material used for replacement of the feedwater line was purchased to Specification DPS 1206.00-02 001, Rev. 7, July 11, 1995, and to the requirements of ASME Code Section II and III, 1989 Edition, 1989 Addenda Articles NC-2000 and NCA-3800 for SA 335, P11 Class 2 Materials. Associated elbows and reducers were purchased to

Specification DPS 1206.00-02-0003, Rev. 6 and to the requirements of the above-mentioned codes for SA 234, WP11 Class 1 material. Material identification and QA traceability numbers were as follows:

Material	Heat No.	QA No.
16" sch/80 Seamless Pipe	942558	MC41918
18" sch/80 Seamless Pipe	194931	MC44334
18" sch/80 Seamless Pipe	195097	MC44336
16" sch/80 45°Elbow	9016A	MC41617

Material	Heat No.	QA No.
16" sch/80 90° Elbow	9017A	MC41618
18" sch/80 90° Elbow	1G4B2U1H9	MC45066
16" sch/80 45° Elbow	1G4B2U1I9	MC45068

By review of certificates of conformance, the inspectors verified that chemical analysis, mechanical tests, hardness, charpy V-Notch impact tests and hydrostatic testing had been performed and that the results were within code allowable limits.

Filler Metal - Material for fabricating feedwater pipe welds was purchased to the licensee's Specification DPS-1206.00-02-0005, Rev. 003 and to the requirements of ASME Code Section II, Part C and Section III (95), NB-2400 for Class 1 material. Filler metal used for this application was as follows:

Material Type	Size	Heat/Lot No.	QA No.
ER80S-B2	0.045"	219389	897241
ER80S-B2	0.035"	219389	897242

Material Type	Size	Heat/Lot No.	QA No.
EN82	1/8" dia.	DN6209	855070
ERN1CR-3	3/32" dia.	CN6830	854652
ERN1CR-3	1/8" dia.	DM6577	854651

By review of certified material test reports, the inspectors verified that chemical, mechanical and weld tests performed, produced results that were within code allowable limits.

Nondestructive Examinations (NDE) Radiography

Main feedwater pipe welds, fabricated onsite, were radiographed as required by the applicable code. The licensee's code implementing procedure for this examination was NDE-10, Rev. 19, General Radiographic Procedure, which referenced ASME Code, Sections V and XI, 1989 Edition. The inspector reviewed radiographs of completed main feedwater welds to verify proper penetrameter type, size, placement, and sensitivity as well as film density, identification, quality and weld coverage. Welds selected for this work effort were as follows:

Steam Generator 1A

Weld No.	Size	Description	Status
CF1FW24-24	16"x.844"	Pipe to Elbow	Accepted: 12/9/96
CF1FW24-23	16"x.844"	Pipe to Elbow	Rejected: 12/12/96 Lack of fusion (LOF) and porosity
CF1FW24-19	16"x.844"	Pipe to Elbow	Rejected: 12/12/96 and 12/17/96 LOF and porosity
CF1FW24-27	16"x.844"	Pipe to Elbow	Rejected: 12/05/96 porosity Accepted 12/10/96
CF1FW24-14	18"x.938"	Pipe to Reducer	Rejected: 12/10/96 porosity Accepted: 12/11/96
CF1FW24-10	18"x.938"	Pipe to Reducer	Accepted: 12/05/96
CF1FW24-09	18"x.938"	Elbow to Pipe	Accepted: 12/11/96

Steam Generator 1D

Weld No.	Size	Description	Status
CF1FW27-05	18"x.938"	Elbow to Pipe	Rejected: 01/07/97 LOF
CF1FW27-06	18"x.938"	Pipe to Elbow	Accepted: 12/17/96
CF1FW27-10*	18"x.938"	Reducer to Elbow	Accepted: 12/19/96
CF1FW27-20	16"x.844"	Pipe to Elbow	Rejected: 01/07/97 LOF

*Records on file indicated that this weld had been postweld heat treated (PWHT). A review of associated records revealed that PWHT parameters were consistent with code and procedural requirements.

Review of Other Secondary Piping System Weld Radiographs

Completed welds in the Auxiliary Feedwater (CA) system, the Blowdown Recycle (BB) system and the Wet Lay-up Recirculation system were radiographed as required by code to determine acceptability for service. These welds were primarily dissimilar metal, nozzle to pipe welds. Welding a short transition piece of stainless steel pipe to the nozzles allowed for the fabrication of the dissimilar metal welds to be made while the replacement S/Gs were still in the onsite manufacturing facilities. Making those welds at this time eliminated the need to fabricate them inside containment where accessibility contributed to welding problems at Catawba.

Steam Generator 1C

Weld No.	Size	Description	Status
BB1F-45	3"x.438"	Pipe to Nozzle	Accepted: 12/12/96
BB1F-22-59	3"x.438"	Pipe to Nozzle	Accepted: 12/10/96
BBW1FW4-23	3"x.438"	Pipe to Nozzle	Accepted: 12/13/96
BBW1FW4-24	3"x.438"	Pipe to Nozzle	Accepted: 12/13/96
CAW1FW18-1	6"x.719"	Pipe to Nozzle	Accepted: 12/11/96

Steam Generator 1D

Weld No.	Size	Description	Status
CA1FW15-1	6"x.719"	Pipe to Nozzle	Accepted: 11/20/96
BB1F-267	3"x.438"	Pipe to Nozzle	Accepted: 01/07/97
BB1F-277	3"x.438"	Pipe to Nozzle	Accepted: 12/18/96
BB1FW5-25	3"x.438"	Pipe to Nozzle	Accepted: 11/21/96

Through this review the inspector ascertained that the radiographs showed the welds were properly evaluated. That radiographic density, and penetrameter sensitivity were sufficient to display the penetrameter image and the specified hole which are essential indications of the radiograph's image quality. In addition, the inspector noted that penetrameter location met the intent of code requirements that the films were free of artifacts, chemical streaks and equipment related problems observed on previous inspections.

This review also disclosed that use of Dimetric TIG machines used to fabricate the CF welds did not meet the licensee's expectations of increased weld production with a corresponding reduction of rejections. As the radiographs indicated, a large number of welds exhibited various amounts of porosity which in many cases exceeded code allowable and some lack of fusion. In an effort to alleviate this problem the licensee made several adjustments to the welding technique without success. Production welding was stopped, and technical experts were contacted for assistance.

Weld samples were fabricated on test coupons from the same material and filler metal. Some of the changes implemented at this time included: 1) a switch in the technique used to deposit weld metal, i.e., weave to stringer bead, 2) change in the position of the tungsten electrode with respect to the weld groove, 3) a change in the travel speed and an incremental increase in amperage. Following the close of this inspection on January 10, 1997, the licensee's cognizant engineer informed the inspectors by telephone, that one of the test samples which had been welded with Argon gas only instead of the Argon-Helium mixture was relatively free of porosity indications. Also at this time, the inspectors were informed that test results indicated the porosity indications were identified as calcium inclusions whose origin was still under investigation. The licensee indicated that additional test samples would be welded using Argon gas only. If successful, this would verify that the problem was not technique or welder related and as such, production would begin again using the single gas procedure. Repairs on completed welds were being made using the manual TIG process.

c. Conclusions

The low rejection rate of dissimilar metal welds on the secondary pipe systems, i.e., BB, BW and CA, attained during this SGRP suggests the licensee has made significant programmatic improvements in this area which would include preparation, allocation of technical resources and training. However, the relatively high rejection rate of machine fabricated CF welds suggests that preparation for machine welding on a production basis was inadequate. Stopping of weld production to investigate and correct the problem was appropriate and had good support from management. However, if deletion of Helium gas from the procedure resolves the porosity problem, it would be reasonable to conclude that this problem should have been identified and corrected before moving into the production phase of the operation.

The quality of radiographs in terms of setup preparation and film development exhibited a significant improvement over that observed on previous inspections at other Duke nuclear stations.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Boraflex Testing

a. Inspection Scope (IP 62703)

The inspectors witnessed portions of the licensee activities to assess the condition of the boraflex neutron absorber material in the Unit 2 Spent Fuel Pool Storage Racks. The testing was performed to evaluate the potential for gamma radiation-induced shrinkage of the absorber material and the long term performance effects as a result of gamma exposure and the wet pool environment. The McGuire Unit 1 and Unit 2 fuel storage racks were provided by Westinghouse.

b. Observations and Findings

The inspectors conducted observations of portions of the storage cell testing. The inspectors noted that fuel building ventilation was in operation with operable radiation monitoring equipment. Control room indications of fuel pool level and fuel pool boron concentration met TS requirements. The test equipment was calibrated onsite using a special calibration/test cell provided by the vendor.

In the licensee's response to NRC Generic Letter 96-04, Boraflex Degradation in Spent Fuel Pool Storage Racks, the licensee committed to perform in-situ testing of the fuel storage racks. The licensee consulted Northeast Technologies to perform the testing. The examination was conducted in accordance with PT/0/A/4550/36, Controlling Procedure for SFP Storage Rack Boraflex Examination, using the Boron Areal Density Gage for Evaluating Racks (BADGER) test equipment. This equipment was developed by Northeast Technologies under contract with

the Electric Power Research Institute. The test also provided additional data to validate the BADGER device for in-situ boron-10 determination in borated fuel storage pools.

The licensee selected several high dose storage cells to quantify Boron-10 areal density, gaps, thinning, and absorber end elevations. The testing was performed by Northeast Technologies under direct supervision of a McGuire site sponsor. Test equipment performance was adequate. The licensee identified some gaps and potential thinning but none was indicative of gross degradation. Qualitative analysis of the test data will be performed by Northeast Technologies and a final report is expected to be issued in approximately 8 weeks. At that time, Duke is expected to compare the information with current criticality calculations and make any necessary revisions to ensure storage rack operability.

c. Conclusion

The inspectors concluded that the preparation and coordination of the testing was good. Previous attenuation test results reviewed by the inspector coupled with the current data including spent fuel pool silica concentration provided good preliminary evidence that no gross boraflex absorber degradation has occurred. Final test results are expected in six to eight weeks.

M2.2 1RN171B Service Water to Diesel Cooling Water Heat Exchanger Supply Valve Failure

a. Inspection Scope (IP 61726)

During VOTES testing of the Service Water to Diesel Engine Cooling Water Heat Exchanger Supply Valve, 1RN171B, the licensee noted abnormally high running loads and the valve failed to stroke as expected. The safety-related valve is designed to open to allow essential cooling water to the Diesel Generator Engine Cooling Water System heat exchanger. The valve was being tested because of indications of rapid degradation based on motor current analysis and thrust data. According to licensee data, the valve failed to operate due to an abnormally high gearbox load. Based on discussions with the licensee, no preventative maintenance procedures or activities were recommended for this type gearbox.

b. Observations and Findings

The licensee responded to the valve failure by placing the valve in the open position and removing power to allow continuous flow through the heat exchanger to ensure adequate cooling would be available if necessary during an event. The licensee evaluated the effects of continued service water flow on the heat exchanger and determined that the continuous flow would not cause any immediate operational problems. Nonetheless, the licensee developed a temporary modification to reduce

the likelihood of accelerated heat exchanger fouling due to continuous cooling water flow.

The temporary modification removed the automatic functions from 1RN171 and transferred these signals to heat exchanger outlet flow control valve, 1RN174. Valve 1RN174 was then placed in the closed position. This allowed cooling water flow to be shut off when the diesel generator was not in operation. The normal inlet valve was secured open with power removed. Testing was performed to verify component performance by simulating a safety injection, emergency diesel generator and auxiliary feedwater starts in accordance with the temporary modification test package. Service Water System flow balancing was not necessary since overall system flows were not affected. The applicable standards continued to be met since both valves were butterfly valves with electric motor operators supplied with safety power, were periodically tested to meet a 60 second stroke time, and there was no physical changes to the valves. The temporary modification was scheduled to be removed during the Unit 1 EOC11 outage.

The inspectors also sampled affected procedures and verified that necessary procedure revisions had been completed.

c. Conclusions

The inspectors noted good response by station Maintenance and Engineering to implement necessary temporary modifications to return the system to operable status without compensatory measures. No Unreviewed Safety Question associated with this modification was identified.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (CLOSED) VIO 50-369,370/96-08-02: Inadequate Containment Annulus Surveillance Procedure. This violation addressed the licensee's failure to ensure the requirements of a TS surveillance test were met. Specifically, procedures used by the licensee to conduct surveillance testing on containment annulus ventilation were not adequate to ensure that heaters remained operable for at least 10 hours. The inspector verified that applicable procedures were corrected to ensure compliance with TS requirements. The inspector also noted that the licensee planned to conduct further study of the broader issue under a comprehensive plan to review the implementation of TS surveillance testing at the station. The inspector considered the immediate corrective actions taken for the violation adequate and the scope of the surveillance testing review to identify other problems appropriate. This violation is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Generic Letter (GL) 89-10 Program Implementation

a. Inspection Scope (TI 2515/109)

This inspection provided an assessment of the licensee's implementation of GL 89-10, "Safety-Related-Motor-Operational Valve Testing and Surveillance". The licensee had notified the NRC that implementation was complete in a letter dated December 28, 1995.

The inspection included evaluations of: the scope of MOVs included, the calculations of the design basis differential pressure, the determinations of MOV settings and verifications of MOV capabilities, the periodic verification of MOV capabilities, the MOV post maintenance and post modification testing requirements, assuring against pressure locking and thermal binding, and trending of MOV performance.

The inspectors conducted their assessment through a review of the licensee's GL 89-10 implementing documentation and through interviews with licensee personnel. The documents reviewed included: "NRC Generic Letter 89-10 Program Plan," Rev. 4; "Guideline for Performing Motor Operated Valve Reviews and Calculations", DPS-1205.19-00-0002, Rev. 0; "Engineering Support Program, Generic Letter 89-10, Motor Operated Valves," Rev. 0; "Evaluation of Rate-of-Loading Effects", DPC-1205.19-00-0002, Rev. 0; DPC-1205.19-00-0001, Rev. 1, "Evaluation of Stem Factor and Stem C.O.F. Assumptions;" a summary matrix of margins available for all MOVs in the GL 89-10 program; and the additional procedures, calculations, test records, etc., referred to in the following paragraphs.

b. Observations and Findings

1. Scope of MOVs Included in the Program

The scope of valves originally in the licensee's GL 89-10 program was reviewed previously by the NRC during Inspection 92-11. This scope would be considered acceptable based on current NRC positions. During the current inspection the inspectors evaluated the subsequent deletions of valves from the program to determine if they had been adequately justified. The deletions were identified by comparing the valves in the original program with those in the current program. The inspectors then assessed the acceptability of the deletions through a review of justifications given in MCC-1205.19.00-0011 and design and functional information described in the FSAR and other licensee documents (including piping and instrumentation diagrams). The inspectors found that the licensee had a satisfactory basis for each MOV deletion.

A total of 121 valves had been removed from the scope of the program that was previously reviewed by the NRC during Inspection 92-11. The inspectors found the basis for the removal of these valves was satisfactory. They had been removed because they were identified not to have a safety function, the power had been removed from the valves, or because they were excluded by Supplement 7 to GL 89-10, "Valve Mispositioning in Pressurized Water Reactors". Some valves were also removed from the program because they were determined not to be gate, globe, or butterfly valves. The MOV program remained very large with a total of 425 MOVs (consisting of 189 gate, 136 globe, and 100 butterfly valves).

2. <u>Revised Design-Basis Differential Pressure Calculations</u>

In inspection 95-06, NRC inspectors noted that the licensee was revising the calculations of MOV maximum differential pressure and stated that a further NRC review of the calculations would be performed. That further review was performed during the current inspection. The inspectors reviewed the differential pressure calculations (such as MCC-1223.42-00-0026 for the auxiliary feedwater system), the calculation reference documentation, and the applicable system flow drawings. The inspectors found that satisfactory calculations had been completed for all systems.

Inspection 95-06 particularly noted that the maximum differential pressure calculation for valves NI-184B and NI-185A did not include pressure locking effects. The inspectors found that these valves had been modified with a bonnet equalization line (drawing MCFD-1562-03.01) to preclude pressure locking. The inspectors found that the calculation (MCC-1223.12-00-0017, Rev. 4) had been appropriately revised after inspection 95-06 and that it accounted for the system configuration applicable to use of these valves.

3. <u>Determinations of Settings and Verifications of Capabilities for Gate Valves</u>

The inspectors selected and reviewed calculations, test data, and evaluations for the following sample of gate valves, in order to assess the licensee's validation of calculation assumptions and their determinations of MOV settings and capabilities:

1CA0042	Auxiliary Feedwater System Pump 1B Isolation Valve
1KC0018	Reactor Building Isolation of Non-Essential Header
1NC0031/3/5	PORV Block Valves
1ND0019	ND Pump Suction Isolation Valve
1NI0009	NC Cold Leg Injection from NV
1NI0100	Suction Valve from RWST to NI Pumps
1NV0244/5	Containment Isolation for the NV Charging Header
2KC0018	Reactor Building Isolation of Non-Essential Header

2NC0031	PORV Block Valve
2NI0100	Suction Valve from RWST to NI Pumps
2NI0333	Safety Injection Pump Suction Crossover from NV
2NV0095	Containment Isolation for the Reactor Coolant Pump Seal Return

The inspectors findings were as follows:

MOV Sizing and Switch Settings

McGuire gate valve thrust calculations typically utilized the standard industry equations. Mean seat diameter was used to calculate valve seat area. Valve factors were based on in-plant test results or results from other industry sources. The licensee used statistical methods to evaluate the effect of MOV performance uncertainties on available margin.

Valve Factors and Grouping

McGuire dynamically tested 53 of their 189 gate valves. Measured valve factors were used for the gate valves that were dynamically tested. The licensee gathered in-plant and other industry valve factor data, and formed 17 gate valve groups to justify the applied valve factors for non-dynamically tested gate valves. The inspectors found that the valve factor justifications for each valve group and the current setup of the MOVs was adequate for design-basis capability. However, they noted that the bases for the valve factors established for a few groups was weak, due to the limited quantity or quality of the supporting data. These weaknesses were of greater concern when the valve groups contained valves with marginal calculated design basis capabilities. The licensee provided actions to address the cases which the inspectors considered more significant through Problem Investigation Process (PIP) 0-M96-3542. Discussions of the weaknesses noted and the licensee's actions are as follows:

Group B consisted of 24 Aloyco split-wedge gate valves of different sizes. The inspectors noted that the four 12-inch valves in this group were marginal and that none had been dynamically tested. Based on the lack of dynamic test data and the marginal capabilities for these valves, the inspectors considered that the justification for the assumed valve factor was weak. The licensee established an action item in PIP 0-M96-3542 to initiate work requests to improve the margin for these valves and to obtain additional test data to support the group valve factor.

Group D consisted of eleven 4-inch Aloyco split-wedge gate valves. The licensee initially based the group valve factor on a single dynamic test. The inspectors observed that this was inconsistent

with GL 89-10, Supplement 6, which indicates that group valve factors could be based on tests of nominally 30%, but no less than 2 valves from the group. The licensee obtained additional test data from similar valves tested at another plant which satisfactorily supported the valve factor used for the group.

Group F consisted of eight 6-inch Borg Warner gate valves. The valve factor for this group was based on test results from two similar valves at another of the licensee's plants. The inspectors considered this weak, as GL 89-10, Supplement 6, recommended testing at least three valves from groups of this size. Further, the inspectors found that the capability of MOV 1CF0129 in this group was marginal. The licensee established an action item in PIP 0-M96-3542 to upgrade this MOV at the next refueling outage.

Group H consisted of eight 4-inch Borg Warner gate valves whose safety function is to close to isolate a faulted steam generator. The inspectors found that the valve factors used in determining the closing torque switch settings for these valves were nonconservative, considering the licensee's limited in-plant test data. However, the torque switches were bypassed for 95% of the closing valve stroke and the inspectors found that the valves were capable of completing at least that much of the closing stroke. They considered this marginally sufficient isolation capability. The licensee's Improvement Plans indicated these MOVs were to be replaced.

Group K consisted of six 3-inch Borg Warner gate valves which functioned as the PORV block valves. The group valve factor was based on test results obtained from similar valves at the licensee's Catawba plant. This data was not directly applicable to the McGuire valves as it was obtained under pumped flow conditions whereas the McGuire block valves experience blowdown flow. Based on their available output thrust at the current torque switch settings and on industry blowdown test data from a similar valve, the inspectors did not have an immediate operability concern regarding these valves. However, they considered the capabilities of these MOVs to be marginal. The licensee established action items in PIP 0-M96-3542 to upgrade these MOVs at the next refueling outage, and to either obtain additional data from other sources or apply the EPRI MOV Performance Prediction Methodology to strengthen support for the group valve factor.

Another weakness noted by the inspectors was that the licensee, sometimes included multiple test data points from a given valve in statistically analyzing the test data from a group of valves. This could bias an analysis. The licensee established an action item in PIP 0-M96-3542 to revise Duke Power Specification DPS-

1205.19-00-0002 to require the number of data points per valve to be equal to avoid biasing the group valve factor in favor of valves with more than one data point.

Load Sensitive Behavior

Gate and globe MOVs that were dynamically tested used the measured load sensitive behavior values. DPC-1205.19-00-0002, "Evaluation of Rate-of-Loading Effects," specified a bias margin of 6.96% and a standard deviation of 13.8% to account for the effects of load sensitive behavior for GL 89-10 MOVs that were not dynamically tested. However, the licensee had noted a continuing improvement in load sensitive behavior performance across all of the Duke nuclear facilities. Therefore, Duke was in the process of revising the load sensitive behavior justification. Duke personnel determined that, based on data from all of the Duke facilities over the last 3 years, a mean of 4% with a standard deviation of 10.56% represented the reduction of thrust at torque switch trip under dynamic conditions as a result of load sensitive behavior. In response to inspectors' questions, the licensee presented information to demonstrate that the load sensitive behavior of the MOVs at the McGuire facility was consistent with the overall Duke assumption of the effects of load sensitive behavior. The licensee established an action item in PIP 0-M96-3542 to incorporate the new plant-specific information in Duke corporate document DPC-1205.19-00-0002.

Stem Friction Coefficient

McGuire's original calculations assumed a stem friction coefficient value of 0.15 in determining actuator output thrust capability. This value was selected based on a review of in-plant test data from all Duke facilities. Based on an updated review, the licensee determined that the original assumptions for McGuire were somewhat nonconservative, and the thrust calculations were revised to incorporate the use of a 0.20 stem friction coefficient, except where specific test data supported a lower value. The inspectors found this to be acceptable. However, the inspectors noted that certain MOV program documents still cited the use of a 0.15 stem friction coefficient. The licensee established action items in PIP 0-M96-3542 to revise McGuire calculation MCC 1205.19-00-0003 and Duke corporate document DPS-1205.19-00-0002 to incorporate the new information for gate and globe valves. The licensee also established an action item in PIP 0-M96-3542 to discuss with plant personnel that the results of its analysis of stem friction coefficient may not be effective and to re-enforce the importance of performing high quality stem lubrication. The licensee stated future plans to obtain more reliable torque data than was currently available from the spring pack curves used for some globe valves.

Diagnostic Equipment Uncertainties

NRC inspection 95-06 determined that McGuire personnel were not accounting for VOTES diagnostic equipment uncertainties in the open direction when measurements were outside the sensor calibration range. These errors can become very large if measurements are significantly outside the calibration range. At the time of that inspection, McGuire personnel initiated PIP 0-G95-0295 to resolve the issue. The methods identified in this PIP included a review of all diagnostic testing that existed at that time. Subsequent to the inspection, McGuire developed DPC-1205.19-00-0003, "Evaluation of MOV Open Direction Issues," Rev. 0, which: 1) established the basis for the methods used to evaluate the issue, and 2) provided the screening of McGuire's MOVs. The licensee also revised their diagnostic procedures to obtain tension in the calibration range (when needed) which reduces the applicable diagnostic error in the open direction. After review of the actions taken to address this concern, the inspectors considered this issue closed for McGuire.

Design-Basis Capability

At the outset of the inspection, the licensee presented a method for calculating the thrust required to operate non-dynamically tested MOVs that relied, in part, on perceived risk contribution (as established using PRA techniques). In this method, the risk assigned to a valve was used to select a confidence level which was used to determine the number of standard deviations to be applied to each uncertainty in MOV performance (i.e., variation in valve factors, load sensitive behavior, and diagnostic equipment uncertainty). In effect, this resulted in operability being determined on the basis of PRA risk. The inspectors noted that this was considered weak and was contrary to GL 91-18, which indicates that PRA or risk should not be used in operability decisions.

During the inspection, the licensee reconsidered their methodology and re-evaluated the current setup of all its GL 89-10 MOVs, applying a 95% confidence level to uncertainties. Additionally, they established action items in PIP 0-M96-3542 to 1) revise specification DPS-1205.19-00-0002 to require minimum thrust calculations to be based on a 95% confidence level for uncertainties, 2) revise their calculations to provide minimum thrust requirements based on the 95% confidence level, 3) include guidance on consideration of potential MOV aging and degradation effects, and 4) ensure that the work documents for the upcoming Unit 1 outage reflect the appropriate valve setup requirements.

The licensee did not apply the risk-based methodology to the butterfly valves or Kerotest globe valves. With respect to the

non-Kerotest globe valves, the licensee established an action item in PIP 0-M96-3542 to revise the thrust calculations to be more consistent with accepted deterministic methods.

To allow a comparison to deterministic methods, the licensee calculated available valve factors that were based on the thrust available in the closing and opening directions. These calculations were adjusted to account for diagnostic equipment uncertainty, torque switch repeatability, and bounding load sensitive behavior assumptions. MOVs with less than a 0.50 available valve factor were then reviewed individually by the inspectors. The inspectors did not identify any immediate operability concerns. However, the inspectors considered the following MOVs to be marginal:

1NC0031	1NC0033	1NC0035	1ND0019	1NI0100
1NV0244				
1NV0245	2NC0031	2NC0033	2NC0035	2NV0095

The licensee agreed and noted that several of these MOVs were already scheduled for margin improvements in the near future. The licensee established an action item in PIP 0-M96-3542 to ensure that these MOVs are upgraded at the next refueling outage.

4. Determinations of Settings and Verifications of Capabilities for Globe Valves

The GL 89-10 program at McGuire included 134 small globe valves manufactured by Kerotest and two globe valves manufactured by Walworth-Aloyco. The licensee was not able to obtain reliable diagnostic data from dynamic testing of the Kerotest globe valves at McGuire. Instead, they conducted a testing program at their flow loop research facility to evaluate the performance of the Kerotest globe valves under pumped flow and blowdown conditions.

The inspectors reviewed the results and evaluation of the Kerotest globe valve test program, as documented in DPC-1205.01-00-0001, Rev. 1, "Evaluation of Flow Loop Tests of Kerotest Valves," and DPC-1205.01-00-0002, "Evaluation of Kerotest Valves in Steam Blowdown Conditions." The inspectors noted that the vendor's method for predicting thrust requirements for the 2-inch soft-seat design Kerotest globe valves had been found nonconservative, based on full flow differential pressure test results. The licensee had recognized this nonconservatism and had included additional margin for these particular valves in their calculations, depending on service application. The licensee established an action item in PIP 0-M96-3542 to include specific guidance in DPC-1205.01-00-0001 for the additional margin and for consideration of service application when applying the Kerotest thrust prediction method to

the 2-inch soft-seat valves in 150 psi or less water service applications.

The licensee tested one of the two globe valves manufactured by Walworth-Aloyco under dynamic conditions. The results supported the 1.1 valve factor assumed for the other globe valve manufactured by Walworth-Aloyco.

The inspectors considered the licensee's thrust requirements for the GL 89-10 globe valves to be acceptable.

5. <u>Determinations of Settings and Verifications of Capabilities for Butterfly Valves</u>

The McGuire GL 89-10 program included 100 butterfly valves that were separated into 14 groups. The licensee conducted dynamic tests on 52 butterfly valves and applied the test results, where applicable, to the valves that were not dynamically tested.

The inspectors reviewed the licensee "validation calculations" which evaluated test data to demonstrate the capabilities of butterfly valves to perform their design basis functions. The inspectors found that the test data and evaluations documented in these calculations demonstrated generally satisfactory settings and capabilities for the licensee's butterfly valves. However, the inspectors noted weaknesses for three groups (E, I, and K) of nuclear service water system valves:

- Group E consisted of twelve (six per unit) 10-inch, class 150, model NMK 11, Henry Pratt butterfly valves. These valves were addressed by validation Calculation MCC-1205.19-00-0030, Rev. 0. The test data used to establish the settings and capabilities of these valves was considered weak by the inspectors, as it was from static and dynamic tests performed on much larger (16-inch) valves. Additionally, relying on this data, the calculated capabilities of several of these valves only marginally exceeded the design basis requirements (by 1% or less).

- Group I consisted of four (two per unit) 6-inch, class 150, model 7620, Fisher Controls butterfly valves. The test data used to establish the settings and capabilities of these valves was considered weak, as it was not quantifiable by the licensee's normal evaluation methods and the dynamic testing had been performed at only about 60% of design basis differential pressure. The calculation employed what was referred to as a "non-typical validation" approach in assuring the capabilities of these valves.

- Group K consisted of four (two per unit) 8-inch, class 150, model NMK11, Henry Pratt butterfly valves. The inspectors found the data which the licensee had used to establish the settings and capabilities of these valves was weak, as it was based on tests performed on tests of much larger valves (16- and 20-inch). The licensee had statically and dynamically tested the group K valves but had determined that the test results could not be relied upon for quantitative evaluations. Further, the licensee did not qualitatively demonstrate the capabilities of the valves through the group K valve dynamic tests, as they were performed at only about 60% of design basis differential pressure.

  Note: Group K Valve 1RN171B, a nuclear service water supply isolation valve to the diesel generator heat exchanger, was tested during this inspection and failed to perform satisfactorily. Refer to paragraph M.2.2 for details on this failure.

To address the above weaknesses, the licensee established an action item in PIP 0-M96-3542 to perform dynamic testing with diagnostics on four Group E valves and all valves in Groups I and K. Additionally, the action item required raising the torque switch settings for the marginal Group E valves to provide increased assurance they would perform their design basis functions.

6. Periodic Verification

The licensee incorporated MOV periodic verification requirements into the MOV preventive maintenance (PM) Program. The inspectors reviewed this PM Program, dated March 26, 1996. The program was computerized and specified stem lubrication, diagnostic test, and actuator inspection intervals. The inspectors concluded that the PM Program for GL 89-10 MOVs was well defined. During the inspection the licensee was in the process of establishing criteria for determining when periodic MOV dynamic testing would be performed.

The licensee's periodic verification actions were considered adequate for closure of the GL 89-10. The NRC may re-assess the licensee's long-term periodic verification program as part of its review of GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves", dated September 18, 1996.

7. Post Maintenance Testing (PMT)

The inspectors reviewed the licensee's PMT requirements and guidance, which were specified in their PMT Program, dated May 29, 1996, and their Work Process Manual, Section 501, dated August 21, 1995. Additionally, the inspectors reviewed the PMT recorded for WOs 94076860 (packing leak repair), 95005056 (adjust packing), 94076858 (adjust packing), and 95015982 (adjust packing). The inspectors found the licensee's PMT acceptable for GL 89-10 closure; however, weaknesses were noted and were addressed by the licensee as described below.

The PMT program only required stroke testing following packing adjustment if the packing torque was not increased above that present in the last base line diagnostic test. From their review of WO PMTs the inspectors found that the MOV engineer assigned a more rigorous PMT than specified by the PMT Program. A diagnostic test or MPM (Motor Power Monitor) test was required following packing adjustment in addition to stroke testing. The inspectors concluded that the PMT requirements specified in the PMT Program for packing adjustments were lacking in detail. During the inspection the licensee provided an action item in PIP 0-M96-3542 to resolve this. The item specified updating of the PMT Program to provide details of the PMT to be performed and review WPM 501 for consistency.

The licensee sometimes performed an MPM test in lieu of a diagnostic force measurement as PMT following packing adjustment or replacement. The inspectors observed that this should be supported by data which demonstrates that the MPM testing is able to discern unsatisfactory increases in packing forces. During the inspection the licensee documented in PIP 0-M96-3542 the need to provide justification to support packing adjustment and replacement without performing diagnostic testing as a PMT, as when MPM is used in place of diagnostic testing.

8. Post Modification Testing

The licensee assigned post modification test requirements for MOV modifications on a case by case basis. The inspectors reviewed modifications MGMM-7097 (Change 2KC0003 Actuator Speed), dated April 10, 1996; MGMM-7155 (Replace 1NI0332 Actuator), dated January 3, 1996; and MGMM-7305 (Replace 1NS0020 Actuator), dated October 2, 1995. The inspectors found that the licensee had implemented acceptable post modification testing.

9. Pressure Locking and Thermal Binding

The inspectors reviewed the evaluation of gate valves susceptible to pressure locking and/or thermal binding which the licensee had

completed in response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves". In its letters dated February 13 and July 31, 1996, the licensee identified valves that were susceptible to pressure locking and/or thermal binding and corrective actions.

The licensee's GL 95-07 submittals stated that analytical methods were utilized to demonstrate that the actuators on valves 1(2)FW0027A, 1(2)ND0058A, 1(2)NI0136B, 1(2)NS0001B, 1(2)NS0018A, 1(2)NS0038B and 1(2)NS0043A could develop adequate thrust to overcome pressure locking. The inspectors were informed that analytical methods were used as short term corrective action and that modifications were scheduled to be implemented during the upcoming Unit 1 and 2 refueling outages to eliminate the potential for pressure locking. The inspectors reviewed Calculation MCC-1205.19-00-0052, "GL 95-07 Pressure Locking & Thermal Binding Evaluation," Rev. 14, and verified that analytical methods were adequate to demonstrate operability for present plant conditions. Additionally, the inspectors verified the planned modifications were prescribed in PIP 0-M96-0460. The licensee planned to update its GL 95-07 response to indicate that these valves would be modified to eliminate the potential for pressure locking. The inspectors considered that short and long term corrective actions to preclude pressure locking in these valves were acceptable.

The licensee's GL 95-07 submittals stated that testing would be utilized to provide reasonable assurance that valves 1(2)NS0012B, 1(2)NS0015B, 1(2)NS0029A and 1(2)NS0032A would not pressure lock. The inspectors were informed that this testing was complete but was only used as short term corrective action and that procedures were being revised to eliminate the potential for pressure locking. The inspectors verified the plans to revise the procedures to preclude pressure locking were prescribed in PIP 0-M96-0460. The licensee planned to update its GL 95-07 response to indicate that procedures would be modified to cycle these valves following operation of the applicable containment spray pump to eliminate pressure in the valves' bonnets. The inspectors considered that short and long term corrective actions to preclude pressure locking in these valves were acceptable.

The licensee's GL 95-07 submittals stated that an analytical method was used to demonstrate that the actuators on valves 1(2)LD0108A and 1(2)LD0113B could overcome pressure locking. The inspectors reviewed the associated Calculation MCC-1205.19-00-0052 and noted that the margin between the thrust required to overcome pressure locking and actuator capability for this type of valve was minimal. During the inspection the licensee reevaluated the use of this analytical method and concluded that it would be prudent to modify these valves to eliminate the potential for pressure locking. The inspectors verified the plans to modify

these valves to preclude pressure locking were prescribed in PIP 0-M96-0460. The licensee planned to update its GL 95-07 response to indicate that these valves would be modified. The inspectors considered that short and long term corrective actions to preclude pressure locking in these valves were acceptable.

The licensee's GL 95-07 submittals stated that valves 1(2)NI0009A and 1(2)NI0010B open with high head injection pump discharge pressure acting on one side of the valve, which prevents the valves from pressure locking. The inspectors reviewed emergency safeguards feature test results and noted that it took approximately two seconds for high head injection pumps to develop full discharge pressure after receiving a start signal. Valves NI0009A and NI0010B are normally shut and receive an open signal at the same time the high head injection pumps receive a start signal. Therefore, valves NI0009A and NI0010B could operate at locked rotor conditions for several seconds. During the inspection the licensee reevaluated this analysis and concluded that it would be prudent to modify these valves to eliminate the potential for pressure locking. The inspectors verified the plans to modify these valves to preclude pressure locking were prescribed in PIP 0-M96-0460. The licensee planned to update its GL 95-07 response to indicate that these valves would be modified. The inspectors concluded that operation at locked rotor conditions for a very short period of time may not adversely effect valve operability but it is not a preferred long term corrective action. As short term corrective action, the PIP 0-M96-0460 indicated the licensee would periodically review emergency safeguards feature test results to ensure that high head injection pumps develop full discharge pressure within several seconds after receiving a start signal. The inspectors considered that short and long term corrective actions to preclude pressure locking in these valves were acceptable.

The licensee's GL 95-07 submittals stated that an analytical method was used to demonstrate that actuators for PORV block valves 1(2)NC0031B, 1(2)NC0033A and 1(2)NC0035B could overcome pressure locking. The inspectors reviewed calculation MCC-1205.19-00-0052 and noted that the margin between the thrust required to overcome pressure locking and actuator capability for this type of valve was minimal. During the inspection the licensee decided to reevaluate the use of this analytical method as long term corrective action to preclude pressure locking. The licensee planned to update its GL 95-07 response to report the results of this evaluation. The inspectors concluded that use of the analytical method was acceptable for short term corrective action.

The PORV block valves were also determined to be susceptible to thermal binding. In order to resolve a GL 89-10 concern, the

licensee was planning to increase torque switch settings on these valves which would increase the valves' susceptibility to thermal binding. During the inspection the licensee decided to reevaluate corrective actions to prevent thermal binding for these valves. The inspectors verified that this reevaluation was prescribed in PIP 0-M96-0460. The licensee planned to update its GL 95-07 response to report the results of the evaluation.

With the exception of the PORV block valves, valves identified by the licensee as susceptible to pressure locking have been modified, are planned to be modified or procedures are to be revised to eliminate the potential for pressure locking. The inspectors independently reviewed pressure locking and actuator capability calculations for selected MOVs and verified that the licensee correctly calculated the thrust required to overcome pressure locking and actuator output capability. The NRC staff is continuing its evaluation of these valves and others within the scope of GL 95-07 as part of its review of the licensee's response to potential pressure locking and thermal binding.

10. Trending

The inspectors determined that a satisfactory description of failure and performance trending requirements was described in the licensee's Engineering Support Program, Rev. 0, section 4.6. They verified the licensee's implementation of adequate trending by reviewing portions of the trend database. Further, the inspectors verified that the licensee performed and documented acceptable periodic reviews of MOV failures. The reports examined by the inspectors were McGuire Failure Analysis and Trending Program Review 1/1/95 through 6/30/96, Yearly Review of Rotork Motor Operator Failure (dated March 11, 1996), and Limitorque Annual Maintenance Review (dated May 13, 1996). The inspectors considered the trending implemented by the licensee satisfactory.

11. Strengths

The inspector observed a number of strengths in the licensee's implementation of GL 89-10. Particular examples included:

- The detailed packages of data and evaluations that were developed for each valve group.
- The strong management and personnel support that was necessarily provided to complete a program encompassing the number of MOVs present at McGuire.
- The application of state of the art technology such as torque test stands and motor power monitoring.

- Leadership in addressing industry problems such as increases in actuator ratings.

c. Conclusions

The licensee had met the intent of GL 89-10 in verifying the design basis capabilities of their MOVs. Several weaknesses were identified in data and justifications of assumptions used in the verifications. The licensee planned actions to resolve these weaknesses which the inspectors determined to be more significant and they were appropriately prescribed in PIP 0-M96-3542. The licensee was requested to notify the NRC of the status of the PIP actions 60 days after the end of the upcoming Unit 1 refueling outage (1EOC11) and at the completion of all of the items by December 31, 1997. The inspectors identified the completion of these actions as Inspector Followup Item 50-369, 370/96-11-01, Actions to Address MOV Weaknesses.

In addition to weaknesses, the inspectors noted several licensee strengths, which are described in b.11 above.

One related NCV was identified, involving periodic MOV lubrication, and is described in paragraph E8.3.

Based on the NRC staff's review of the McGuire GL 89-10 program and its implementation, and the actions established by the licensee in PIP 0-M96-3542, the NRC staff is closing its review of the GL 89-10 program at McGuire. The completion of these licensee actions will be assessed as part of the NRC staff's monitoring of the licensee's long-term MOV program.

E2 Engineering Support of Facilities and Equipment

E2.1 Vital Area Access

a. Inspection Scope

During routine tours of the protected area, the inspectors noted the addition of a locked security fence at the Unit 1 exterior doghouse and a security barrier near the Unit 1 Refueling Water Storage Tank. The inspectors recognized the barriers potential impact on operator emergency response and initiated activities to evaluate operator awareness of this change to a station structure.

b. Observations and Findings

To verify that necessary training and/or procedure revisions had been completed, the inspectors conducted interviews and reviews of the station emergency procedures. The inspectors noted that no formal guidance or procedure revisions had been provided to Operations personnel describing the physical change. As a result of these initial

findings, the inspectors requested a copy of the modification package to implement the change and was informed that the installation of the barrier had not been completed as a modification.

The inspectors reviewed the Duke Power Nuclear Station Directive 301 and the McGuire Modification Manual and determined that the installation of the temporary barrier met the temporary modification criteria. Therefore the modification should have been processed in accordance with the licensee approved temporary modification process as outlined in Nuclear Station Directive (NSD) 301 which defined a physical change or addition to a station's structures system or components for a finite period as a temporary modification.

Since the change to the facility was not handled in accordance with the Temporary Modification process, no formal training or procedure revisions were provided to Operations personnel describing the change. As a result, some operator response times under certain accident conditions were slightly affected. The slight increase in operator response times did not exceed UFSAR assumptions. The licensee recognized the oversight and developed immediate Operations training packages for shift operations personnel and a followup reading package discussing the barrier and necessary actions to take in emergency situations. Unannounced drills were conducted and operator response times were validated.

c. Conclusions

Following additional reviews of the McGuire Modification Manual, the inspectors concluded that the installation of the barrier located at the Unit 1 exterior valve vault was a temporary modification and should have been controlled and implemented in accordance with NSD 301. The failure to meet the requirements of the modification manual is considered a Violation and will be documented as VIO 50-369/96-11-02: Failure to follow procedure for temporary modification installation.

E4 Engineering Staff Knowledge and Performance

E4.1 Engineering Staff Knowledge and Performance

a. Scope

The inspector reviewed the material upgrade processes to determine if commercial grade purchased items were appropriately evaluated and tested for use in safety related applications. Approximately 30 commercial grade dedication (CGD) packages were reviewed. These included mechanical and electrical items. Applicable regulatory requirements included 10 CFR 50 Appendix B, FSAR, and the following:

ANSI N45.2.13-1976, QA Requirements for Control of Items and Services for Nuclear Power Plants

RG 1.123, QA Requirements for Control of Procurement of Items and Services for Nuclear Power Plant

GL 91-05, Licensee Commercial Grade Procurement and Dedications Programs

b. Observations and Findings

Commercial grade dedication evaluations were performed by the corporate procurement engineering organization for all Duke Power Nuclear Stations. Station procurement engineering resolved receipt inspection inconsistencies. The sample of CGD purchased items reviewed were appropriately evaluated and tested for use in safety related systems. Critical characteristics for the items were adequately identified and verified. Commercial grade vendor audits and surveys appropriately evaluated the critical characteristics referenced in the CGD evaluations. Adequate documentation was maintained to validate the items' CGD. Traceability of parts and warehouse storage were adequate. Critical characteristics were adequately incorporated into receipt inspection requirements. The use of standardized commercial grade procurement acceptance procedures was a good practice and assured consistent quality verification actions at receipt inspection. Deficiencies or inconsistencies in technical or quality requirements identified in receipt inspection were adequately resolved by station Procurement Engineering.

c. Conclusion

The commercial grade dedication process was implemented effectively at McGuire and was consistent with applicable regulatory requirements. No examples were identified in which unqualified material or components were installed in safety related applications.

E4.2 Resolution of Procurement Problems

a. Scope

The inspector reviewed station Procurement Engineering's resolution of procurement related problems identified in the licensee's Problem Investigation Program reports (PIPs). These issues were not limited to CGD purchases. Applicable regulatory requirements included 10 CFR 50 Appendix B and 10 CFR 50 Appendix A.

b. Observations and Findings

The inspector noted one issue of concern related to the purchase of Grinell hydraulic pipe supports (snubbers) for the Steam Generator Replacement Project. PIP 0-M96-2408, dated August 26, 1996, identified that recently purchased Grinell snubbers were not qualified to the

original purchase specification's environmental parameters. The parameters included a 350 °F temperature and 2 X 10E8 rad value limit which was not met by the manufacturer. These specifications were also not met for the Grinell snubbers originally installed in the plant, approximately 150 snubbers. These snubbers were installed in safety related systems including the Reactor Coolant (RCS), Chemical & Volume Control, Safety Injection, and Component Cooling.

The purchase specification temperature parameter was of concern because it was based on worse case environmental conditions in containment. This condition occurred only during a Main Steam Line Break (MSLB). The containment temperature was anticipated to reach approximately 326 °F after a MSLB. The original specification of 350 °F assured that the snubbers would remain operable in this environment. The vendor informed the licensee that the snubbers' polycarbonate hydraulic reservoir would experience significant degradation above 250 °F. This could result in loss of seal and subsequent loss of hydraulic fluid which would make the snubber unable to perform its design function.

A licensee test conducted in December 1980, evaluated the durability of the snubbers subjected to high temperatures. The test heated the snubber in an oven in stages up to 400 °F. The snubber was disconnected and not subject to any internal hydraulic pressure forces that may exist on an installed snubber; therefore, the internal contribution to reservoir deformation was not included. Reservoir warpage and loss of hydraulic fluid began at 285°F. The test conclusion was that, "the snubbers should be considered unsafe at any temperature near 285 °F since the reservoirs started to visibly disfigure (not leak) at that point after brief exposure to elevated temperatures." There was no additional documentation which addressed the acceptability of the snubber design or the motivation for performing the test.

The operability evaluation in PIP 0-M96-2408 concluded that the design of the snubbers was acceptable and that there was no operability concern with the snubbers or associated piping. The evaluation stated that following the MSLB, the design load case is over and the snubbers are no longer required to be operable. That is, the snubbers are no longer components important to safety. The licensee indicated that no dynamic piping loads were anticipated following the MSLB. Although actuation of a RCS [illegible] Operated Relief Valve would induce a dynamic load, this actuation was not expected after a MSLB.

The Grinell snubber design appeared to conflict with 10 CFR 50 Appendix A, General Design Criteria four (GDC 4) which required that structures, systems and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, testing and postulated accident conditions. Based on the vender information and the licensee's test, the snubbers can not be assured of performing their design restraint function after exposure to temperatures above 285 °F, which was

anticipated in the MSLB event. It was unclear if the design or licensing basis required the snubbers to be operable after a design pipe rupture event (MSLB) or if all potential dynamic loads following a MSLB shutdown had been evaluated. This issue is identified as Unresolved Item URI 50-369,370/96-11-03, Environmental Qualification of Safety Related Piping Grinell Snubbers in Containment. The item remains open pending further NRC review to determine if the snubbers meet or are required to meet the GDC 4 requirements.

c. Conclusion

Review of procurement related items in the licensee's problem identification program did not identify problems with the commercial grade procurement process. An issue was identified related to the purchase of Grinell hydraulic pipe supports (snubbers) which did not meet the purchase specifications for environmental qualification. This item was identified as an unresolved item pending further NRC review.

E7 Quality Assurance in Engineering Activities

E7.1 Quality Assurance in Engineering Activities - Procurement Engineering

a. Scope

The inspector reviewed the licensee's self-assessment activities associated with procurement engineering processes. Applicable regulatory guidance was provided by 10 CFR 50 Appendix B.

b. Observations and Findings

A self assessment of McGuire CGD activities was performed in May 1996. The scope was adequate to assess station performance in this area. Findings were appropriately resolved. Additional self assessment activity included a benchmarking survey of procurement engineering activities at five other nuclear facilities to identify opportunities to enhance Duke procurement engineering practices. An assessment of in-service CGD part failures was completed in July 1996. This assessment was to provide information for modification of receipt inspection requirements or change of vendor suppliers to identify and address equipment with repeated in-service failures.

c. Conclusion

The licensee performed appropriate self assessment to monitor performance of the commercial grade procurement process.

E8 Miscellaneous Engineering Issues (92902 and TI 2515/109)

E8.1 (CLOSED) VIO 50-369, 370/95-06-01, Analysis and Calculation Errors.

This violation identified errors in MOV thrust calculations and analyses of static diagnostic test data. The inspectors verified the licensee's overall records of identification and correction of these errors in Problem Investigation Processes (PIPs) 1-M95-0622, -0636, and -0619. These PIPs referenced other documents for specific corrective actions and the inspectors verified a sample of these, including: correction of the thrust calculation for MOV 1CA0050 in Calculation MCC-1205.19-00-0003, Rev. 4; and evaluation of structural limitations for Valve 1CA0038 in Calculation MCC-1205.19-00-0023, Rev.1. The inspectors also verified that the licensee had investigated the extent of condition and a provided appropriate corrections for other MOVs. As an example, the inspectors verified correction for MOV 2CA0050 in Calculation MCC-1205.19-00-0003, Rev. 4.

E8.2 (CLOSED) VIO 50-369, 370/95-06-02, Errors in Entries of Test Data.

This violation involved entries of incorrect test data. The inspectors verified the licensee's overall records of identification and correction of these errors in PIPs 0-M95-0627 and -0641. These PIPs referenced other documents for specific corrective actions and the inspectors verified a sample of these, including: correction of the bench test torques recorded for MOV 2RN137 (identified WR# 94021156002) to increase the values by 5 ft-lbs; and correction of procedure IP/0/A/3066/02H (changes 0 to 8) to assure the correct structural limit is recorded on the test data sheet. No additional problems were identified. This item is closed.

E8.3 (CLOSED) URI 50-369, 370/95-06-03, Inappropriate Deferral of MOV Stem Lubrication.

This item involved the licensee's inappropriate deferral of stem lubrications specified for several GL 89-10 MOVs. The stem lubrications (specified as preventive maintenance) had been deferred from a refueling outage without recognizing that they could only be completed while the plant was shut down. This resulted in the intended lubrication frequency being exceeded. The licensee identified this improper deferral and, recognizing that missed stem lubrications could degrade MOV performance, they documented the improper deferrals for resolution through PIP 1-M95-0576. When NRC inspectors were informed of this issue, they identified it as an unresolved item for further evaluation of the safety significance and generic implications.

The inspectors assessed this item during the current inspection through a review of evaluations and corrective actions documented in completed PIP 1-M95-0576. They agreed with licensee determinations that the missed lubrications did not result in inoperability and that the

inappropriate lubrication deferrals resulted from inadequate procedures/instructions. The inspectors reviewed work orders (e. g., work orders 94030172 and 940301870) which documented the licensee's subsequent completion of the improperly deferred lubrications. Additionally, they reviewed the licensee's database listing of deferred GL 89-10 preventive maintenance as evidence that the licensee's procedures/instructions were now properly controlling deferrals. This licensee identified and corrected violation is being treated as a Non-Cited Violation, identified as NCV 50-369, 370/96-11-04: Inappropriate Deferral of MOV Stem Lubrication. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation meet the criteria specified in Section VII.B of the Enforcement Policy.

E8.4 (CLOSED) URI 50-369, 370/95-06-04, Adequacy of Actions to Address Pressure Locking and Thermal Binding.

Adequacy of the licensee's actions to address pressure locking and thermal binding will be addressed during the closure of GL 95-07. The licensee's response to GL 95-07 is discussed in section E1.1b.12.

E8.5 (OPEN) URI 50-369,370/96-10-01: Failure to Ensure Installation of Correct Heaters in FWST Enclosure

The licensee has installed the correct heaters in the enclosure and added high temperature alarms to alert control room operators of a failed thermostat that could result in excessive enclosure temperatures. The inspectors are continuing to evaluate the effects of excessive temperatures on the level and temperature transmitters. This item remains open.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Transportation of Radioactive Materials

a. Inspection Scope (86750, TI 2515/133)

The inspectors evaluated the licensee's transportation of radioactive materials programs for implementing the revised Department of Transportation (DOT) and Nuclear Regulatory Commission (NRC) transportation regulations for shipment of radioactive materials as required by 10 Code of Federal Regulations (CFR) 71.5 and 49 CFR Parts 100 through 177.

b. Observations and Findings

The inspectors reviewed procedures and determined that they adequately addressed the following: assuring that the receiver has a license to receive the material being shipped; assigning the form, quantity type, and proper shipping name of the material to be shipped; classifying waste destined for burial; selecting the type of package required; assuring that the radiation and contamination limits are met; and preparing shipping papers.

Licensee's records for the three shipments of radioactive material performed since the last inspection of this area were reviewed and the inspectors determined the shipping papers contained the required information. The inspectors also determined the licensee had maintained records of shipments of licensed material for a period of three years after shipment as required by 10 CFR 71.91(a). In addition, the licensee possessed a current certificate of approval (NRC Form 311) for their "Quality Assurance Program Description for Radioactive Material Shipping Packages Licensed Under 10 CFR 71".

c. Conclusions

Based on the above reviews, the inspectors determined that the licensee had effectively implemented a program for shipping radioactive materials required by NRC and DOT regulations.

R1.2 Occupational Radiation Exposure Control Program

a. Inspection Scope (83750)

The inspectors reviewed implementation of selected elements of the licensee's radiation protection program. The review included observation of radiological protection activities including personnel monitoring, radiological postings, high radiation area controls, and verification of posted radiation dose rates and contamination controls within the Radiologically Controlled Area (RCA). The inspectors also reviewed licensee records of personnel radiation exposure and discussed ALARA program details, implementation and goals.

b. Observations and Findings

The inspectors toured Auxiliary Building facilities, Units 1 and 2 Turbine Buildings, and selected radioactive waste storage areas. At the time of the inspection, radiological housekeeping was observed to be good. Records reviewed determined the licensee was tracking and trending personnel contamination events (PCEs). The licensee had tracked approximately 196 PCEs for 1996 which included skin and clothing contaminations. Radiologically controlled areas observed were appropriately posted and radioactive material observed was appropriately stored and labeled.

The inspectors reviewed Operational and Administrative controls for entering the RCA and performing work. These controls included the use of RWPs to be reviewed and understood by workers prior to entering the RCA. The inspectors reviewed selected RWPs for adequacy of the radiation protection requirements based on work scope, location, and conditions. For the RWPs reviewed, the inspector noted that appropriate protective clothing, and dosimetry were required. During tours of the plant, the inspectors observed the adherence of plant workers to the RWP requirements. The inspectors also performed independent radiation and contamination surveys of selected areas in the Auxiliary Building and confirmed RWP information.

The inspectors reviewed Total Effective Dose Equivalent Exposures (TEDE) for 1996 and 1997. All exposures were well below regulatory limits. No workers had received internal exposures at investigative limits in 1996 or 1997 at the time of the inspection.

The inspectors discussed ALARA goals and annual exposures with licensee management and determined the organizational structure and responsibilities for the ALARA staff were clearly defined in organizational charts. Areas reviewed included source term reduction, ALARA accomplishments, and future ALARA plans. A discussion with licensee representatives and a review of pertinent records determined the licensee had established an annual site exposure goal for 1996 of approximately 241.5 person-rem. Site exposure actually accrued in 1996 was approximately 237.1 person-rem. The site's actual 1996 exposure was based on operational exposure, completion of a Unit 1 refueling outages that began in 1995, a Unit 2 refueling outage, and the addition of two forced outages. The site's three year average through 1996 was approximately 129 person-rem per Unit. Some ALARA initiatives reviewed included the permanent installation of 50 remote radiation monitors in the Auxiliary Building, reactor coolant letdown filtration downsizing, utilization of mockups, extended controlled bursts during shutdowns, and post outage RHR system flushes.

The inspectors also observed facility preparations for the upcoming 1997 Units 1 and 2 Steam Generator (S/G) replacement outages and discussed preparations with the Radiation Protection (RP) S/G management and As Low As Reasonably Achievable (ALARA) coordinator. The inspectors toured the recently constructed Retired Steam Generator Storage Facility and reviewed the radiological aspects of facility design with RP management. In the area of Radiation Protection, the licensee planned to use basically the same management and work crews that Duke had used to perform the recent 1996 Catawba Nuclear Station Unit 1 S/G replacement outage. The licensee had incorporated RP/ALARA lessons learned into a management database to be used during the preplanning and execution of work during the 1997 S/G replacement outages. Unit 1 "B" S/G, scheduled for replacement in March of 1997, had a primary to secondary leak which the licensee was tracking. The licensee was tracking and trending the leak rate following EPRI guideline methodologies.

c. Conclusions

Radiological facility conditions and housekeeping in radioactive waste storage areas were observed to be good, material was labeled appropriately, and areas were properly posted. In addition, RP/ALARA preplanning for the upcoming S/G replacement outages was progressing satisfactorily. Radiation worker internal and external doses were being maintained well below regulatory limits and the licensee was continuing to maintain exposures ALARA.

R1.3 Water Chemistry Controls

a. Inspection Scope (84750)

The inspectors reviewed implementation of selected elements of the licensee's water chemistry control program for monitoring primary and secondary water quality. The review included examination of program guidance and implementing procedures and analytical results for selected chemistry parameters.

b. Observations and Findings

The inspectors reviewed Technical Specifications (TSs), which described the operational and surveillance requirements for reactor coolant activity and chemistry, and Final Safety Analysis Report (FSAR) Section 10.4.7.2.1. The section indicated that guidelines for maintaining reactor coolant and feedwater quality were derived from vendor recommendations and the current revisions of the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) Primary and Secondary Water Chemistry Guidelines. The FSAR also indicated that detailed operating specifications for the chemistry of those systems were addressed in the Station Chemistry Section.

The inspector reviewed selected analytical results recorded for Units 1, and 2 reactor coolant and secondary samples taken between July 1996, and December 1996. The selected parameters reviewed for primary chemistry included dissolved oxygen, chloride, fluoride, and sulfate. The selected parameters reviewed for secondary chemistry included hydrazine, iron, and copper. Those primary parameters reviewed were maintained well within the relevant TS limits and within the EPRI guidelines for power operations and cold shutdown modes.

Those secondary parameters reviewed were maintained according to station procedures with the exception of hydrazine. During a review of hydrazine levels for Unit 1 feedwater, the inspectors noted the licensee had lowered the hydrazine levels in the final feedwater system. On March 25, 1996, the licensee lowered the hydrazine levels for a period of several months to evaluate if the hydrazine levels currently specified in the Station Chemistry Manual were in excess of what was needed to provide optimum corrosion protection during power operations.

This temporary change in chemistry parameters was addressed as a trial in a memo from chemistry to engineering describing the details of the trial. In addition, the licensee was monitoring the blowdown demineralizers to validate any benefits in extending demineralizer resin capacity based on the expected lower ammonia concentrations resulting from the lower hydrazine levels. The inspectors determined the licensee conducted this test or experiment, referred to by the licensee as a trial, without a written safety evaluation that provides the bases for the determination that the test or experiment, not described in the FSAR, did not involve an unreviewed safety question. Accordingly, this failure to perform a written safety evaluation is a Violation of 10 CFR 50.59 (b) (1). This issue is identified as VIO 50-369/96-11-05; Failure to Perform a 10 CFR 50.59 Review Prior to Performing a Test or Experiment Not Described in the FSAR.

The inspectors reviewed the Problem Investigation Report (PIP) Number 2-96-3238 dated November 6, 1996. This PIP involved start-up activities associated with the addition of hydrazine with the Chemical & Volume Control System demineralizers bypassed. The preliminary results of the investigation indicated that the demineralizers were placed back in service before the hydrazine concentration had reached the planned level resulting in the solubilization and release of Cobalt-58 removed during the shutdown cleanup. The licensee estimated that between 300 and 400 Curies were released to the primary water system from the demineralizer beds. These estimates were based on remote teledosimetry measurements. This issue is a weakness in the chemistry control program, in that, the increase in primary plant source term radioactivity resulting from the solubilization of cobalt 58 will result in an increase in containment radiation levels. This issue will be tracked to verify licensee corrective actions/root cause analysis to prevent reoccurrence and is identified as Inspector Followup Item IFI 50-369/96-11-06 Followup on Licensee Closeout Actions on PIP Number 2-M96-3238.

c. Conclusions

Based on the above reviews, it was concluded that the licensee's water chemistry control program for monitoring primary and secondary water quality had been implemented, for those parameters reviewed, in accordance with the TS requirements, the Station Chemistry Manual, and the EPRI guidelines for PWR water chemistry with the exception of secondary water hydrazine addition parameter changes in Unit 1 as described above. A Violation was identified for the failure to conduct a 10 CFR 50.59 written safety evaluation prior to performing a chemistry test or experiment not described in the FSAR. An Inspector Followup Item was identified to track the closeout actions on the PIP associated with the solubilization of the Cobalt-58.

R2 Status of Radiation Protection (RP) Facilities and Equipment

R2.1 Process and Effluent Radiation Monitors

a. Inspection Scope (84750)

The inspectors reviewed selected licensee procedures and records for required surveillances on process and effluent radiation monitors and for radiation monitor availability.

b. Observations and Findings

The inspectors toured the facility to observe the physical operation of selected process and radiation monitors in use. The inspectors reviewed selected radiation and process monitor surveillance procedures and records for performance of channel checks, source checks, channel calibrations, and channel operational tests.

Performance of those surveillances was required by the TSs and/or the Offsite Dose Calculation Manual (ODCM) to demonstrate that the instrumentation was operable. Those records reviewed indicated that the surveillances were current and had been performed in accordance with the applicable procedures. The most recent system status report available, which covered the period January through November 1996, indicated that the overall availability for the Radiation Monitoring System remained at greater than 95 percent operability. The inspectors reviewed and discussed operability trending records for both safety related and non safety related monitors with the radiation monitor system engineer and engineering management.

c. Conclusions

Based on the above reviews, it was concluded that the licensee had effectively implemented procedures to track the availability of radiation monitors and to demonstrate operability of process and effluent radiation monitors by performance of surveillances at the frequencies specified in the TSs and the ODCM. Discussions with cognizant licensee personnel and a review of performance records determined the licensee had maintained an overall high level of operability for radiation monitors in 1996 and was effectively tracking monitor performance.

R2.2 Environmental Monitoring Program

a. Inspection Scope (84750)

The inspectors reviewed selected licensee procedures and records for required surveillances on environmental monitors and monitor availability.

b. Observations and Findings

The inspectors reviewed an ongoing licensee project, MG-95-0449, Environmental Sampling Deviation Reduction Plan. The plan, discussed in a previous NRC inspection report (IR 96-06), initiated actions to reduce equipment malfunctions which included: surge protection installation; heat tracing lines for freeze protection; movement, water proofing, and grounding of electrical outlets; air sampler housing physical modifications to increase air flow; and the addition of two backup portable water backup samplers and eight additional air samplers. Based on corrective actions, the licensee had reduced the number of equipment malfunctions in 1996 to 15 as of July 1996 and the inspectors determined only one malfunction had occurred during the last six months of 1996.

The inspectors selected two environmental air sample locations and one surface water sampling location. The inspectors found the equipment operable and the material condition acceptable. Environmental thermoluminescent dosimeters (TLD's) at sample locations 190, 152 and 153 were visited and the inspectors noted that they were in place; however, their position (height) above the ground varied from about 18 inches to about seven feet. When the inspectors pointed this out, the licensee noted that this could result in inconsistent background dose contributions and agreed to evaluate placement of environmental TLDs at a consistent height.

c. Conclusions

Based on a review of this area, the inspectors determined the licensee had continued to maintain effective capabilities to perform environmental samples.

R4 Staff Knowledge and Performance in Radiation Protection & Chemistry

a. Inspection Scope (83750 and 84750)

Training was reviewed to determine whether radiation protection technicians and chemistry technicians were receiving appropriate training to accomplish their work assignments.

b. Observations and Findings

The inspectors reviewed training agendas and schedules for radiation protection and chemistry technicians for 1996 and also discussed tentative training plans for 1997 with cognizant training personnel.

c. Conclusions

Based on a review of training activities reviewed, the inspectors determined radiation protection technicians and chemistry technicians were receiving an appropriate level of refresher training to support ongoing work activities.

R7 Quality Assurance in Radiological Protection and Chemistry

a. Inspection Scope (83750 and 84750)

Licensee activities and self assessment programs were reviewed to determine the adequacy of identification and corrective action programs for deficiencies in the areas of RP and Chemistry.

b. Observations and Findings

Reviews by the inspectors determined that Quality Assurance audits and Self Assessment efforts in the area of RP and Chemistry were accomplished by reviewing RP procedures, observing work, reviewing industry documentation, and performing plant walkdowns to include surveillance of work areas by supervisors and technicians during normal work coverage. Documentation of problems by licensee representatives was included in Quality Assurance Audits and Self Assessment Reports. Corrective actions were included in the licensee's Problem Investigative Process and were being completed in a timely manner.

The inspectors reviewed the initial calibration and Quality Assurance of the Wholebody Counting System. The third Quarter 1996 Interlaboratory Cross Checks and the Control Chart Calculations were reviewed and found acceptable.

c. Conclusions

The inspectors determined the licensee was effectively conducting formal RP and Chemistry audits as required by Technical Specifications and completing corrective actions in a timely manner.

F5 Fire Protection Staff Training and Qualification (64704)

Annual Fire Drill

a. Inspection Scope

On December 11, 1996, the inspector witnessed plant personnel and off-site emergency personnel response to an off-hour annual fire drill.

b. Observations and Findings

The drill scenario involved a fuel delivery truck accident which resulted in a significant postulated fire around the Unit 1 FWST area. Offsite agency response to the drill was provided by the Gilead and Cornelius Volunteer Fire Departments, which included 23 fire fighters. Five members of the onsite McGuire team also responded, and with the incident commander, acted as lead for the offsite departments. All of the stated annual fire drill objectives were satisfactorily accomplished. The inspector noted that the drill personnel realistically participated in the drill scenario. Communication between the fire location and the control room appeared to be good, with the control room operators monitoring the postulated threat to plant status through the incident commander.

The inspector also reviewed the post drill critique and concluded that the players and controllers provided good feedback on potential areas for improvement. Numerous issues were identified during the critique including:

- Several of the fire fighters that responded to support the drill were less than 18 years of age. These persons were escorted into the protected area; however, did not enter any radiation controlled areas or zones while onsite. The licensee recognized this problem and at the end of the inspection period were taking actions with the local fire departments to prevent recurrence.

- In previous drills, offsite agency responders had entered the protected area via the North gate; however, during this drill, the control room requested that security admit them at the South gate. This request was not conveyed to the responders via the 911 operator, therefore all engines initially responded to the north gate, delaying their entry.

c. Conclusions

The inspectors concluded that the drill was conducted in a realistic and professional manner. The licensee's critique of the drill was candid and identified several issues, which, once resolved, will improve the plant and offsite agency response to a fire emergency. Operators cognizance of the postulated fire threat during the drill was good.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on January 15, 1997. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Barron, B., Vice President, McGuire Nuclear Station
Boyle, J., Civil/Electrical Systems Engineering
Byrum, W., Manager, Radiation Protection
Cline, T., Senior Technical Specialist, General Office Support
Cross, R., Regulatory Compliance
Davison, Valve Supervisor
Dolan, B., Manager, Safety Assurance
Geddie, E., Manager, McGuire Nuclear Station
Herran, P., Manager, Engineering
Jones, R., Superintendent, Operations
Karriker, S., Valve Engineer (Site GL 89-10 Program Lead)
Lamb, J., Valve Engineer
Michael, R., Chemistry Manager
Nazar, M., Superintendent, Maintenance
Painter, D., Valve Engineer
Sample, M., Manager, Steam Generator Maintenance Group
Setzer, F., Valve Engineer
Snyder, J., Manager, Regulatory Compliance
Thomas, K., Superintendent, Work Control
Travis, B., Manager, Mechanical/Nuclear Systems Engineering
Tuckman, M., Senior Vice President, Nuclear Duke Power Company

NRC

S. Shaeffer, Senior Resident Inspector, McGuire
M. Sykes, Resident Inspector, McGuire
G. Harris, Resident Inspector, McGuire

INSPECTION PROCEDURES USED

IP 93702:	Prompt Onsite Response Event
IP 71707:	Conduct of Operations
IP 71750:	Plant Support
IP 62703:	Maintenance Observations
IP 61726:	Surveillance Observations
IP 40500:	Self Assessment
IP 37551:	Onsite Engineering
IP 37550:	Engineering Staff Knowledge and Performance
IP 50001:	Steam Generator Replacement Project Inspection
IP 83750:	Occupational Radiation Exposure
IP 84750:	Radioactive Waste Treatment, AND Effluent AND Environmental Monitoring
IP 86750:	Solid Radioactive Waste Management AND Transportation Of Radioactive Materials
TI 2515/133:	Implementation of Revised 49 CFR Parts 100-177 AND 10 CFR Part 71
IP 64704:	Fire Protection Program

ITEMS OPENED, CLOSED, AND DISCUSSED

OPENED	TITLE
IFI 50-369, 370/96-11-01	Actions to Address MOV Weaknesses (paragraph E1.1)
VIO 50-369/96-11-02	Failure to Implement Temporary Modification Process (paragraph E2.1)
URI 50-369, 370/96-11-03	Environmental Qualification of Safety Related Piping Grinell Snubbers in Containment (paragraph E4.3)
NCV 50-369, 370/96-11-04	Inappropriate Deferral of MOV Stem Lubrication (paragraph E8.3)
VIO 50-369/96-11-05	Failure to Perform a 10 CFR 50.59 Review Prior to Performing a Test or Experiment not Described in the FSAR (paragraph R1.3)
IFI 50-369/96-11-06	Followup on Licensee Closeout Actions on PIP Number 2-M96-3238 (paragraph R1.3)

CLOSED	TITLE
LER 50-369/96-02	Inadvertent Manual Initiation of a Unit 1 Feedwater Isolation due to an Inappropriate Action (paragraph O8.1)
VIO 50-369,370/96-08-02	Inadequate Containment Annulus Surveillance Procedure (paragraph M8.1)
NCV 50-369, 370/96-11-04	Inappropriate Deferral of MOV Stem Lubrication (paragraph E8.3)
VIO 50-369, 370/95-06-01	Analysis and Calculation Errors (paragraph E8.1)
VIO 50-369, 370/95-06-02	Errors in Entries of Test Data (paragraph E8.2)
URI 50-369, 370/95-06-03	Inappropriate Deferral of MOV Stem Lubrication (paragraph E8.3)
URI 50-369, 370/95-06-04	Adequacy of Actions to Address Pressure Locking and Thermal Binding (paragraph E8.4)

DISCUSSED	TITLE
URI 50-369,370/96-10-01	Failure to Ensure Installation of Correct Heaters in FWST Enclosure (paragraph E8.5)

LIST OF ACRONYMS USED

AFW - Auxiliary Feedwater
ALARA - As Low As Reasonably Achievable
BB - Blowdown Recycle
CA - Auxiliary Feedwater System
CF - Main Feedwater
CGD - Commercial Grade Dedication
CFR - Code of Federal Regulations
DEV - Deviation
DOT - Department of Transportation
EPRI - Electric Power Research Institute
FSAR - Final Safety Analysis Report
FWDS - Field Welds Data Sheets
FWST - Feedwater Storage Tank
GL - Generic Letter
IFI - Inspector Followup Item
INEL - Idaho National Engineering Laboratory

ENCLOSURE 2

LER - Licensee Event Report
MOV - Motor-Operated Valve
MPM - Motor Power Monitor
MSLB - Main Steam Line Break
NCV - Non-Cited Violation
NDE - Nondestructive Examination Radiography
NRC - Nuclear Regulatory Commission
NRR - NRC Office of Nuclear Reactor Regulation
ODCM - Offsite Dose Calculation Manual
PCE - Personnel Contamination Event
PIP - Problem Investigation Process
PM - Preventative Maintenance
PMT - Post Maintenance Test
PORV - Power Operated Relief Valve
PQR - Procedures Qualification Record
PRA - Probabilistic Risk Assessment
QA - Quality Assurance
RCA - Radiologically Controlled Area
RCS - Reactor Coolant System
RP - Radiation Protection
RTD - Resistance Thermal Detector
RWST - Refueling Water Storage Tank
SG - Steam Generator
SGRP - Steam Generator Replacement Program
SFP - Spent Fuel Pool
TI - Temporary Instruction
TIG - Tungsten Inert Gas
TS - Technical Specification
URI - Unresolved Item
VOTES - Valve Operation Test and Evaluation System
VIO - Violation
WO - Work Order
WPCS - Work Process Control Sheets