U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Inspection Report 50-244/92-15 License: DPR-18

Facility: R. E. Ginna Nuclear Power Plant
Rochester Gas and Electric Corporation (RG&E)

Inspection: September 9 through October 26, 1992

Inspectors: T. A. Moslak, Senior Resident Inspector, Ginna
E. C. Knutson, Resident Inspector, Ginna

Approved by: John T. Shedlosky for 11/2/92
W. J. Lazarus, Chief, Reactor Projects Section 3B Date

INSPECTION SCOPE

Plant operations, radiological controls, maintenance/surveillance, security, emergency preparedness, engineering/technical support, and safety assessment/quality verification.

INSPECTION OVERVIEW

Plant Operations: The plant operated at approximately 98% power throughout the inspection period. No challenges to stable operation occurred.

Radiological Controls: Prior to entry into the reactor building, the site Health Physics staff provided detailed ALARA briefings to assure personnel were knowledgeable of dose reduction measures and that tasks were effectively coordinated. Corporate engineering is working closely with the site Health Physics staff to evaluate various ALARA program initiatives.

Maintenance/Surveillance: Strong engineering support and good coordination between the maintenance shops was evident in response to a degrading fuel oil booster pump discharge pressure on the "B" EDG.

Security: Compensatory measures have been effective to support site security system upgrades.

Emergency Preparedness: No programmatic weaknesses were identified during the annual emergency preparedness exercise.

Engineering/Technical Support: Corporate engineering has accelerated replacement of the containment air coolers with an upgraded design to the 1993 refueling outage, vice the 1995 outage.

Safety Assessment/Quality Verification: Administrative controls were ineffective in identifying an error in a minor change to a procedure, resulting in the mispositioning of a locked valve (non-cited violation). The PORC and NSARB have been effective in providing a safety oversight of plant activities.

TABLE OF CONTENTS

Table of Contents

DETAILS

1.0 PLANT OPERATIONS (71707)

1.1 Operational Experiences

The plant operated at approximately 98 percent power throughout the inspection period. No challenges to stable operation occurred.

1.2 Control of Operations

Overall, the inspectors found the R. E. Ginna Nuclear Power plant to be operated safely. Control room staffing was as required. Operators exercised control over access to the control room. Shift supervisors consistently maintained authority over activities and provided detailed turnover briefings to relief crews. Operators adhered to approved procedures and were knowledgeable of off-normal plant conditions. The inspectors reviewed control room log books for activities and trends, observed recorder traces for abnormalities, assessed compliance with Technical Specifications, and verified equipment availability was consistent with the requirements for existing plant conditions. During normal work hours and on backshifts, accessible areas of the plant were toured. No operational inadequacies or concerns were identified.

2.0 RADIOLOGICAL CONTROLS (71707)

2.1 Routine Observations

The inspectors periodically confirmed that radiation work permits were effectively implemented, dosimetry was correctly worn in controlled areas and dosimeter readings were accurately recorded, access to high radiation areas was adequately controlled, and postings and labeling were in compliance with procedures and regulations. Through observations of ongoing activities and discussions with plant personnel, the inspectors concluded that radiological controls were conscientiously implemented.

2.2 ALARA Briefings

2.2.1 Containment Recirculation Fan Cooler Repairs

On Monday, September 14, 1992, the inspector attended the pre-job ALARA (As Low As Reasonably Achievable) briefing for repair of service water system leaks in the "C" containment recirculation fan cooler. Following repair of one leak over the weekend, three additional service water system leaks had been identified in the "C" recirculation fan cooler; consequently, the scope of this maintenance had increased with little time for in-depth job preparation prior to the briefing. The health physics supervisor appropriately delayed the start of the meeting until knowledgeable mechanical maintenance supervision was available. Discussion was open and constructive, addressing maintenance task planning and ALARA considerations. The inspector considered that the briefing was valuable and that ALARA concerns were both appropriately addressed and adequately resolved.

2.2.2 Multi-task Reactor Building Entry

On September 30, 1992, the inspector observed an ALARA briefing presented by the site Health Physics and Safety Departments to personnel entering the reactor building. This particular entry involved coordination of large diversified work groups, totaling 23 individuals, from corporate engineering, operations, the results and test section, and maintenance department. Tasks that were performed by these groups in containment included a premodification walkdown of the "C" recirculation fan cooler, general inspection of equipment material condition, quarterly surveillance test of post-accident charcoal filter system dampers (PT-2.3.1), and pressurizer snubber inspections. Detailed presentations were made using visual aids to assure personnel were aware of travel paths, low dose waiting areas, high dose areas, the proper wearing of personal dosimetry, heat stress concerns, and contamination control measures. From this briefing, the inspector determined that the Health Physics Department effectively sequenced the jobs, provided the coverage necessary for dose intensive tasks, and overall, controlled the entry as required by administrative (A) procedure A-3, "Containment Vessel Access Requirements." Through discussions with licensee personnel and review of relevant records following the entry, the inspector concluded that the reactor building entry was well planned and measures were implemented to minimize personnel dose.

2.3 Corporate ALARA Subcommittee Meeting

On September 24, 1992, the inspector attended a Corporate ALARA Subcommittee Meeting. The purpose of the meeting was to review radiological controls effectiveness and evaluate additional measures that could be used to minimize worker radiation exposure. Meeting topics included review of radiological controls performance trends, report on the Cobalt Reduction Program, proposed measures to improve the decontamination of the reactor refueling cavity, discussion of a Westinghouse Owners Group Report addressing corrosion control and dose reduction gained through zinc addition to the reactor coolant system, and implementation status of a laser disk, surrogate tour system for the Ginna facility.

The inspector concluded that the meeting reflected a proactive approach by RG&E corporate management to evaluate engineering controls to minimize worker dose by integrating corporate engineering support into the routine site Health Physics program.

2.4 Licensee Findings on Previous Inspection Findings

2.4.1 (Closed) Inspector Follow Item (50-244/92-02-02) Accountability of Unmonitored Releases Via the Steam Generator Atmosphere Relief Valves

In response to this item, the licensee developed and implemented primary chemistry (PC) procedure PC-12-1, "Guideline for Health Physics Actions for Off-Normal Plant Operating Condition," effective September 4, 1992. The inspector reviewed this procedure and determined that it contained the detailed steps necessary to perform sampling, analysis, and

reporting of radioactive effluent releases from various pathways including the steam generator atmospheric relief valves. The inspector concluded that adequate measures are in place to assure accountability of releases through unmonitored pathways. This item is closed.

3.0 MAINTENANCE/SURVEILLANCE (62703, 61726, 71710)

3.1 Corrective Maintenance

3.1.1 "B" Emergency Diesel Generator Low Fuel Oil Pressure

On September 16, 1992, during the conduct of monthly surveillance testing (PT-12.2) on the "B" emergency diesel generator (EDG), the final value of fuel oil pressure (recorded after one hour of operation at 2000 KW) was in the low alert range (34 psig, alert range: 33-34 psig). This parameter had also been in the low alert range (33.5 psig) during the previous monthly surveillance test. On that occurrence, subsequent calibration of the fuel oil discharge pressure gauge, PI-2856, had shown the indication to have drifted low. With the resultant correction applied, the value obtained during the surveillance test was within the normal range, and therefore no additional action was taken. Following the September 16 surveillance test, a calibration check was again performed on the fuel oil discharge pressure gauge; in this case, however, as-found calibration was satisfactory, indicating that a fuel oil system mechanical problem existed.

Through review of EDG historical information, maintenance personnel determined that a fuel oil system problem which displayed similar symptoms (normal fuel oil pressure under no-load conditions, but low fuel oil pressure at full-load) had developed with the "A" EDG in 1988. In that case, the problem had been a degraded fuel oil booster pump. Based on this experience, maintenance supervision chose to replace the "B" EDG fuel oil booster pump as the next step in troubleshooting.

Above cold shutdown, technical specification 3.7.2.2 allows one EDG to be inoperable for seven days before requiring the plant to be returned to cold shutdown. Additionally, it requires that operability of the remaining EDG be verified by startup from normal standby conditions at least once every 24 hours. This technical specification action statement was entered at 6:14 AM on September 29, 1992, when the "B" EDG was declared inoperable for replacement of the fuel booster pump.

Fuel oil booster pump replacement was performed under work order 9221570, "Diesel Generator B - Replace Fuel Booster Pump", per maintenance procedure M-15.1, "A or B Diesel Generator Inspection and Maintenance", revision 41, dated April 17, 1992, PCN 92T-721. During subsequent testing, both no-load and full-load fuel oil pressures were found to be lower than they had been before the pump replacement. Additional investigation revealed a problem with the fuel oil regulating valve, V-5926; specifically, the lock nut on the set point adjustment screw had worked loose. This had allowed engine vibration to rotate the adjustment screw such that the regulating pressure setpoint had gradually been reduced.

Work order 9201860, "Replace "B" D/G Fuel Oil Reg/Relief V-5926", was generated to support further corrective action. With the EDG running at no-load, V-5926 setpoint was adjusted and the adjusting screw was locked per maintenance procedure M-37.38.1, "Safety and Relief Valve Inspection and Maintenance for Valve No. 5926", revision 19, dated April 2, 1992, PCN 92T-728. Subsequent acceptance testing was satisfactory and the "B" EDG was declared operable on September 30, 1992.

The inspector reviewed the completed work packages for the booster pump replacement and the regulating valve setpoint adjustment. In both cases, the majority of job-specific instructions were contained in the temporary procedure change notices (PCNs). The inspector observed that these temporary changes had been developed and approved in accordance with administrative procedure A-601.3, "Procedure Control - Temporary Changes", and that the instructions were appropriate to the work to be performed. The inspector also reviewed the licensee's root cause analysis and proposed long term corrective actions delineated in RG&E Inter-Office Memorandum, "Diesel Generator Fuel Oil Pressure Degradation Cause Analysis." Loosening of the lock nut on the setpoint screw was attributed to a combination of the penetrating quality of diesel fuel oil and the vibration associated with operation of a large reciprocating engine; setpoint drift was similarly attributed to engine vibration. Corrective action consisted of 1) in the near term, verifying lock nuts tight on the set point adjusting screws for similar valves on both EDGs (work order 9221610, two of three valves complete), and 2) as permanent corrective action, adding these checks to the procedure for annual EDG maintenance. The inspector considered that the root cause analysis and resultant corrective actions were appropriate and thorough.

In summary, prompt, proactive corrective action was taken in response to degrading fuel oil booster pump discharge pressure on the "B" EDG. The decision to replace the fuel oil booster pump as an early step in troubleshooting was soundly based on historic information and reflected a well-established maintenance data collection program. Strong engineering support and good coordination between the shops was evident when the scope of work shifted to include the fuel oil regulating valve. Technical specification action statement requirements for an inoperable EDG were satisfied. Root cause analysis and corrective actions were comprehensive and well-considered. The inspector had no additional concerns on this matter.

3.2 Surveillance Observations

Inspectors observed portions of surveillances to verify proper calibration of test instrumentation, use of approved procedures, performance of work by qualified personnel, conformance to Limiting Conditions for Operation (LCOs), and correct system restoration following testing. The following surveillances were observed:

1. Performance Test (PT)-2.2M, "Residual Heat Removal System - Monthly", revision 1, dated June 8, 1990, observed on September 14, 1992

    -- The inspector determined that the surveillance was properly conducted.

2. PT-12.2, "Emergency Diesel Generator 1B," revision 71, dated April 17, 1992, procedure change notice (PCN) number 92T-712, observed on September 16, 1992

-- The final value of fuel oil pressure (recorded after one hour of operation at 2000 KW) was in the low alert range (34 psig, alert range: 33-34 psig). Corrective action is discussed in section 3.1.1 of this report. The inspector identified no additional concerns.

3. PT-9.1.14, "Undervoltage Protection - 480 volt Safeguard Bus 14", revision 6, dated June 19, 1992, observed on September 18, 1992

-- Administrative (A) procedure A-1408, "Independent Verification", paragraph 3.5.1 states, "The normal method is by use of double procedure step signoffs, with the first signoff documenting the actual manipulation to realign and the second signoff documenting the independent check that verifies the realigned position. In some cases, the verification signoff line may appear at a different location in the procedure from the task completion signoff line." Independent verification in PT-9.1.14 is performed using the second format. From prior observations, however, the inspector noted that independent verification using this format is sometimes performed as a two-party check; that is, subsequent to the procedure step which realigned the equipment to its normal configuration, equipment status is independently verified and signed for by two operators. The inspector was concerned that inconsistency in the method of accomplishing independent verifications could lessen the value of these checks.

-- Conduct of the surveillance was observed by an RG&E QC inspector. Results of the licensee inspection were documented in Quality Control Report 92-0818. The RG&E inspector also noted the single signoff of independent verification steps as a deficiency. As a result, independent verification documentation is to be further examined under corrective action report (CAR) 1990, "System Alignments and Independent Verification, NRC Inspection Reports 89-15 and 89-16". The RG&E QC inspector noted no other deficiencies.

-- The licensee is in the process of developing an action plan to address the topic of independent verification. In that this concern 1) was independently identified by RG&E, and 2) is being positively tracked by their corrective action reporting system, the inspector considered that no specific additional NRC action was required.

4. PT-32B, "Reactor Trip Breaker Testing - "B" Train", revision 11, dated September 22, 1992, PCN 92T-716, observed September 22, 1992.

-- The inspector determined that the test was properly conducted.

3.3 Component Cooling Water System Walkdown (71710)

The inspector conducted a detailed walkdown of a representative sample of the accessible portions of the component cooling water (CCW) system. Primary emphasis was placed on inspection of system material conditions for items that might degrade plant performance. Items examined included installation of hangers and supports, housekeeping, material condition of valves, correct valve positions, and component labeling. No safety-significant deficiencies were noted; a number of minor material deficiencies (such as missing valve packing gland nuts and loose instrument line wall mountings) were discussed with the licensee. The inspector concluded that the material condition of the CCW system was satisfactory and verified that the system was operable.

4.0 SECURITY (71707)

4.1 Routine Observations

During this inspection period, the resident inspectors verified that x-ray machines and metal and explosive detectors were operable, protected area and vital area barriers were well maintained, personnel were properly badged for unescorted or escorted access, and compensatory measures were implemented when necessary. Adequate compensatory measures were provided to support ongoing site security upgrade modifications. No unacceptable conditions were identified.

5.0 EMERGENCY PREPAREDNESS (71707)

5.1 1992 Annual Emergency Preparedness Exercise

The inspectors participated as members of the NRC inspection team during conduct of the 1992 R. E. Ginna Nuclear Power Plant partial participation emergency preparedness exercise on October 8, 1992. Other participants included emergency response organizations in Wayne and Monroe counties (resident and adjacent counties, respectively), and the state of New York. No significant deficiencies were noted; detailed assessment of the exercise is presented in inspection report 50-244/92-013.

5.2 Status of Emergency Preparedness Program

On September 17, 1992, members of the RG&E Emergency Preparedness (EP) Department met with members of the NRC regional staff in King of Prussia, Pennsylvania. The purpose of the meeting was to discuss the status of various aspects of the EP program.

6.0 ENGINEERING/TECHNICAL SUPPORT (71707, 92701)

6.1 Spent Fuel Examination

During the period September 3-15, 1992, the licensee performed examinations of two irradiated test fuel assemblies. These assemblies, XT03 and XT04, were fabricated by the Siemens Power Corporation (SPC), and tested at Ginna as part of a joint research project

between RG&E, SPC, and the Empire State Electric Energy Research Corporation (ESEERCO). The purpose of the project was to test advances in fuel rod design and fabrication. During the period 1985 to 1990, these two assemblies were irradiated over multiple fuel cycles (five for XT03 and four for XT04). The primary purpose of the September 1992 examination was to prepare selected fuel rods from these test assemblies for off-site laboratory testing. This required that the fuel rods be removed from the assembly and cut into shorter segments to facilitate shipment.

The inspector reviewed the work package for examination of fuel assemblies XT03 and XT04, which consisted of the following:

-- RG&E Inter-Office Correspondence, "Fuel Examination", from John Cook to Operations Department, dated August 20, 1992

-- Refueling (RF) Procedure RF-8.4, "Fuel and Core Component Movement in the Spent Fuel Pit", revision 39, dated May 29, 1992

-- RF-42.2, "SNP Examination of Fuel Assembly XT03 and XT04", revision 4, PCN 92T-705, dated August 26, 1992

-- Siemens Power Corporation, Document No. EMF-1497, "Procedure for Destructive Testing of Ginna/ESEERCO Lead Assemblies XT03 and XT04 Containing Barrier Cladding and Annular Pellets at Ginna September 1992", revision 1, dated August 21, 1992

Technical specification 3.11.1 addresses auxiliary building ventilation requirements during handling of fuel assemblies; however, these requirements only apply if the fuel stored in the spent fuel storage pool has decayed for less than 60 days. Similarly, technical specification 3.11.3, addressing load restrictions on movement of canisters containing consolidated fuel rods within the spent fuel storage pool, does not apply if the spent fuel rack beneath the transported canister contains only spent fuel that has decayed at least 60 days. Since reactor refueling was last completed in May 1992, there were no technical specifications which uniquely applied to the spent fuel examination. Although modified to take advantage of these technical specification allowances, the licensee procedures in use were otherwise the same as would be used during reactor refueling operations and adequately addressed safety considerations for handling spent fuel. Additionally, the fuel assemblies had decayed for approximately two years, which significantly lowered the risk associated with gas release from a cladding breach. The fuel rods had been designed to be segmented, with cuts to be made through spacer material rather than fuel; therefore, the cladding integrity was preserved throughout the cutting operations. Actions specified in the event of cladding damage during cutting operations were clearly addressed and appropriate. The inspector noted no significant deficiencies in the work package.

On September 15, 1992, the inspector observed fuel rod cutting operations. Actual operations were conducted by contractor personnel, under supervision of an RG&E engineer. The technician used video from an in-pool camera to position and operate the cutting tool and manipulator arm. The inspector observed the operation to be deliberate and well controlled. No deficiencies were noted.

In summary, engineering preparation for examination of fuel assemblies XT03 and XT04 was thorough. Site engineering support was readily available throughout the operation. No significant deficiencies were noted in the conduct of work or in the completed work package.

6.2 Replacement of Containment Recirculation Fan Cooler Heat Exchangers

The reactor containment building contains four recirculation fan cooler units. The safety function of these units is to condense steam released during either a loss of coolant or steam line break accident, and thereby limit the time that the containment building is subject to high internal pressure. Each unit contains three separate cooling coils that are supplied by the service water system.

Over the past year, leaks have been developing in the recirculation fan cooler heat exchangers at an increasing rate; inspection reports 50-244/92-10, 92-09, and 91-23 provide the history of recent service water system leaks and repair activities. During this inspection period, the "C" containment recirculation fan cooler was twice declared inoperable due to service water leaks in the primary heat exchanger. As before, corrective action consisted of removing the leaking heat exchanger tubes from service by cutting them at the supply/return headers and then plugging the stub tubes.

Containment recirculation fan cooler heat exchangers had been scheduled for replacement in 1995; however, in light of the recent trend of service water leaks, RG&E has accelerated this schedule, with replacement of all service water heat exchangers in the four units to occur during the 1993 refueling outage. The replacement primary heat exchangers will be of an advanced design, with greater erosion/corrosion resistance and a physical configuration which will provide both better access for maintenance and greater flexibility in selecting repair techniques.

In conclusion, the engineering department has demonstrated continued responsiveness in light of declining service water system performance in the containment recirculating fan coolers.

7.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (90712, 90713, 92701, 40500)

7.1 Inadvertent Addition of Makeup Water to the Spent Fuel Pool

On October 9, 1992, while conducting rounds, an Auxiliary Operator (AO) observed that water level in the spent fuel pool (SFP) appeared to be about four inches higher than normal. Upon informing the Control Room, the on-shift operations staff determined that a cross-connect valve (V-788A) between the condensate transfer system and the primary makeup water system was out of its correct position by using available drawings and system alignment

procedures. The valve had been repositioned from its normal locked closed position to the locked open position 21 days earlier, during realignment of the SFP cooling system in accordance with procedure S-9X, "Placing Spent Fuel Pool Purification and/or Cooling System B in Service". This created a flow path between the condensate transfer system and the SFP, although normal operation of the condensate transfer system did not develop sufficient pressure to establish flow into the SFP, and a check valve prevented flow out of the SFP. An infrequently-used procedure, T-6.12, to fill the chemistry laboratory deionized water storage tank, was performed on October 8, 1992, and resulted in the transfer of water to the SFP.

The mispositioning of V-788A occurred as the result of an error in the most recent change to the procedure for transferring operation of SFP cooling loops (S-9X). Procedure change notice (PCN) 92-3521 had been initiated in June, 1992, to update S-9X to recent valve designation changes. In accordance with administrative procedure A-601.2, "Procedure Control - Permanent Changes," the change request was forwarded by the initiator to the responsible operations department manager for review. Along with reviewing this request, the operations department manager inserted another change to include the position of V-788A in the procedure. Although actual operation of V-788A is not required during the transfer of in-service SFP cooling loops, it was included in the proposed change to S-9X because the operations department had determined that no other operating procedure established the position of V-788A. In preparing this additional change, the operations department manager correctly specified the position of V-788A as locked closed on the PCN coversheet; however, the draft change as entered on the applicable page in the procedure incorrectly indicated that the valve was to be locked open. As specified by A-601.2, the position of V-788A in S-9X was not required to be reviewed by any other individuals prior to presentation to PORC, because this was a minor administrative change. In accordance with A-601.2, the change was presented to PORC only as a summary in the draft meeting minutes. The error was therefore not detected, and the change was approved by PORC on August 6, 1992.

The SFP contains approximately 258,000 gallons of borated water. Technical specification 5.4.6 requires SFP boron concentration be maintained at 2000 ppm or greater, based on shutdown margin considerations for a postulated fuel handling accident; with this consideration eliminated (i.e., all fuel stored in the storage racks), the design of the SFP is such that adequate shutdown margin will be maintained even if the pool were filled with pure water. The SFP also has a high level alarm to alert operators to such an abnormal condition. The amount of water added in this case was insufficient to actuate the high level alarm. Based on the level change in the condensate storage tank which occurred during the procedure to fill the chemistry laboratory deionized water storage tank, the licensee estimated that approximately 2200 gallons of pure water had been transferred to the SFP. The observed increase in SFP level of 3.5 to 4.5 inches was consistent with this estimate. The SFP was sampled for boron concentration; the 15 parts per million (ppm) decrease from an initial concentration of 2330 ppm further supported the estimated volume of pure water added.

Chemical and radiological analyses of condensate samples indicated that no backleakage from the SFP into the condensate transfer system had occurred.The inadvertent addition of pure water to the SFP was therefore of minimal safety significance.

As immediate corrective action, a temporary change to S-9X was generated to reposition V-788A to the locked shut position.

In response to this incident, site operations management promptly directed that a Human Performance Enhancement System (HPES) evaluation be performed to identify causes contributing to the valve mispositioning. As immediate response, all operating shifts were informed of the nature of the incident and the lessons learned. Operations management acknowledged the astute observation by the Auxiliary Operator in identifying the small SFP level change. The AO's performance exemplified keen attention-to-detail while performing routine rounds. As long term corrective action, changes to administrative procedures are being evaluated to improve the control of minor procedure revisions. These actions include requiring a third independent reviewer, subsequent to reviews by the procedure change initiator and the Responsible Manager, to assure the accuracy of inconsequential procedure changes.

The inspectors reviewed the circumstances resulting in the mispositioning of V-788A and the licensee's response. The inspectors concluded that the mispositioning resulted from a human error when transcribing information from the procedure change form to the revised procedure. Present licensee administrative controls were ineffective to identify the error prior to implementation. Such weakness is contrary to the requirements of 10 CFR 50, Appendix B, Criterion VI, "Document Control" which states in part, that, "measures shall be established to control the issuance of documents, such as...procedures...including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy..." However, under the provisions of 10 CFR 2, Appendix C, Section VII.B a violation was not cited because the relevant criteria were met. These criteria include the licensee promptly identified the plant condition, the condition was of minor safety significance, immediate corrective actions were appropriate to prevent a recurrence, the condition was evaluated for reportability, and the occurrence was not identified as willful or as preventable through corrective actions taken on a prior violation.

7.2 Periodic Reports

Periodic reports submitted by the licensee pursuant to Technical Specification 6.9.1 were reviewed. Inspectors verified that the reports contained information required by the NRC, that test results and/or supporting information were consistent with design predictions and performance specifications, and that reported information was accurate. The following report was reviewed:

-- Monthly Operating Report for September 1992

No unacceptable conditions were identified.

7.3 Plant Operations Review Committee (PORC) Meeting

On October 21, 1992, the inspector attended a regularly scheduled weekly meeting of the Plant Operations Review Committee (PORC). Items discussed included proposed compensatory measures for an upcoming planned outage of the primary meteorological instrument tower and a proposed technical specification amendment request regarding containment isolation boundaries. Discussion among committee members was both uninhibited and unbiased by concerns for production. The inspector concluded that the PORC was effective in meeting its technical specification requirement to advise the plant manager on matters related to nuclear safety and for referral of appropriate matters to the Nuclear Safety Audit and Review Board.

7.4 Nuclear Safety Audit and Review Board (NSARB)

The inspector attended the NSARB meetings held on October 13 and 14, 1992. Topics included review of proposed changes to containment isolation technical specification 3.6.3, review of training program audit results/status of corrective actions, review of PORC minutes, and review of the Quality Assurance assessment into the verification of plant records (IE Notice 92-30). The board discussed in great detail the proposed actions to formalize auxiliary operator watchstanding practices and assigned follow-up items to site operations management that will be addressed in the next NSARB meeting. The inspector concluded that topics were candidly discussed with sufficient depth for board members to assess the safety significance of the agenda issues.

Overall, the inspector determined that the licensee complied with the requirements of Technical Specification 6.5.2 regarding NSARB composition, qualifications, meeting frequency, and topics of review.

7.5 Quality Assurance/Quality Control (QA/QC) Subcommittee Meeting

On September 24, 1992, the inspector attended the quarterly meeting of the RG&E QA/QC subcommittee. The inspector determined that senior corporate management was adequately briefed on the status of audit findings, performance indicator trends, quality performance program status and plans, and the progress of internal self-assessments. Areas requiring increased attention were identified by the participants and action items assigned. Particular attention was given by RG&E management to discriminating outage related data from normal operational data when interpreting performance indicator trends. Additionally, upon being informed of the results of a recent QA surveillance regarding the accountability and watchstanding practices of auxiliary operators, senior RG&E management tasked site operations and Quality Performance Department management with improving administrative controls and oversight, respectively, in these areas.

From this meeting, the inspector concluded that the QA/QC subcommittee was an effective management tool for the self-identification and timely resolution of programmatic issues.

8.0 ADMINISTRATIVE (71707, 30702, 94600)

8.1 Backshift and Deep Backshift Inspection

During this inspection period, backshift inspection was conducted on October 12, 1992. Deep backshift inspections were conducted on the following dates: September 13, 19, 20, 27, October 18, 20, and 24, 1992.

8.2 Exit Meetings

At periodic intervals and at the conclusion of the inspection, meetings were held with senior station management to discuss the scope and findings of inspections. The exit meeting for inspection report 50-244/92-14 (radiological and non-radiological chemistry inspection, conducted September 15-18, 1992) was held by Mr. J. Kottan on September 18, 1992. The exit meeting for inspection report 50-244/92-13 (emergency preparedness inspection, conducted October 8, 1992) was held by Mr. L. Eckert on October 9, 1992. The exit meeting for inspection report 50-244/92-15 was held on October 27, 1992.