U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Inspection Report 50-244/95-08 License: DPR-18

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

Inspection: May 7, 1995 through June 17, 1995

Inspectors: T. A. Moslak, Senior Resident Inspector, Ginna
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Approved by: James Scott Stewart for 7/7/95
W. J. Lazarus, Chief, Reactor Projects Section 3B Date

INSPECTION SCOPE

Plant operations, maintenance, engineering, plant support, and safety assessment/quality verification.

INSPECTION EXECUTIVE SUMMARY

Operations

Operator response to the IA header leak was excellent. By rapidly locating and isolating the leak, operators averted a reactor trip due to low steam generator water level. Despite limited procedural guidance on reactor plant control during a sustained loss of IA, operators continued to demonstrate excellent coordination and skill in reducing plant power with limited means to control RCS pressure and inventory. Maintenance support was promptly available and the IA leak was quickly repaired. Licensee review of the event was ongoing at the close of the inspection period.

Maintenance

On May 15, 1995, during routine surveillance testing of the B-emergency diesel generator (EDG), diesel exhaust entered the technical support center (TSC) ventilation system and caused a fire alarm. This occurred due to the close proximity of the EDGs to the TSC, and the wind direction. To avoid false alarms in the future, the test procedures for both EDGs were changed to include having the TSC fire alarm defeated prior to operating the EDG. Additionally, an engineering work request was initiated to evaluate the possible detrimental effect of diesel exhaust on the charcoal filters that are used during TSC emergency ventilation.

Engineering

On May 23, 1995, the loop-A cold leg resistance temperature detector (TE-402B) for reactor protection system channel 2 average temperature and differential temperature instruments failed. Troubleshooting revealed that the cause was an open circuit in the detector. Since TE-402B is an immersion RTD, repair while at power is not possible. However, the licensee determined that another RTD installed in loop-A, TW-450, could provide equivalent input to the affected RPS instrument channel. TW-450 is physically identical to TE-402B, but is mounted in a thermowell. The inspector considered that the licensee adequately evaluated the effect of the increased lag time on RPS performance prior to conducting this temporary modification. The inspector concluded that the licensee's actions to temporarily connect TW-450 to substitute for TE-402B had been appropriate and were well developed.

Plant Support

Routine observations in the areas of radiological controls, security, and fire protection indicated that these programs were effectively implemented.

Safety Assessment/Quality Verification

The Plant Operations Review Committee (PORC) ensured that measures to provide for adequate plant control were in place prior to the start of corrective maintenance to repair the IA leak in containment.

At a meeting of the Nuclear Safety Audit and Review Board, the inspector observed that a broad scope of topics were well developed and knowledgeably discussed.

TABLE OF CONTENTS

DETAILS

1.0 OPERATIONS (71707)

1.1 Operational Experiences

At the beginning of the inspection period, the plant was operating at full power (approximately 97 percent). On June 7, 1995, power was reduced to approximately 20 percent in response to a loss of instrument air due to a system leak in containment. Following completion of repairs, the plant was returned to full power operation the following day. There were no other significant operational events or challenges during the inspection period.

1.2 Control of Operations

Control room staffing was as required. Operators exercised control over access to the Control Room. Shift supervisors 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 working hours and on backshifts, accessible areas of the plant were toured. No operational inadequacies or concerns were identified.

1.3 Instrument Air System Leak in Containment

At 6:58 p.m. on June 7, 1995, a leak occurred in the instrument air (IA) system piping in containment. The source of the leak was a separated solder joint at a 90-degree elbow in the two-inch header downstream of the containment isolation valves. Operators were alerted to the problem when main control board annunciator H-8, "Instrument Air Low Pressure," alarmed, and entered alarm procedure AP-IA.1, "Loss of Instrument Air." IA system pressure continued to decrease, and as a result, serviced components throughout the plant began to shift to their IA-vented condition. The most immediate operational effect was that the main feedwater regulating valves (FRVs) began to drift shut, resulting in lowering steam generator (SG) water levels.

Approximately five minutes after the leak occurred, operators isolated IA to containment as directed by AP-IA.1. This stopped the leak, and pressure within the intact portion of the IA system was rapidly restored to normal. With motive force restored, the FRVs began to reopen, and continued beyond their steady state positions as the automatic feedwater control system attempted to restore SG level to normal. Due to the large magnitude error signal that had developed between valve position and water level, and the response time of the automatic feedwater control system relative to the rate of SG water level increase, water levels in both SGs overshot the desired value quickly reaching the engineered safety features (ESF) feedwater isolation setpoint of 67 percent. Operators manually adjusted both FRVs to reduce feedwater flow and restore water levels to normal.

Loss of IA in containment had a significant effect on reactor coolant system (RCS) pressure and inventory control. The pressurizer spray valves and power

operated relief valves (PORVs) would no longer operate in automatic. Additionally, the letdown portion of the chemical and volume control system isolated due to closure of air operated valves. Letdown provides for water removal from the RCS and normally operates at equilibrium with the coolant charging pumps to maintain a constant pressurizer level. With coolant charging still required to supply reactor coolant pump seal injection, the result of operating with letdown secured was that pressurizer level gradually increased. As pressurizer level increased, the steam bubble compressed, causing RCS pressure to increase. To counter the rising pressurizer level, operators commenced a power reduction in accordance with operating procedure O-5.1, "Load Reductions," to reduce RCS temperature. Since pressurizer spray was not available, operators took manual control of pressurizer heaters to improve control of RCS pressure.

In parallel with the power reduction, a containment entry was conducted to determine the location of the IA system leak. Maintenance personnel then performed a temporary repair by reassembling the separated elbow and securing it with clamps. The repair was tested satisfactorily and IA was reestablished to containment approximately two hours after the leak occurred. Operators stopped the power reduction and stabilized plant conditions at approximately 17 percent reactor power.

Due to the location of the leak, isolation for permanent repair would again isolate IA from all components in containment. However, utilizing a test connection on the remaining pressurized portion of the IA header inside containment, maintenance personnel determined that temporary hoses could be installed to provide IA to critical components during the repair. Operations personnel determined that maintaining RCS letdown during the repair would provide the greatest plant stability. Prior to the start of repair activities, PORC met to review the event, and to discuss plant operation and repair options. PORC concurred with installing a temporary modification to supply IA to those components necessary to maintain RCS letdown, and concluded that the plant could safely continue to operate at power during the repair.

To repair the IA header in containment, a new section of piping (consisting of two short sections of piping soldered into an elbow) was fabricated in the shop. Following installation of the temporary modification to maintain RCS letdown, IA was isolated from the work area and the failed section was removed by cutting the piping on either side of the elbow. The prefabricated section was then joined to the installed piping with union couplings and soldered in place.

Repair of the IA header was completed on the morning of June 8, 1995, and IA was restored to containment at 4:15 a.m. Operators commenced a power escalation at 4:22 a.m., and full power was reached at 1:40 p.m.

The inspector observed the latter portion the power reduction and the establishment of stable plant conditions after completion of the temporary repair and restoration of IA to containment. The inspector noted excellent coordination between operations personnel in controlling pressurizer level, RCS pressure, and main generator load during the power reduction.

The inspector reviewed operator logs, data archived from the plant computer, and interviewed on-shift operations department personnel to assess operator response to the event. The inspector concluded that the operators had displayed good plant knowledge in rapidly determining the location of the IA leak by integrating reports from personnel outside the control room with indications in the control room. The inspector noted that the load reduction had not been initiated in response to lowering steam generator water level at the onset of the event, but rather was commenced approximately 10 minutes into the event. The first step of AP-IA.1 states that, "If steam generator levels are drifting low, then decrease load as necessary to stabilize levels." However, the inspector considered it reasonable that a load reduction had not been commenced, given that IA was restored within a period of only five minutes.

The inspector noted that AP-IA.1 contained very little guidance on continued operation with IA to containment isolated. Step 17.b states, "If IA to containment is isolated, then neither normal nor auxiliary spray is available. Rapid power reductions should be avoided." Appendix 2 to AP-IA.1 lists affected components and their mode of failure for loss of IA to different portions of the plant, but provides no operational guidance. The inspector could locate no other specific procedural guidance for RCS pressure control with IA to containment isolated (that is, with the pressurizer spray valves inoperable, letdown isolated, and charging in operation). The inspector noted, however, that operators had experienced similar scenarios during simulator training.

The inspector reviewed the completed work package that was used to conduct the repair of the failed IA header (work order 19502101). The IA system is considered a balance-of-plant system and the work package consequently did not include extensive documentation; however, the inspector considered that the repair requirements and work performed were adequately discussed. Similarly, the safety analysis performed for temporary modification 95-034, "Instrument Air Jumper in Containment," noted that, "A loss of IA has been analyzed in the UFSAR section 9.3.1.1 and found not to affect safe operation of the plant," and concluded that a safety evaluation in accordance with 10 CFR 50.59 was not required.

At the close of the inspection period, the licensee was in the process of conducting a root cause analysis of the failed IA header. The operations department was also conducting a review of the event to determine if procedural modifications are required. Additional actions to determine if additional problems in the IA system piping exist, were under consideration.

In conclusion, the inspector assessed that operator response to the IA header leak was excellent. By rapidly locating and isolating the leak, operators averted a reactor trip due to low steam generator water level. Had a trip occurred at that point in the event, the resultant transient would have presented a significant challenge to the plant due to the pressurizer spray valves, PORVs, and steam dump valves being inoperable. Despite limited procedural guidance, operators continued to demonstrate excellent coordination and skill in reducing plant power with limited means to control RCS pressure and inventory. Maintenance support was promptly available and the IA leak was

rapidly repaired. PORC ensured that measures to provide for adequate plant control were in place prior to IA in containment again being isolated. Licensee review of the event was ongoing at the close of the inspection period.

2.0 MAINTENANCE (62703, 61726)

2.1 Preventive Maintenance Observations

The inspector observed portions of maintenance activities to verify that correct parts and tools were utilized, applicable industry code and technical specification (TS) requirements were satisfied, adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components, and to ensure that equipment operability was verified upon completion. The following maintenance activity was observed:

- Work Order 19502073, "Intermediate range nuclear instrument N-35 startup rate is indicating a constant positive value (0.15) at power"

2.2 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:

- CPI-TRIP-TEST-5.30, revision 13, effective date June 12, 1995, observed June 13, 1995

- PT-12.2, "Emergency Diesel Generator B," revision 82, effective date June 8, 1995, observed June 14, 1995

The inspector determined through observing this testing that operations and test personnel adhered to procedures, corrective action was promptly initiated if test results and equipment operating parameters did not meet acceptance criteria, and redundant equipment was available for emergency operation.

2.3 Technical Support Center Ventilation

On May 15, 1995, during routine surveillance testing of the B-emergency diesel generator (EDG), a fire alarm occurred in the technical support center. The fire brigade responded and determined that there was no fire. The cause of the alarm was that exhaust from the B-EDG had been directed toward the TSC ventilation system inlet by the outside wind. The event was documented in Ginna Station Event Report 95-072.

The licensee concluded that diesel exhaust had entered the TSC ventilation system due to the close proximity of the EDGs to the TSC, and the wind direction (280 degrees). To avoid false alarms in the future, the monthly surveillance test procedures for both EDGs were changed to include having the TSC fire alarm (fire system S-33) defeated prior to operating the EDG.

Additionally, an engineering work request was initiated to evaluate the possible detrimental effect of diesel exhaust on the charcoal filters that are used during TSC emergency ventilation. Licensee evaluation was in progress at the close of the inspection period.

3.0 ENGINEERING (71707, 37551)

3.1 Reactor Coolant System Temperature Detector Failure

On May 23, 1995, an RCS loop-A temperature instrument malfunction occurred. On the main control board (MCB), TI-402 (Loop A-2 Tavg) failed to greater than full scale high, and TI-406B (Loop 1A-2 Delta-T) failed to less than full scale low; accompanying MCB annunciators further confirmed that the problem was associated with the loop-A average temperature (Tavg) and differential temperature (Delta-T) instruments for reactor protection system (RPS) channel 2. Operators responded in accordance with ER-INST.1, "Reactor Protection Bistable Defeat After Instrumentation Loop Failure," to stabilize affected systems and defeat the failed instrument channel. No protective functions were initiated by the instrument failure and plant power was not affected. The event was documented in Ginna Station Event Report 95-074.

Troubleshooting revealed that the cause of the instrument failure was an open circuit in the resistance temperature detector (RTD), TE-402B. Since TE-402B is an immersion RTD (that is, the detector is in direct contact with RCS coolant), repair would not be possible until the plant was shut down, and the RCS cooled down and depressurized. However, the licensee determined that another RTD installed in loop-A, TW-450, could provide equivalent input to the affected RPS instrument channel.

TW-450 is physically identical to TE-402B, but is mounted in a thermowell (that is, the RTD is not in direct contact with the RCS coolant). It supplies instrumentation that provides temperature indication to the plant computer and controls a MCB annunciator that warns operators when approaching the low temperature overpressure protection setpoint. Although TW-450 serves a non-safety related function, it had been purchased, installed, and maintained to nuclear safety grade standards.

The licensee performed a safety evaluation (SEV-1049, dated June 5, 1995) in accordance with 10 CFR 50.59, and determined that temporary use of TW-450 in the Tavg reactor protection circuitry was acceptable. Modification specifics were developed in temporary modification TM-95-032 and implemented under work orders 19501997, "Probable failure of RCS "A" loop cold leg temperature indication," and 19501998, "Inspect and take readings on TW-450 to verify operability." Work and testing was completed and the loop-A Tavg and Delta-T instruments for RPS channel 2 were returned to service on June 5, 1995.

The inspector reviewed the licensee's safety evaluation for temporary use of TW-450 in place of TE-402B. The most significant operational difference between the two RTDs is the time to respond to temperature changes; the immersion RTD responds more quickly than the well-mounted RTD, since the temperature change must be transmitted through the thermowell before it can be sensed by the RTD. Response time constants were determined by the licensee to

be 0.5 second for the immersion RTD and 3.43 seconds for the well-mounted RTD, which was within the design basis for the RPS. The inspector considered that the licensee had adequately evaluated the effect of the increased lag time on RPS performance. Other considerations, such as the effects of eliminating the normal TW-450 functions, had also been appropriately addressed.

The inspector reviewed work orders 19501997 and 19501998 and noted no deficiencies. The inspector noted that replacement of TE-402B has been scheduled for the upcoming annual refueling outage under work order 19501691. The inspector concluded that the licensee's actions to temporarily connect TW-450 to substitute for TE-402B had been appropriate and were well developed. The inspector had no additional concerns on this matter.

4.0 PLANT SUPPORT (71750)

4.1 Radiological Controls 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, survey information was kept current, and postings and labeling were in compliance with regulatory requirements. Through observations of ongoing activities and discussions with plant personnel, the inspectors concluded that the licensee's radiological controls were effective.

4.2 Security Observations

During this inspection period, the 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. No unacceptable conditions were identified.

4.3 Fire Protection Observations

The inspectors periodically verified the adequacy of combustible material controls and storage in safety-related areas of the plant, monitored transient fire loads, verified the operability of fire detection and suppression systems, assessed the condition of fire barriers, verified the stationing of fire watches, and verified the adequacy of required compensatory measures. No discrepancies were noted.

5.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (71707)

5.1 Nuclear Safety Audit and Review Board Meeting

On May 17-18, 1995, the inspector attended a meeting of the Nuclear Safety Audit and Review Board (NSARB). Topics of discussion included: Operational issues, including plant performance, outage review, and Licensee Event Reports; assessments, including NRC inspection reports, Quality Assurance/ Quality Control (QA/QC) assessments, and the Systematic Assessment of Licensee Performance (SALP) report; status of the technical specification improvement

program; status of the steam generator replacement project; and a review of organizational initiatives. The inspector considered that the topics were well developed and knowledgeably discussed.

5.2 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 reports were reviewed:

- Monthly Operating Report for May 1995

No unacceptable conditions were identified.

5.3 Licensee Event Report

Licensee Event Reports (LERs) submitted to the NRC were reviewed to determine whether details were clearly reported, causes were properly identified, and corrective actions were appropriate. The inspectors also assessed whether potential safety consequences were properly evaluated, generic implications were indicated, events warranted additional follow-up, and applicable requirements of 10 CFR 50.73 were met.

The following LERs were reviewed (Note: date indicated is event date):

- 95-003, Unblocking of Safety Injection Actuation Signal While at Low Pressure Conditions, Due to Misleading Procedural Directions, Results in Inadvertent Automatic Safety Injection Actuation (April 7, 1995).

- 95-004, Steam Generator Tube Degradation Due to IGA/SCC, Causes Quality Assurance Manual Reportable Limits to be Reached (April 7, 1995).

The inspector concluded the LER met regulatory requirements and appropriately evaluated the safety significance. These LERs are closed.

6.0 ADMINISTRATIVE

6.1 SALP Management Meeting

On May 9, 1995, NRC Regional management met with RG&E officials, at the Ginna Training Center, to discuss the results of the Systematic Assessment of Licensee Performance (NRC SALP Report 50-244/93-99). At this meeting, the licensee identified initiatives that are planned to address program weaknesses identified in the SALP Report. Copies of the transparencies used by the NRC during the course of this meeting are provided as an Attachment.

6.2 Senior NRC Management Site Visits

During this inspection period, one senior NRC manager visited Ginna Station. On June 14, 1995, Mr. James C. Linville, Chief of Reactor Projects Branch 3, toured the site and met with senior licensee management.

6.3 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 the current resident inspection report 50-244/95-12 was held on June 19, 1995.

ATTACHMENT 1

Ginna SALP Management Meeting

Assessment Period

September 12, 1993 to

March 11, 1995

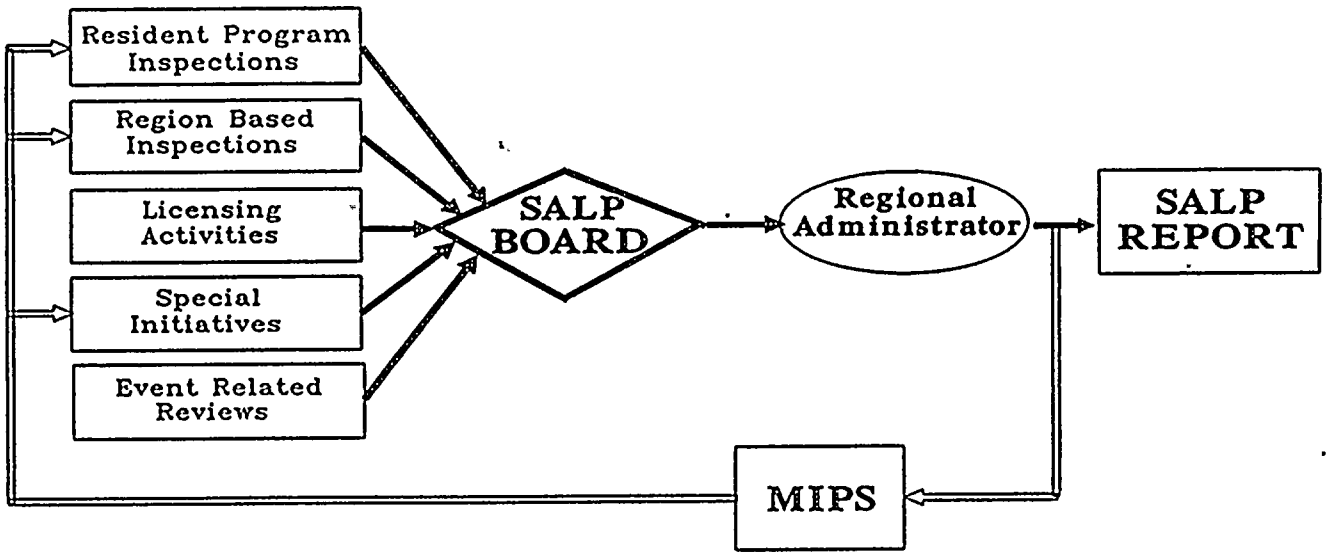
Board Meeting: March 23, 1995

Management Meeting: May 9, 1995

Agenda

• Introductory Remarks	W. Kane
• RG&E Comments	R. Smith
• Report Presentation	C. Hehl
• RG&E Response	R. Mecredy
• Closing Remarks	W. Kane
• RG&E Response	R. Mecredy

SALP Process



Performance Category Ratings

- Category 1: Superior Safety Performance
  - » Programs and procedures provide effective controls
  - » Self-assessment efforts are effective
  - » Corrective actions are comprehensive
  - » Recurring problems are eliminated
  - » Resolution of issues is timely
  - » Minimum inspections to verify safety
- Category 2: Good Safety Performance
  - » Corrective actions are usually effective, although some may be incomplete
  - » Licensee programs and procedures normally effective, however deficiencies may exist
  - » Root cause analyses are normally thorough
  - » Additional inspections necessary
  - » Self-assessments are normally good, although some issues may escape identification
- Category 3: Acceptable Safety Performance
  - » Insufficient control of activities in important areas
  - » Self-assessments ineffective in preventing problems
  - » Lack of understanding of safety implications of significant issues
  - » Corrective actions are not thorough
  - » Shallow root cause analyses
  - » Significant NRC and licensee attention required

Performance Analysis Areas for Operating Reactors

- Plant Operations
- Engineering
- Maintenance
- Plant Support
  - » Radiological Controls
  - » Chemistry
  - » Emergency Preparedness
  - » Security
  - » Fire Protection
  - » Housekeeping

SALP Category Ratings for the Previous Period Ending September 11, 1993

• Operations	2
• Engineering	1
• Maintenance	1
• Plant Support	2

SALP Category Ratings for Period Ending March 11, 1995

• Operations	1
• Engineering	1
• Maintenance	2
• Plant Support	2

Plant Operations

Category 1

- Continued strong operations performance
- Improved management involvement in operations
- Some past problems were corrected, such as licensed operator requalification program performance and configuration controls
- Questioning attitude of operators allowed early identification of plant problems

Maintenance
Category 2

- Well-maintained plant with continued long-term equipment upgrades in progress
- Very good maintenance procedures
- A number of significant maintenance related installation or re-assembly problems were noted, indicating weakness in preventing deficient conditions
- Root cause determinations for maintenance related problems varied in quality

Engineering
Category 1

- Effectively supported plant operations
- Strong management oversight of engineering activities
- Effective resolution of past technical problems
- Outstanding support of facility initiatives, such as steam generator replacement and conversion to standard technical specifications

Plant Support
Category 2

- Effectively supported safe plant operation
- Steady improvement in ALARA (exposure reduction program) performance, although some challenges remain
- Good emergency preparedness performance, with some attention to detail problems noted
- Security program performance was strong, with one significant vulnerability identified in the fitness-for-duty program
- Fire protection and housekeeping were effectively implemented

Overall Conclusion

- Very good safety performance
- Proactive self-assessment
- Improved management oversight and involvement
- Root cause and corrective action programs were thorough and comprehensive with some exceptions
- A broad review of maintenance related problems to prevent recurrence is warranted

NRC Planned Inspection Activities

- About 2000 hours of regular safety inspections will be completed over the next year
- An additional 60 hours of inspection of maintenance practices have been added to review performance in the Maintenance area
- An additional 40 hours of inspection of the emergency preparedness area have been added to review performance in the Plant Support area
- An inspection is planned to verify completion of the motor-operated valve design verification and testing program which is still in progress
- Approximately 600 hours of additional inspections have been added to review the steam generator replacement activities scheduled for completion in 1996