U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos. 50-272/88-80 and 50-311/88-80

Docket Nos. 50-272 and 50-311

License Nos. DPR-70 and DPR-75

Licensee: Public Service Electric and Gas Company
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Facility Name: Salem Units 1 and 2

Inspection At: Hancocks Bridge, New Jersey

Inspection Conducted: October 17-28, 1988

Inspectors: D. Caphton, Sr. Technical Reviewer, Team Leader
J. Carrasco, Reactor Engineer
P. Drysdale, Reactor Engineer
B. Hughes, Operations Engineer
H. Kaplan, Sr. Reactor Engineer
R. McBrearty, Reactor Engineer

Approved by: [signature] 1/17/89
Norman J. Blumberg, Chief, date
Operational Programs Section, Operations
Branch, Division of Reactor Safety

Inspection Summary: An announced outage team inspection on October 17-28, 1988 (50-272/88-80; 50-311/88-80)

Areas Inspected: Inspection of refueling outage activities which included design change modifications/installations; inservice inspection; and licensee action on previous inspection findings for Units 1 and 2. The inspection also included modifications to control panels at the training simulator in Salem, New Jersey.

Results: No violations were identified; however, two unresolved items were identified. Installation of modifications was determined to be adequate; however, a number of concerns were identified with the licensee's management controls relative to the design change and modification installation process. The inspector noted that in several cases NRC identified concerns were already known to the site management; however, an apparent lack of direct management action allowed these concerns to persist without a clear plan for resolution. In some cases where the inspector identified inadequate work, concerns were raised which reflect little or no documented evidence of an effective oversight directly coupled to the work product (See Attachment C for a listing of identified concerns).

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DETAILS

1.0 Persons Contacted

The names and positions of individuals contacted during this inspection are listed in Attachment A to this report.

2.0 General

2.1 Objective and Scope of Inspection

The objective of this inspection was to evaluate the licensee's performance in implementing the Unit 2 outage activities with particular emphasis placed on design change modifications and their installations. Licensee corrective actions taken on previous inspection findings were inspected with particular focus on structural items. The licensee's inservice inspection outage work, including the progress being made by the licensee's plant piping erosion/corrosion prevention and control program, was also inspected.

The licensee has implemented a new program for controlling design change modifications/installations for Salem. However, for this Unit 2 outage the majority of the modifications were performed under the old program. The inspectors selected three modifications being performed under the new program (this is annotated in the title for these modifications in the pertinent paragraphs of this report) and the remainder of the modifications inspected were under the old program.

Since some outage activities were still in progress at the conclusion of the inspection, it was not possible to confirm final closeout of each outage activity. The inspection did examine the licensee's controls to assure that each activity had progressed properly through the system and appropriate controls and procedures existed to ensure final closeout prior to plant restart.

2.2 Summary of Conclusions and Findings

The licensee is in a transition period of implementing a new design change modification process. Modifications were being performed under both the old and new process.

Overall, the installation work for modifications was found acceptable whether under the old or new process. However, certain management controls were found for both processes to be lacking in attention to details in a number of areas inspected by the team (refer to Attachment C for a listing of identified concerns). The team concluded that an increased level of management attention and involvement is needed to improve effectiveness of the design change/modification/installation process.

The licensee's approach to handling 10 CFR 50.59 reviews exhibited a lack of preciseness and attention to detail (Refer to Attachment C). Design analyses for potential consequences of system or component failures was also noted to exhibit weaknesses, for example, during the inspection of the design change involving the P-9 modification (2EC-2193, reference paragraph 3.9.c.), the analysis failed to examine potential consequences of system or component failures.

The licensee's QA audits are capable of identifying program problem areas to the plant management as noted in QA's audit of the Engineering and Plant Betterment (E&PB) group's design change/modification process. These program audits are, however, relatively infrequent (approximately on a two year cycle). The inspection team concluded that without aggressive management involvement to assure that corrective actions to audit findings (including reaudit of deficient areas) are properly pursued and resolved, QA's overall effectiveness in program and process improvement will be limited. The inspectors also noted during this inspection that quality assurance of the ongoing work exhibited lapses, i.e., where direct QA involvement was absent.

The licensee's inservice inspection program (ISI) was effective in meeting applicable ASME code and regulatory requirements. The licensee's plant piping erosion/corrosion prevention and control program is being implemented, however, it needs to be strengthened in several areas to assure its effectiveness.

During the inspection of modifications to control room panels, several questions arose regarding dual-licensed reactor operators shifting work stations between Units 1 & 2 control rooms without restriction. These concerns were addressed and resolved. (See paragraph 3.1.b.)

3.0 <u>Design Change/Modifications</u> (Modules 37700, 37701, 37702, 37828, 55050, 57050, and 72701)

a. Scope

The following is a list of the modifications inspected. At the time of the inspection, most of the work had been completed on the listed modifications.

Modification Description	Design Change Request (DCR) Identification No.
- Correct Human Engineering Discrepancies	2EC-2151
- Install ATWS Mitigation System Actuation Circuitry	2EC-2174
- Incore Instrumentation Mods	2EC-1915A
- Service Water Fan Coil Mods	2EC-2232
- Replace Service Water Butterfly Valves	2EC-2270
- *Replace Service Water Expansion Joints	2EC-2207
- *Diesel Cable Reroute	2SC-2011
- Reactor Control and Protection Mod, P-9	2EC-2193
- *Auxiliary Feed Water Pump 2 inch Bypass	2SC-2003

*Modifications worked under the new Engineering and Plant Betterment (E&PB) design change modification installation program.

b. Details of the Inspection Activities Performed

The inspection included specific observations concerning each of the modifications and included:

- Conducting system/equipment walkdowns in the field to confirm as-built information per installation drawings.
- Verifying that installed conditions conformed to modification specifications and drawings.
- Observing ongoing installation work, inspection and testing.
- Reviewing portions of the work that were already completed.
- Verifying that engineering work was technically sound.
- Verifying that the level and type of verification of quality was adequate for selected work.

- Determining proper classification of work according to standards, e.g., ASME requirements.

- Verifying that field changes were dispositioned properly.

- Verifying that personnel were being trained as appropriate.

In addition to checking the above items on each of the selected modifications, certain modifications were checked for the following:

- That installation and inspection procedures were adequate.

- That onsite and offsite review committees performed their review responsibilities concerning the modifications.

- That there was proper level of QA/QC involvement in inspection activities and problems.

Specific inspection findings and pertinent inspector observations concerning each of the selected modifications are discussed below.

3.1 Correct Human Engineering Discrepancies (HED) in the Salem Unit 2 Control Room (DCR 2EC-2151)

a. Scope

Design change request 2EC-2151 made numerous changes to the switch/control locations on the control room panels. These changes were generated during the Control Room Design Review performed in accordance with NUREG-0700.

The inspection reviewed the design input and review process, the completed field installation, workmanship, training, staffing documentation, housekeeping, fire barrier control, welder qualifications, control room access, and various work procedures.

The inspector interviewed craft supervision, craft fire barrier installers, the Station QA manager, off-site review engineers, control room operators, simulator instructors, emergency procedure co-ordinator, contractor engineers, and the plant operations engineer. In addition, the inspector visually inspected the Unit 2 control panels, the changes made inside the Unit 2 control panels, the mockup facility, and the revised control panels of the training simulator.

b. Findings

The design process utilized a full scale mockup and solicited licensed operator feedback regarding improvement changes being proposed in addition to a detailed control room design review. The simulator was modified prior to the control room and again operator feedback was utilized for this design change. The workmanship was adequate. Measuring and test equipment (M&TE) was properly controlled and documented. The fire barriers were adequately controlled and found to be reinstalled. Operator training was conducted and documented. New control room panel labels have been installed which enhance performance of the emergency operating procedures.

Two discrepancies were identified. The 50.59 review was not properly executed inaccordance with procedure GM8-EMP-009. The 50.59 Safety Evaluation Form (VPN-030) was not signed by the designated reviewer or by the department manager. Visual inspection beneath the Unit 2 control room console panel identified that the relocated recorder had a double nut installed, which was contrary to the analyzed design. The unauthorized double nut arrangement was corrected promptly when brought to the licensee's attention.

The inspector compared the Unit 2 control room changes to the existing unchanged Unit 1 control room. Due to the many observed differences between the units, resulting from the changes made to the Unit 2 control room, a concern developed regarding whether or not dual-licensed reactor operators should be restricted from rotating between the units. A meeting was held between Salem Operations, Training, and Engineering staff at the NRC Region I office to determine if dual-licenses should be modified under 10 CFR 55.61(b)(2). Additional control room inspections and interviews with control room personnel were conducted, and a course of action was prepared to define new requirements for dual-licensed operators. Additional NRC and licensee activity regarding this matter was conducted outside the scope of this inspection and results are detailed in NRC Combined Inspection Report Nos. 50-272/88-19 and 50-311/88-20. New staffing restrictions have been finalized in a letter: Labruna, PSE&G to US NRC, dated October 28, 1988.

c. Conclusion

This design change was extensive, involving approximately 10 volumes of documentation. Operator feedback was acted upon where possible. Training was adequate for both operators and craft personnel. Control room access was maintained in a controlled manner during the installation. The 50.59 signoffs were missed by several reviews, indicating that the reviews, including QA's, were not effective in this case.

3.2 ATWS Mitigation System Actuation Circuitry (AMSAC) (DCR 2EC-2174)

a. Scope

Design Change Request Package (DCR 2EC-2174, AMSAC) adds a process cabinet, signal isolators and cables, and modifies existing connections.

The inspector reviewed the completed installation, and visually inspected the new process cabinet and interconnections in the field. The inspector interviewed the team leader and jointly walked down the process cabinet wiring changes, fire barrier installations, and evaluated the quality of workmanship performed in the field. The inspector reviewed the 50.59 evaluation for adequacy and completeness, and checked document control and cable records.

b. Findings

The inspector reviewed controlled prints for the cable pull cards used for the installation. The pull cards matched the controlled drawings and Loop 529 was found correctly installed in the field. Workmanship was adequate. The accessible field run cable was visually inspected and found free of nicks, abrasions, cuts or any evidence of damage. The 50.59 review was properly executed. Housekeeping was adequate on the new installation but debris was found in the safety related protection cabinets. Further visual inspection of the nuclear instrumentation (NIS) cabinets revealed a cigarette butt which apparently had been extinguished on the fire stop inside the cabinet. It was then identified that the rear doors to the nuclear instrumentation cabinets are normally left open because of interference problems. Leaving the door open has potential to compromise fire protection.

c. Conclusions

Foreign material in the Reactor Protection and process cabinets has been accumulating over a period of time. QA reviewed the installed wiring changes and either did not notice this debris or accepted the condition. The cigarette butt in the NIS Cabinet demonstrates a lack of adequate control and supervision of personnel having access to this safety related equipment.

The Nuclear Instrumentation System rear doors are always open due to a cable run petruding from the instrumentation inside the cabinet. Station Management was apparently aware of the condition and had not taken action to correct the condition to permit closure of the rear panel doors.

The licensee took corrective action during the inspection to clean out all the process and protection cabinets. Four bags of debris and a flashlight were removed during the licensee's cleanup of the reactor protection and process cabinets.

Except as noted above, the AMSAC modification was found to be installed in accordance with the design package.

3.3 Incore Flux Monitoring (DCR 2EC-2232) and Core Exit Thermocouple (DCR 2EC-1915A) Systems Modifications

a. Scope

This modification work involved removing the 64 top-mounted core exit thermocouple assemblies and the 58 bottom-mounted incore flux monitoring thimbles. Both systems were then converted to an integral bottom-mounted flux thimble thermocouple (FTTC) system which includes associated incore detectors, external cabling, junction boxes, containment penetrations, signal processors, and control room instrumentation, etc. The modifications are a design upgrade to make the new system "Safety Related Equipment." The system now meets the Seismic Class I and Environmental Class 1E criteria, and also conforms to the requirements of Regulatory Guides 1.89 and 1.97, and NUREG-0737.

The inspection effort in this area involved a review of the DCR work packages to ascertain that these modifications are in conformance with the Technical Specification, 10 CFR 50.59 and other regulatory requirements; and that the licensee has implemented a QA program to control these plant modifications. At the time of this inspection, all installation work associated with these modifications had been accomplished. Functional and operational system testing could not proceed until plant startup, which would be subsequent to this inspection period.

Specific areas covered in the inspection of FTTC modifications are as follows:

- Review of detailed work instructions to ensure technical adequacy and proper implementation of administrative requirements.

- Review of fire safety practices associated with FTTC modification work.

- Review of control room Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs) affected by FTTC modifications.

- Review of 10 CFR 50.59 safety evaluations performed on new FTTC systems and equipment.

- Direct inspection of installed equipment to ensure conformance with procedure requirements and high quality work practices.

- Review of QA/QC program to verify that appropriate controls of modification work were executed in a satisfactory manner.

b. Findings

1. Procedures prepared for DCR Packages 2EC-1915A and 2EC-2232 for Unit 2 were virtually identical in content and structure to the corresponding packages prepared for the same modifications performed during the last outage of Unit 1. The administrative controls over the preparation of the Unit 2 procedures had not been officially superseded by new administrative controls of the Engineering and Plant Betterment Department. New administrative requirements were never the less imposed on the conduct of this modifications work, and complete and timely documentation of work accomplished was therefore cumbersome. Further review of detailed work instructions indicated that all procedures had received appropriate reviews and approvals prior to the start of modifications work. In the areas inspected, work instructions were determined to be technically adequate, and were maintained current. Procedure steps were observed to be verified by the installation contractor's supervision and by the PSE&G Project Supervisors. Deficient conditions encountered during work performance were adequately documented and the required engineering resolutions were obtained and approved prior to continuing with further work.

2. The inspector reviewed selected procedure sections which invoked fire protection requirements and also interviewed fire department supervisors to assess the fire safety activities and controls imposed on FTTC modifications work. Specific items inspected were fire seal impairment permits and fire watch coverage for new and modified cable penetrations in the Auxiliary Building. The inspector verified that the impairment of all fire seals and barriers had prior approval and that the necessary permits were issued by the station fire department. The inspector also verified that the necessary fire watches were provided during the impairment periods. A review of selected fire watch logs was conducted for six separate shifts during the time that the open 5 inch penetration in the control room

floor caused impairment of a fire barrier. The inspector confirmed that the fire watch required by technical specification for this barrier had been provided on an hourly basis for the necessary time period. No discrepancies were noted in this area.

3. Review of control room Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs) revealed that six EOPs and no AOPs were affected by FTTC modifications. These procedures were being revised during the inspection period to reflect changes in operators responses and instrumentation differences resulting from these modifications. The inspector reviewed the revisions with the cognizant operations staff engineer and determined that the revisions were appropriate and technically adequate. All revisions for Unit 2 EOPs reflect the corresponding changes made to EOPs on Unit 1 for the same instrumentation modifications. The inspector verified that all revised procedures subjected to this inspection were completed, approved, and in place in the control room prior to achieving Mode-4 plant conditions.

4. The inspector reviewed the principal engineering document, DE-AP.ZZ-008(Q) (supersedes GM8-EMP-028), which provides guidance for personnel conducting, reviewing, and approving 10 CFR 50.59 safety evaluations. This procedure has recently been implemented in 50.59 evaluations at the Salem Station. It provides a systematic and logical approach for performing these evaluations based on five different categories of design changes, and provides a significant improvement over the procedure it replaced. The procedure does not, however, provide a mechanism for dealing with 50.59 reviews which must be amended or revised by unforeseen field conditions that require a change in actual modification designs or installation details.

   It was observed that the 50.59 evaluation for the 5 inch core bore penetration in the Auxiliary Building (DCR 2EC-1915A) stated that no work would degrade the Seismic I integrity of the building because no rebar would be disturbed in the control room floor. In fact, three sections of rebar were cut during this operation, and an engineering analysis was performed to accept the altered condition. The analysis concluded that the condition did not affect the original 50.59 evaluation.

   However, the inspector concluded that cutting the rebar did affect the 50.59 evaluation. The inspector reviewed the engineering analysis with the cognizant civil engineer and found it to be technically sound. The altered installation condition was also discussed with the individual who prepared the original 50.59 evaluation and with the Station Licensing Engineer. Both agreed that the original 50.59 evaluation now presents an incorrect conclusion because that evaluation presumed that no rebar would be cut. They also agreed that the civil engineering analysis does not validate the 50.59 evaluation, even though it does support its conclusion. Although the 50.59 evaluation does not specifically prohibit cutting rebar, the resulting weakness in that evaluation would have been prevented by a more thorough review, acknowledging a highly probable condition e.g. cutting rebar, and identifying existing engineering controls and practices that deal with such conditions.

The inspector also noted that the 10 CFR 50.59 safety evaluation performed on the monorail installation in the seal table room (DCR 2EC-2232) did not account for the trolley assembly suspended from the monorail beam. The evaluation concluded that the integrity of the primary pressure boundary components and safety related equipment located at the seal table could be violated or degraded only through gross failure of the monorail. The inspector noted that although the trolley and monorail were adequately load tested after installation, the trolley is not restricted in any way from motion along the rail. Furthermore, the trolley is assembed from standard commercial catalog components of significant mass which reside approximately 20 feet directly above the seal table.

Based upon visual observation of the seal table area, and review of NRC Information Notice IN-84-55 and PSE&G's Safety Evaluation S-C-R200-MSE-0322, the inspector concluded that sensitive primary pressure boundary components and safety related equipment would be in direct jeopardy if the overhead trolley disassembled or failed and impinged upon the seal table. The inspector discussed this situation with the FTTC modifications Project Manager who agreed that a complete safety analysis should be performed on the entire monorail and trolley system. It was further agreed that plant maintenance procedures should include appropriate instructions to inspect and restrain or remove the trolley in Units 1 & 2 prior to plant operation. This is an unresolved item (50-272/88-80-01 and 50-311/88-80-01) pending completion of an adequate safety analysis and revision of applicable maintenance procedures to address the above concerns.

5. Direct inspection of installed FTTC equipment and components was performed to ensure that the work met the specified requirements in the design documents, and that the work had been performed in accordance with approved procedures and instructions. The installations reviewed appeared to have been performed with good quality workmanship and were in accordance with specified technical requirements. No deficiencies were noted in this area.

6. The inspector reviewed the FTTC modification DCRs and eight Station QA Surveillance Reports (SRs) to assess the extent and adequacy of QA involvement in the modifications work. It was noted that Station QA engineers had reviewed and concurred in these modification packages and had incorporated necessary notifications and hold points, however, the work instructions and design packages had not received any QA review for technical adequacy. Station QA engineers interviewed indicated that limited time and resources precluded technical reviews for these modifications. The Station QA Manager stated that technical reviews of maintenance and modification work packages are periodically performed by his organization. Selected QA SRs reviewed by the inspector revealed that adequate oversight functions were performed by Station personnel to assure that contractor work practices, procedure controls, QC methods, and personnel qualifications were proper, effective, and in accordance with PSE&G requirements. Except for the concern regarding lack of technical review of work instructions and design packages by QA, no discrepancies were noted in this area.

c. Conclusion

The FTTC modifications installed during the current outage have been accomplished using technically adequate design practices. Personnel accomplishing the modification work, and the engineering and QA services supporting the work were deemed to be adequate. The concerns raised by the inspector over the adequacy of 10 CFR 50.59 safety evaluations reflect a lack of attention to detail and thoroughness in the design and design review processes. No adverse affects regarding FTTC functional capability or plant startup were identified.

3.4 Auxiliary Feed Water Pump Bypass Line (DCR 2SX-2003)
Note: This DCR was performed under the new Engineering and Plant Betterment procedures.

a. Scope

This modification installs a 2 inch recirculation line across the No. 23 turbine driven auxiliary feed water pump. The new recirculation line will permit achieving 25% of rated flow and permit stable flow for conducting the technical specification required inservice test on the pump. The existing recirculation line only permits 100 gpm flow and the licensee's representative stated that the pump manufacturer recommends 245 gpm to achieve stable flow conditions. The modification required structural changes to provide for seismic grade hangers for the piping and included penetrations of the metal enclosure room surrounding the turbine driven auxiliary feed water pump. In addition, the DCR involves installing a clamp-on type flow measuring transducer having a digital display of flow in the vicinity of the auxiliary feed water pump. The new flow measuring instrument (trade name is Controlotron) will be used for inservice testing of the pump to determine operability under the technical specifications.

At the time of the inspection the installation was complete except for the hydro testing of newly installed piping which was scheduled to be accomplished during plant start up when steam is available to operate the turbine driven auxiliary feed water pump.

b. Findings

The inspector visually examined the newly installed piping, the welds, hangers, valves and the penetration through the auxiliary feed pump room metal enclosure wall. No deviations were noted regarding the actual installation versus the DCR design. The workmanship appeared adequate. QC hold points were utilized during pipe fit up and welding of the piping. The weld records were included in the DCR package.

The inspector noted an incorrect checkoff on the mechanical package DCR Exhibit 7, Internal Hazards Analysis Specialty Review Checklist, Procedure DE-AP.ZZ-0007(Q). Question 6 asked "...does the DCP involve deletion or modification of the structures?" This was checked No in the mechanical package. However, on the civil package exhibit 6, the question was checked, Yes. During the visual inspection of the piping and hangers, the inspector noted that revisions had been made to existing pipe and pipe hangers, in addition new hangers were attached via welding to existing structures and piping and hanger penetrations were made through the auxiliary feed water pump room metal enclosure wall and ceiling. The inspector reviewed this finding with the cognizant engineering group and it was acknowledged to be an incorrect mechanical package checkoff in view of the changes made. Detailed Technical Standards for engineers using the new E&PB procedures had not been published. It was stated that engineers and project managers were being provided training in the use of the new procedures.

The inspector noted that the DCR procedure check off list DE-AP.ZZ-0001(Q), Exhibit 3, item D. "Interface Review" was checked "No" in the mechanical package to questions 14, 16, and 17 which related to the operability interface on the front end of developing the design. The result of checking "No" took away the operations and maintenance department interface with the DCR on the front end of its development. Question 17, which was checked No, asked "Are there any human factors considerations?" A human factor consideration question was raised by the inspector and is discussed below.

Inspection of the "Controlotron" installation (ultrasonic flow measurement) noted that the electronic cabinet containing the flow indicator would be observable by an operator at the recirculation line throttle valve No. 146 by looking through the turbine driven auxiliary feed pump room's door opening. However, the electronic flow cabinet was mounted next to a hydrazine tank and hydrazine fumes were noticeably present at the electronic cabinet while the inspection was ongoing. The DCR did not consider the potential for a hydrazine environment for the operators or equipment. The inspector noted that the Controlotron manufacturer's installation manual indicated that an independent air source would be needed if the electronic cabinet would be subject to a corrosive environment. No independent source of air was provided by the design. Licensee representatives subsequently stated that it was planned to relocate the hydrazine tank to another area.

The DCR included completed separate 50.59 reviews and safety evaluations for mechanical, electrical and civil areas. The inspector noted that the mechanical 50.59 review did not discuss the consequences of a malfunction of a different type, for example, inadvertently leaving open valve 2AF-144, the recirculation line block valve. The cognizant design engineers stated that if the valve was left open adequate flow would still be provided by the pump due to its large capacity and adequate time would exist to permit manual closing of the valve. As previously stated, the inspector noted an apparent lack of formal guidance for engineers in completing the check list procedures that make up the DCR package.

The engineering manual system is described in OA-AP.ZZ-0002(Q), Revision 0, approved May 13, 1988. The overall system consists of five manuals: Engineering, Project Management, Technical Standards, Programmatic Standards and Design Basis Documentation. The licensee refers to the new E&PB Engineering Manual

System procedures as "new paper" and to the old system procedures as "old paper". Instructions for changing from "old paper" to "new paper" were covered by documented directives and letters. However, several instances were noted where it was not specifically documented as to e.g. which procedure was the preferred procedure where a new procedure had been issued before superseding the old procedure, indicating a lack of management preciseness during the transition period. The manual system is still under development, e.g., Technical Standards not yet developed. A lack of guidance for insuring consistency in performance of the new E&PB procedures is considered a weakness based upon the inspector's observations.

The station's valve lineups for operation are placed on a computer system TRIS (Tagging Request and Information System). The system's print out for control of the three newly installed auxiliary feed water system recirculation bypass line valves was inspected. The print out showed only two of the three valves were entered into TRIS at the time of the inspection. The 144 block valve was listed as locked closed and the 145 drain valve was listed as closed. The 146 throttle valve was not listed. It was noted that there was no valve position dual verification indicated for block valve No. 144 on the TRIS.

An Inservice Testing-Auxiliary Feed Pump procedure SP(0)4.05-D-AF(23), Revision 8, has been prepared for use in conducting the pump operability testing using the new recirculation line and clamp-on flow meter. This procedure requires that block valve 144 be locked closed upon completion of the procedure. It also requires independent verification of the 144 valve position. This procedure also requires the 146 valve to be locked in the throttled position (this was not shown as such on the TRIS).

c. Conclusions

The auxiliary feedwater pump recirculation line installation appeared to be installed in accordance with the DCR package. Guidance for the engineers completing the DCR package is less than adequate to achieve consistency during package preparation. A number of specific instances were noted where a lack of attention to detail, lack of preciseness and a general looseness in the implementation of the DCR work existed. The interface between design and operations appears to need improvement. Improvement is needed in the DCR package review process.

3.5 Service Water Fan Coil Modification (DPR 2EC-2270)

a. Scope

As the result of serious corrosion and erosion problems experienced in the Service Water System (SWS), the licensee initiated Modification No. DCR 2EC-2270 to replace the existing type 316 stainless steel and cement lined carbon steel piping with AL-6XN, a new relatively highly corrosion resistant stainless steel material. The modification covered piping systems associated with three of five fan cooling units (FCU) namely #21, #22, and #23. The remaining portions of the SWS will be replaced during subsequent outages. AL-6XN is an austenitic stainless steel consisting of 20% Cr-24% Ni-6% Mo with nitrogen addition. The filler material for the girth welds was alloy 625, a 60% Ni-20%Cr-9% Mo alloy. The system was being replaced in accordance with USAS B31.7, 1969 Edition and 1970 Addenda. Nondestructive examination (NDE) requirements included 100% visual and 100% liquid penetrant inspection. To provide control of welder performance and to monitor corrosion behavior of welded joints, the licensee voluntarily imposed a 10% radiographic inspection requirement on 3 inch and 10 inch welds.

b. Findings

The inspector reviewed the basis for the selection of the new materials as recommended by the licensee's consultants Stone & Webster and MPR Associates. The inspector determined that the selection of the new materials was based on a comprehensive corrosion prevention and control program including laboratory testing and turbine building loop tests covering various flow and temperature conditions. In house development of both automatic and manual welding procedures was performed in parallel with the corrosion testing with the aid of information obtained in visits to European manufacturers and installers. The licensee informed the inspector that installation of the SWS piping was being performed by Stone & Webster, hydrostatic testing and review of Code packages by Bechtel, and NDE by Magnaflux Quality Services (MQS-Wilmington, Delaware).

The inspector reviewed the manufacturing and fabrication history of the ALX-6N piping components and obtained the following information. The pipe material was purchased by Connex, the shop fabricator (formerly Dravo, Marrietta, Ohio) from Trent Tube, a division of Crucible Steel. In accordance with Stone & Webster Specification No. 001-P-301D Trent Tube, the pipe manufacturer, produced the piping by rolling and welding plate furnished by Alleghney Ludlum in accordance with the requirements of SB688 (plate) SB675 (pipe), SA312 and Code Case N438. The inspector reviewed random Trent Tube certified mill test reports (CMTRs) which showed acceptable mechanical properties and chemistry results. The CMTRs and attached furnace charts indicated that the welded pipe had been solution annealed at 2175°F and held a minimum of 15 minutes followed by water quenching. The reports also indicated that the material had successfully passed corrosion, liquid penetrant, x-ray, hydrostatic testing, metallurgical testing, and macro/micro examination. The latter included checks for inclusions, undesirable sigma phase, weld undercut, and heat affected zone cracks. The inspector verified by a review of licensee's surveillance Report VS 87-122 dated December 30, 1987, that Donovan Co., a subcontractor of Connex, had solution annealed pipe bends as required by Specification 001-P301D.

The inspector observed the automatic welding of a 3 inch schedule 40 pipe girth butt weld for spool piece C-S2-SWP-569. Welding was performed using the Diametric Gold Track II machine in accordance with automatic Tungsten Inert Gas (TIG) Welding Procedure NDWP-58. The root pass had been deposited using manual TIG procedure NDWP-46. The inspector visually noted the machine settings for the parameters employed during welding including amperage, voltage and wire speed. The heat input based on these parameters was calculated to be 19,320 joules/in, well below the licensee's self imposed value of 35,000 joules/in. The inspector verified that welding procedures and automatic machine operators (P62 and P69) qualifications conformed to ASME IX welding procedure and performance requirements. The inspector visually examined the deposited intermediate layers and found the welds to be free of discernible defects with good fusion along the side walls.

The inspector also reviewed other welding procedures used in the replacement program utilizing various combinations of automatic and manual welding processes, TIG and SMA (shielded metal arc), open butt and consumable insert for root passes, and found them to have been qualified in accordance with Section IX requirements.

The inspector reviewed two final document packages representing Test #2 and Test #14 field hydros in FCU-21 and FCU-22 systems respectively. The records showed that testing was performed successfully in accordance with specified engineering requirements of 300-315 psig for a minimum of 10 minutes. Weld History Records 4831 and 5078, representing welds C-S2-SWP-556-1 and C-S2-SWP-3291-1 were selected from these packages for review. The former weld (556-1) was welded with ASME IX qualified procedure NDWP-47, the latter (3291-1) with NDWP-58 and 46. The records showed that the final weld layers had been subjected to liquid penetrant and visual inspection. The former by MQS inspectors, and the latter by PSE&G inspectors as identified by their initials. Base metal and filler metal heat/lot numbers identified in these records were compared to appropriate CMTRs. The AL-6XN CMTRs (pipe or fittings) were identified as Trent Tube ht-821481, ht-711631, ht-LBVM, and WF1 ht-628 PNE1. The

Alloy 625 filler material CMTRs were identified as Techalloy VX-0160AY, Lehigh Test Lab ht-1X12 and ht-D4647, Huntington Alloy ht NX04E2AK and Acros YN5859. No deviations to SA or SFA material specifications were observed.

The inspector requested the licensee to provide the results of the self imposed 10% radiographic sampling program. The licensee reported a significant rejection rate for the 3 inch welds-27% (11 of 40 welds). Only three 10 inch welds were radiographed. Of these, two were rejected. The automatic process was primarily used for the 3 inch welds, whereas the manual process was used for the 10" welds. For the most part, the majority of the defects in the 3" welds appeared to be due to lack of fusion that occurred during automatic machine welding of the intermediate fill passes. Some minor root conditions (e.g., lack of penetration) were observed in the root passes. The inspector reviewed some of the rejectable radiographs and concurred with the licensee's interpretation. The licensee attributed the lack of fusion to the unauthorized use of higher than normal travel speed. The defects in the 10" welds were attributed to tungsten inclusions and lack of fusion. All of the rejected welds were successfully repaired or cut out and replaced with new welds. The licensee chose not to expand the 10% radiographic sampling plan for the following reasons: (1) radiography was not a Code requirement (2) excessive repair could lead to undesirable sigma formation in the heat affected zone and (3) the type of defects found would have minimal effect on the serviceability of the SWS because of the low operating temperature and pressure involved. In addition the licensee supported their decision not to expand the radiographic sampling plan, providing the inspector with a fracture mechanic analysis of 3 inch welds with an internal defect (2½" long x 1/32" deep) that represented a flaw twice the size observed in the radiographs. The analysis showed that internal defects of the size described would not initiate and grow into fatigue cracks of critical size, and also would not result in structural failure of the pipe because of a reduction in cross sectional area. The latter is supported by the fact that the joint is four times thicker than required. The additional thickness is intended for corrosion resistance. In addition, it is noted that all welds were liquid penetrant inspected. The inspector agreed with the licensee's conclusion regarding the major internal defects, but expressed concern about the potential effects of root defects, albeit minute, which could act as initiation sites for crevice corrosion. Because of this concern, the license decided to manufacture weld coupons with intentionally induced root defects to be placed in the presently operating SWS corrosion test loops for subsequent inspection and evaluation.

The inspector reviewed the licensee's QA surveillance program which was employed during the manufacture of the spool pieces at Connex. The inspector concluded that after reviewing numerous reports that the license had conducted an intensive surveillance program covering all phases of fabrication including bending, welding, CMTR review, heat treatment after bending, and NDE. All deviations and findings were reportedly resolved. It is noted that radiographs of girth welds were reviewed by the licensee during his surveillance activities at Dravo, whereas radiographs of the longitudinal seams as produced by Trent Tube were not reviewed by the licensee. The inspector requested that the licensee verify that these radiographs had been reviewed or if they were not reviewed, initiate review of same.

On October 25, 1988, the licensee reported that four weld history records showed evidence that signatures of MQS liquid penetrant inspectors had been falsified in four instances. It is noted the problem was discovered by Stone & Webster and reported to the licensee. The licensee investigated the incident and reported in Memorandum NQ5-88-0006, dated November 1, 1988, that four (4) of six hundred and seventy three (673) weld history records exhibited apparently falsified signatures. These welds have been reinspected. Also fifty four (54) weld records generated in DCR 2EC-2187 were reviewed by the licensee. No suspect records were found. The person responsible for the apparently falsified signatures has not been identified. The licensee's investigation in this matter is still in progress. The apparent falsification of weld records will be an Unresolved Item 50-311/88-80-02 pending the results of the licensee's investigation.

c. Conclusion

The work involving the SWS piping replacement was found to be in accordance with specified Code requirements and performed under a comprehensive QA program. The licensee's decision not to expand a self imposed radiographic sampling plan when significant defects were found was supported with adequate justification. In addition, the licensee plans to prepare mockups with similar defects for corrosion testing. The incident involving apparent falsified signatures was immediately reported by the licensee. An intensive licensee investigation ensued which preliminarily indicated that this was an isolated incident.

3.6 Replacement of SWS Expansion Joint (DCR 2EC-2207)

a. Scope

As the result of determining that seven existing rubber expansion joints in the SWS Intake Structure were not required, Modification 2EC-2207 was initiated to replace the joints with Belzona (ceramic epoxy) carbon steel spool pieces. The use of carbon steel spool pieces is intended to reduce future material and manpower costs.

b. Findings

The inspector verified that seven Belzona lined spool pieces, 2 feet long and made of SA 106 Gr B carbon steel (identified as 2-SW-P-133, 131, 135, 137 141, 139 and 143), were installed in the intake structure. The pieces had been shipped with one slip-on flange welded on and one-slip on flange shipped loose for field fit-up and welding. The inspector visually inspected a flange to pipe fillet weld and found no discernable defects. A review of weld history records showed that the welds had been welded with a combination of TIG and SMA processes in accordance with qualified Section IX welding procedures NPWP-13 and NPWP-2.

The record showed that the weld had been subjected to visual and magnetic particle inspection.

c. Conclusions

The work described in the subject modification was found to have been performed as specified. No deficiencies or violations were observed.

3.7 Replacement of SWS Butterfly Valves (DCR 2EC-2203)

a. Scope

As the result of deterioration of the rubber lining and attendant corrosion, seven existing carbon steel butterfly valves were replaced with new aluminum bronze valves in the intake structure (DCR 2EC-2203).

b. Findings

The inspector verified that valves identified as 24SW20, 23SW2C, 21SW17, 21SW20, and 22SW20 were installed in the intake structure. The inspector reviewed the CMTR's and verified that the properties conformed to the requirements of SA-148-Gr C 95400. The certification indicated that the valves were temper annealed at 1175°F for 7 1/2 hours.

c. Conclusion

The work described in the subject modification was found to have been performed as specified. No deficiencies or violations were observed.

3.8 Diesel Cable Reroute (DCR 2SC-2011)

Note: This DCR was performed under the new E&PB procedures

a. Scope

A licensee's design review identified a design deficiency relating to 10 CFR 50, Appendix R, Section III, G.3. The deficiency was that the emergency diesel generator cable 2CDC22-CT, which provides an alternate source of control and field flashing power to the three diesel generators during a postulated fire that requires alternate shutdown measures, was not physically independent of the ceiling area for the "zone under consideration." The licensee's corrective action initiated by DCR 2SC-2011 was to remove the cable and reroute the cable to comply with the Appendix R criteria.

A new seismically mounted conduit run was required to be installed by this modification. No electrical loads or circuitry changes were required. The new 2 inch conduit run was from an existing tray (2A089) in the auxiliary building where the cable was intercepted, through a newly drilled 4 inch diameter concrete wall penetration to the 480 volt switchgear room cabinet. The cable terminated in the same cabinet as did the original design.

This design change, including the installation work, was made under the licensee's revised Engineering and Plant Betterment procedures and program for implementing design changes.

b. Findings

The inspector noted that the work order for this modification had been signed off as complete at the time the inspection was initiated. On October 18, 1988, the inspector walked down the revised conduct run inside the auxiliary building and the 480 volt switch gear room. The 2-inch conduit run within the auxiliary building was visually observed to be wrapped with fire wrapping from the tray to the wall penetration as required by the DCR. The tray had also been restored to match the appearance of the undisturbed tray run. The new penetration through the concrete wall was observed to have been grouted around the conduit. The seismic supports for the conduct run were also inspected and noted to be installed per the DCR.

Inspection of the 2CDC22-CT cable inside the 2C125VDC bus panel in the 480 volt switch gear room noted that the excess cable resulting from the new shorter cable route was handled by making large loops in the front of the cabinet. Inspection of this accessible cable showed what appeared to be some minor nicks and abrasive damage to the insulation. The inspector asked to see the acceptance criteria used to assess the insulation damage and was told that none existed. An engineering group representative stated that damage to insulation was normally determined by meggering the cable. However, at this time the work had been completed and no meggering of this cable run had been done. There was no QC hold point in the modification "step by step" instruction to witness the megger of the cable. The QA manager stated that it was not intended to be meggered. The project manager stated that a megger test would be conducted in accordance with the procedures specified in the Installation Verification Procedure, Insulation Resistance, Continuity and Integrity Checks M13-IVP-501, Revision 0. The inspector noted that this procedure was part of the DCR installation package, however the installation instruction appeared to permit interpretation regarding the intent to megger the cable.

The 2CDC22-CT cable was satisfactorily meggered by the electrical contractor as witnessed by the inspector. The above stated procedure was used during the meggering. This megger test was witnessed by QA. The inspector requested to see the post calibration test of the megger instrument. The inspector witnessed the satisfactory post calibration test at the calibration lab.

A check at the tool room which issues measuring and test equipment (M&TE) found the control of M&TE equipment issued to contractors to be under adequate procedural control with one exception. The M&TE control procedure requires that before an M&TE item can be issued to a contractor, the contractor's name shall be on the approved list. The M&TE supervisor was found to have recorded on the M&TE issue log a megger EG-ZNM-0653 Serial No. G4539 issued (contrary to procedure) to a person not on the approved list to receive the device. The person was verified later by the inspector to have been subsequently added to the approved list.

The inspector reviewed the licensee's approved 50.59 Review and Safety Evaluation. The inspector had no questions regarding the 50.59 review.

c. Conclusion

The installation workmanship observed by this inspector appeared to be adequate. The DCR lacked specificity in the installation procedure to clearly and precisely specify that the cable run be meggered following installation. The DCR package provided no acceptance criteria for use by the craft or QA personnel to assess potential cable insulation damage although the potential existed for such damage in that the old cable had to be removed then rerun through new conduct. Some apparent minor surface damage to that portion of the cable insulation visually accessible was noted by the inspector. A satisfactory megger test was subsequently conducted.

3.9 P-9 Modification (DCR 2EC-2193)

a. Scope

Design change request DCR 2EC-2193, the P-9 modification, replaces the existing reactor trip on turbine interlock permissive C-8, turbine trip with permissive P-4, reactor trip.

The inspector visually inspected the NIS drawer installation, the soldered connections, the qualifications of craft persons performing the work, the relays installed in the reactor trip breakers, the connecting cabling, seismic installation of a cable pull box, and the torque wrench used in the installation.

The inspector interviewed the cognizant lead engineer, licensing engineer, system engineer, operations engineer and craft supervision. A lamp test in the Unit 2 control was visually inspected to verify indication.

At the training simulator, the inspector observed a Turbine Trip at 25% power without a reactor trip, which verified that the software was changed in the training simulator. The station Operations Review Committee Meeting Minutes were examined to ensure that this modification had been reviewed by them. The 50.59 review was inspected for adequate technical basis and completeness.

b. Findings

This modification was found to be installed without a properly executed 50.59 review in that Form VPN-030 was missing a department head approval signature. Torque wrench EG-ANM-0146 with a scale range of 25 to 250 ft-lbs was used at a 30 ft-lbs setting. This is in violation of station maintenance procedure M-23 "Torquing Guidelines", which states on page 7 of 41 "Do not use a Torque wrench to apply values that are below 20% or above 100% of the torque wrench scale." The inspector determined that the wrong size torque wrench was used to install four anchor bolts holding a seismically mounted pull box located above the reactor trip breakers. Post calibration of this torque wrench was performed at the inspectors request. The inspector witnessed the post calibration test and observed that this wrench failed. During the post calibration lab's inspection, it was found that the lead seal on the adjusting screw was missing. The licensee does have procedural provisions requiring rework upon identification of failed M&TE equipment, however, subsequent engineering evaluation determined that the applied torque was adequate and no further action is required.

The licensee's approach to 10 CFR 50.59 safety evaluations placed significance on identifying potential failure modes instead of examining the potential consequences of system or component failures. The review covered normal system operation but not allowable system operation, e.g., control rods in manual vs. failure of power mismatch circuit in the rod control system. When this was brought to the attention of the licensee by the inspector, changes were made to the operating procedures to limit operation with control rods in manual.

Visual inspection of the NIS field change kits verified adequate installation and workmanship. Visual observation of the wiring and relay in the Reactor Trip Breaker Cabinet indicated proper installation. M&TE control was not adequate in that when requested to locate a stopwatch used during the post test it took a day to find it, even though it was supposed to be in the issue room. Procedures are in place to checkout M&TE on the backshifts but in some cases the proper documentation is not filled out or the wrong entry log is used.

c. Conclusions

The licensee's review process including the QA review did not identify the missing 50.59 approval signatures. There was a QA holdpoint to verify torque on the anchor bolts but proper tool usage was not evaluated by the QA inspector.

The failure of the licensee to examine the potential consequences of system and component failures indicates an inadequate review process. It appears that the vendor's evaluation of the design change was accepted without a thorough review of supporting material or use of adequate independent review.

Based on the inspection with the exception cited above, the P-9 modification installation was found to be installed in accordance with the requirements established in the modification package.

3.10 Design Change/Modification Overall Conclusion

The design change/modification process was in the process of being upgraded by a new matrix type system. New procedures and management controls were in the process of being implemented at the time of this inspection. Most design change/modifications for the outage were being accomplished using the old process, however some were being accomplished using the new matrix process. This performance oriented inspection examined design changes/modifications performed under both the old and new processes. Overall, the installation work for modifications was found generally acceptable whether under the old or new process. However, management controls for both processes were found to lack preciseness and attention to details in a number of the areas inspected by the team (Refer to Attachment C for specific concerns). The team concluded that an increased level of management attention is needed to improve the effectiveness of the design change/modification process.

4.0 Inservice Inspection (Modules 73755, 73051, 57080 and 73753)

a. Scope of Inspection

The licensee performed inservice inspection during this outage to comply with requirements of the ASME Boiler and Pressure Vessel Code, Section XI, and with its inservice inspection schedule for the 1984 outage. The licensee additionally performed examinations in accordance with document S-2-VARX-MFD-0517, Revision 0, entitled "Ultrasonic Thickness Examination of Piping Systems with High Rate Probability of Erosion - Salem Generating Station, Unit No. 2".

The following areas were selected for inspection:

- Examination data related to RPV 60° azimuth meridianal weld No. 2-RPVCH-1446 C, Head to Flange weld No. 2-RPVCH-6446A, and weld No. 12-CF-1243-1A, 12 inch diameter chemical and volume control system weld.

- Control of ISI related nonconforming items.

- ISI vendor visual examination personnel qualification/certification records.

- ISI implementing NDE procedures.

- QA/QC involvement in ISI.

- Facility's plant piping erosion/corrosion prevention and control program.

The above areas were inspected with regard to compliance with applicable ASME Code and regulatory requirements and, in addition, NDE procedures were considered with respect to technical adequacy. Nonconforming ISI items were inspected with regard to proper closeout based on technical justification, disposition and the adequacy of the tracking system. The QA/QC involvement was examined by reviewing QA surveillance reports of ISI activities which were performed during the 1988 refueling outage.

b. Findings

Inservice inspection is mandated by the ASME B&PV Code, Section XI, and the Code edition applicable to a specific facility is identified by 10 CFR 50.55a(g) based upon the issuance date of its construction permit. The Salem Unit 2 facility is committed to the 1974 edition through the Summer 1975 Addenda.

The inspector determined that the examinations represented by the reviewed data met the applicable Code and regulatory requirements regarding test method, recording, evaluation, plotting and reporting of results. The inspector further determined that each data sheet was reviewed by the licensee. The initialled stamp appearing on each sheet indicated that the item represented by the data sheet was acceptable and met applicable ASME Code requirements. Licensee procedure No. M9-11P-01C, Revision 0 entitled "Review and Acceptance of NDE Data Result Records of ISI Long Term Plan Examinations" is in the PSE&G procedure review process and is intended to govern the review and acceptance of ISI data when it is adopted.

Visual examination methods are not included in SNT-TC-1A. ASME Section XI, IWA-2300 requires that personnel performing nondestructive examination methods not covered by SNT-TC-1A documents shall be qualified to a program that follows the guidelines of SNT-TC-1A. The program must be established by the plant owner or his agent. The ISI vendor visual examination personnel were qualified and certified in accordance with the vendor's program which was patterned after SNT-TC-1A. Visual examination personnel qualification/certification records confirmed that applicable requirements of ASME Section XI, 1974 Edition through Summer 1975 Addenda were met.

Nonconforming items were documented, tracked and closed out in conformance with the licensee's program. The inspector traced the items, which were documented by South West Research Institute (SWRI) Customer Notification Forms, through the system and verified that dispositions were technically adequate and that the closeout of each item was properly done.

The licensee has incorporated its ISI vendor's procedures into its program, renumbered them according to the PSE&G system, and has approved each procedure for use at Salem Unit 2.

ASME Section XI requires that NDE procedures must be approved and qualified. Liquid penetrant examination procedure M9-ISV-01S, Revision 1 (SWRI-MDT-200-1/68) was qualified in accordance with ASME code requirements. Deviation 4 was written to permit the use of non-flammable penetrant materials which were different than the materials originally specified by the procedure, therefore, in accordance with ASME Section V, the procedure was required to be requalified with the new materials. The qualification record of the new materials was not available and, at the inspector's request, the qualification examination was performed. The licensee's test block No. 08 was used and the qualification examination was performed by a Southwest Research Institute Level II examiner in the presence of the licensee's NDE supervisor and the inspector. By virtue of detecting the defects in the test block the procedure deviation 4 was considered acceptable and qualified to perform its intended function. Based on the above, the inspector had no further questions regarding this matter.

The inspector determined that the procedures were approved by the licensee, met applicable Code and regulatory requirements, and were technically adequate for their intended use.

Quality Assurance involvement in ISI activities was inspected by examining selected QA surveillance reports issued during the period from September 7, 1988 to October 12, 1988. The specific reports selected by the inspector were from the licensee's Surveillance Log which is maintained by the QA department. The reports covered a variety of activities including 10-year hydrostatic tests, personnel qualification/certification records, steam generator activities (tube plugging, J-nozzle replacement, cleanliness and tool control, helium leak testing, tube sheet marking verification), calibration of ultrasonic equipment, liquid penetrant examination, ultrasonic thickness measurements and functional testing of mechanical and hydrodraulic snubbers. Failed snubbers resulted in the testing of an expanded sample, leaks observed during hydrostatic tests were corrected and each report indicated that applicable procedures were followed, personnel performing the activities were properly qualified and that the final results were acceptable in each case.

Concern regarding erosion/corrosion in balance of plant piping systems has been heightened as a result of the December 9, 1986 feedwater line rupture that occurred at Surry Unit 2. This event was the subject of NRC Information Notice 86-106 issued on December 16, 1986 and its supplement issued on February 13, 1987 and NRC Bulletin 87-01.

The licensee's plant piping erosion/corrosion prevention and control program for this outage included approximately 160 scheduled examinations, all of which were completed at the time of this inspection. The document entitled, "Ultrasonic Thickness Examination of Piping Systems With High Rate Probability of Erosion" identifies the systems and areas which are included in the program and requires that each area be scheduled for examination for three consecutive refueling outages, including the current outage. Data collected from each examination will be used to calculate the rate of erosion/corrosion of each area and to determine the schedule for future examinations.

Information regarding nominal pipe wall thickness and design minimum thickness is included in the aforementioned document, and the senior staff engineer in charge of the project has prepared an engineering analysis logic diagram to control the evaluation and disposition of examination data. No areas requiring replacement have been identified at this time.

The program at Salem Unit 2 is new and the examinations performed during the 1988 refueling outage represent the first time the specific areas have been examined for erosion/corrosion. The inspector noted that the responsibilities for evaluating and dispositioning examination results and for the prioritization of future examinations are not well defined in the program.

c. Conclusions

The licensee's ISI program is effective in meeting the applicable ASME code and regulatory requirements. Quality assurance surveillances have included a wide enough variety of activities to provide assurance that the program is being properly implemented.

The licensee's plant piping erosion/corrosion prevention and control program is new. Responsibilities for evaluating and dispositioning examination results and for the prioritization of future examinations are not well defined in the program.

5.0 Quality Assurance Audits (Module 35701)

a. Scope

An inspection was made of audits relating to the Unit 2 outage and engineering/modification work for the outage.

b. Findings

The QA Audit Group's Audit Schedule for 1987-1988, Revision 3 dated September 8, 1988, listed one recent June through September 1988 audit, NM-88-35, for Engineering and Plant Betterment. Another earlier audit was performed for the Unit 1 outage during October and November 1987. No recent audit of the Unit 2 outage had been performed. The audit frequency was stated to be approximately two years. The audit NM-88-35 of Engineering and Plant Betterment (E&PB) focused on technical and programmatic areas and concluded that E&PB implementation of design controls had not been fully effective and listed specific program findings. Technical specialists were used to supplement the QA staff auditors. Corrective action was being initiated by E&PB to the audit findings.

Seven "Corrective Action Requests" (CAR) were listed in the NM-88-35 audit package. At the time of the inspection, one draft response (CAR No. SA-88-Q044-0) was provided to the inspectors. A letter from the Nuclear Engineering Standards Manager to the Manager-QA Programs and Audits, dated September 29, 1988, requested an extension in time for due dates on four CARs C017, C019, Q045 and Q043. These responses are due after the conclusion of this inspection.

Review of the licensee's NM-88-35 audit of E&PB conducted June 20 through September 2, 1988, identified that the audit focus although different from the focus of this inspection, identified some similiar type findings thus indicating to the inspectors the relative effectiveness of the licensees audit to provide to upper management indications of problems, e.g., "that E&PB implementation of design controls has not been fully effective."

c. Conclusions

The licensees audit NM-88-35 of E&PB appeared effective in identifying specific technical problems upon which to base a general conclusion that E&PB implementation of design controls had not been fully effective. The audit findings were addressed to the General Manager by letter dated September 27, 1988 and corrective action responses were pending at the time of the inspection. The audit findings had similarities to findings by this inspection e.g., instances where no acceptance criteria were specified, no evidence of QA involvement, lack of attention to detail, no QC hold point specified, implementing procedures/instructions unclear or lacking specificity and design review inadequacies. The licensee's audit results reinforce the conclusion that the E&PB implementation of management controls on a broad base has not been fully effective.

6.0 Containment Spray Valve Cracking - Unit 2 (Module 57050)

During this inspection the licensee reported that two 8 inch containment spray valves exhibited weepage and staining. The valves are utilized for test purposes. The test valves were temporarily bypassed with installation of a new line using flange connections. A preliminary metallurgical investigation indicated that the leakage was due to chloride stress corrosion cracking starting on the outside surfaces apparently from service water system leakage. The licensee was in the process of cleaning and liquid penetrant inspecting piping components which may have come in contact with the SWS leakage. The results of the cleanup, the liquid penetrant inspection and the metallurgical report of the failed valves have been requested by the inspector. This incident will remain an open item pending completion of licensee actions (50-311/88-01-03).

7.0 Status of Previously Identified Items (Module 92701)

7.1. (Closed) Violation (50-272/87-08-01 and 50-311/87-09-01) Wall Survey Conducted Without Written Procedure

The inspector verified and reviewed a written procedure number S-C-S000-SDM-0582-1 dated May 6, 1988 which established an annual Civil Engineering inspection program to verify the continuous structural integrity of masonry block walls in Salem safety related structures. The inspector found this procedure adequate and self contained.

Inspection of block walls and their drawings identified specific cases where improper labeling of the block walls on the drawings as well as the lack of physical labels on block walls existed. This concern was expressed to the licensee, who acknowledged the comment and agreed to implement further changes in order to improve the already established system of control of block walls.

Based upon the licensees existing procedural controls and commitments to further improve his controls, the violation 50-272/87-08-01 and 50-311/87-09-01 is closed.

7.2 (Closed) Violation (50-272/87-08-02 and 50-311/87-09-02) Wall Calculations Were Not Recorded Nor Controlled to Demonstrate the Structural Adequacy of the Modification

The inspector verified the existence of a well documented and self contained Computech Engineering Report for the assessment of the structural integrity/qualification of the masonry walls. In addition, the inspector verified the design modification (see reference on Attachment A) based on the Computech analysis. The inspector found the analysis/design modification to be adequate and properly controlled. Therefore, violation 50-272/87-08-02 and 50-311/87-09-02 is closed.

7.3 (Closed) Open Item (50-272/87-08-03 and 50-311/87-09-03) Provide Description of Analysis Techniques and Results for Cracked Block Wall

Upon examination of wall designated 2-4A (at elevation 100'-0", separating Units 1 and 2); the inspector determined the existence of a detailed evaluation that demonstrates the adequacy of the wall after eight supports were mounted on the Unit 2 (north) side of this wall.

A PSE&G document titled "Masonry Wall Evaluation," clearly shows the calculation for the most critical of the eight supports based on the loads. This support is labeled CTAT-11 23. The calculation shows that the overall structural integrity of the concrete block wall is maintained and the actual stress values are within allowable limits (this is based on block wall capacities). In addition, the inspector verified physically that the crack on wall 2-4A was properly repaired in accordance with a technically adequate procedure.

Therefore, item (50-272/87-08-03 and 50-311/87-09-03) is closed.

7.4 <u>(Closed) Unresolved Item (50-272 and -311/87-02-01) Lack of Evaluation of Pipe Supports for Seismic Stresses Induced by Self Weight Excitation</u>

The inspector reviewed selected calculations covering the inclusion of self weight excitation of pipe support frames prepared by CYGNA Energy Service, and verified that this inclusion does not affect the structural integrity of the support frames. This conclusion is based on the summary of stress interactions for critical members (calculation package P-2110, Revision 1, dated 6-11-87 page 10).

7.5 <u>(Open) Violation (50-272 and -311/87-02-02) Use of Uncontrolled Instructions in Performance of Piping and Pipe Support Design Activities</u>

The inspector determined that the licensee did not incorporate the U-bolt, strap load capacities and the requirements for evaluation of locally induced stress at U-bolt anchor support in the master pipe stress/pipe support specification. However, the licensee is taking steps to correct this issue and for this purpose the licensee prepared the drafts for Sargent and Lundy (contractor for the procedure consolidation task).

Therefore, this item, Violation (50-272 and -311/87-02-02) will remain open until the inspector verifies the final draft of the consolidated pipe stress and pipe support specifications.

7.6 <u>(Open) Violation (50-272 and -311/87-02-03) Lack of Documented Procedures and Instructions in Piping and Pipe Support Activities</u>

The inspector determined that the licensee did not include a quantitative acceptance criteria to perform the check of pipe support displacements and rotations under applied design loads to insure acceptability. Also, the licensee did not show any documented criteria for pipe supports.

However, the licensee has prepared a draft for Sargent and Lundy to consolidate all issues on self contained pipe stress and pipe support specifications; meanwhile, item 50-272 and -311/87-02-03 will remain open.

7.7 (Closed) Unresolved Item (50-272 and -311/87-02-04) Technical Concerns Related to the Use of Infinitely Rigid Supports in Piping Stress Analyses

This item consists of two parts which were resolved by the licensee in adequate and acceptable fashion as follows:

- The approach of considering support hangers, guides and anchors as infinitely rigid in the restrain directions triggered a safety concern of underestimation of seismic piping response.

  The inspector verified the technical justification for using rigid support models in piping design basis analysis, prepared by Sargent and Lundy engineers and concluded to be acceptable and technically adequate. Therefore, this issue is resolved on the conservatism of the design.

- The flexibility and the stiffness matrices for U-bolts, row 2, column 2 had zero value. This was a mistake which was corrected by Report No. S-C-MP00-VDC-0133-0 prepared by Franklin Research Center.

  Therefore, Unresolved Item (50-272 and -311/87-02-04) is closed.

7.8 (Closed) Unresolved Item (50-272 and -311/87-02-05) Failure to Implement Design Interface Requirements Between Mechanical and Civil/Structural Groups

The inspector verified the existence of stress directive No. 18, which is the identification and control of design activities between participating design disciplines. The inspector verified the implementation by reviewing a design change No. 2SC-2003 package 1 of 3 which in exhibit 2 and 3 delineate the interdiscipline interface record and design consideration check list respectively.

This verification is sufficient for the inspector to determine that there is adequate communication among disciplines involved in design activities.

Therefore, Unresolved Item (50-272 and -311/87-02-05) is closed.

7.9 (Closed) Unresolved Item (50-272 and -311/84-05-04) Justification is Lacking for Utilization of U-bolt for Axial and Torsional Restrain

The inspector verified the existence of an established base line for torque values for safety related U-bolts piping assemble, including specific diameters of 1¼ inches and 1½ inches, which were pointed out on a previous inspection.

This is shown on Field Directive No. S-C-VAR-NFD-0460, Rev. 4. The specific torque values calculated for 1¼ inches and 1½ inches diameters are shown on document P-12SWA-5 and 2C-CVCA-518 respectively.

The inspector also verified the existing on-going program to evaluate the locally induced stress on the pipe at U-bolt anchor locations. The licensee informed the inspector that the large bore analysis is completed. The inspector verified selected calculations to be adequate.

Nevertheless, the small bore piping remains to be completed. For this purpose, the licensee has committed resources and budget to complete the program in its entirety.

Therefore, this Unresolved Item is closed.

7.10 <u>(Closed) Open Item (50-272/85-08-01) Catalytic Welding Procedure M13A-7 for Gas Tungstan Arc Did Not Include Three Non-essential Variables Specified by ASME Section IX</u>

The inspector reviewed Public Service Welding Procedure NDWP-7 (similar to M13A-7), which is presently contained in the Public Service Welding and Brazing Manual, and verified that all non-essential variables are included in the subject procedure. The licensee stated that all procedures presently in the manual contain non-essential variables listed in Section IX. It is noted that at the time the finding was reported the licensee was in the process of upgrading the manual in anticipation of applying for National Board "R" and "NR" Certificates.

8.0 Unresolved Items

Unresolved items are matters about which more information is required to ascertain whether they are acceptable or violations. Unresolved Items are discussed in paragraphs 3.3.b.4. and 3.5.b.

9.0 Management Meetings

Licensee management was informed of the scope and purpose of the inspection at an entrance meeting conducted on October 17, 1988. The findings of the inspection were periodically discussed with licensee representative during the course of the inspection. An exit meeting was conducted on October 21, 1988 for team members concluding their inspection at that time and a final exit meeting was conducted on October 28, 1988, at the conclusion of the inspection. The findings of the inspection were presented at the exit meetings. See Attachment A for persons attending the exit meetings.

At no time during this inspection was written material concerning inspection findings provided to the licensee by the inspectors. The licensee did not indicate that any proprietary information was involved within the scope of this inspection.

ATTACHMENT A

1.0 Persons Contacted

Public Service Electric and Gas Company (PSE&G) and Contractors

	L. Adams, Senior Installation Engineer
c	R. Burricelli, General Manager, E&PB
	M. Bursztein, Principal Safety Review Offsite
	P. Benini, Principal QA Engineer
	H. Berrick, Principal Engineer
b	R. Best, Nuclear Training Supervisor
	D. Bhavnani, Senior Staff Engineer
b	P. Cartellano, SW Project Engineer, Stone & Webster
	B. Connor, Operations Staff Engineer
	C. Connor, ISI Supervisor
	R. Connors, Mechanical Systems Engineer
b	J. Cortez, Staff Engineer
	L. Doyle, Calibration Coordinator, Bogan, Inc.
abc	R. Donges, Senior Staff Engineer
	W. Denlinger, NDE Supervison, ISI
	J. Elwood, Insulator, Bechtel
b	J. Gorga, Stress Supervisor
c	H. Gross, Team Leader, UE&C
	M. Gross, Quality Assurance Engineer
b	J. Hawks, Project Manager
b	J. Jackson, Tech Manager, Salem OPS
	A Kao, Civil/Structural Supervisor
	G. Kapp, Project Manager
	J. Kerin, Senior Fire Protection Supervisor
	P. Kwok, Senior Staff Engineer
	J. Lark, Station QA Engineer
b	M. Leach, Technical Staff Engineer
c	S. Lehman, General Physics Craft Supervisor
b	L. Leitz, Project Manager
	J. Lloyd, Principal Nuclear Training
b	D. Dongo, Stress Supervisor
	T. McIvaine, Fire Protection Supervisor
abc	L. Miller, General Manager Salem Operations
	M. Morroni, Technical Engineer
	V. Morton, NDE Level III, Southwest Research Institute
	J. Musumeci, Salem Operations Engineer
	D. Namit, Senior Staff Engineer
	P. O'Donnell, Principal Engineer
bc	A. Orticelle, Outage Manager
	P. Ott, Technical Engineer

a Denotes attendance at the entrance meeting on October 17, 1988
b Denotes attendance at the exit meeting on October 21, 1988
c Denotes attendance at the exit meeting on October 28, 1988

Persons Contacted (continued)

Public Service Electric and Gas Company (PSE&G)

abc D. Perkins, Salem QA Manager
bc M. Raps, Standards and Assurance Supervisor
R. Raymond, Lead Civil Engineer
F. Ricart, Offsite Safety Review Engineer
D. Rice, Installation Engineer, M&M Contracts
A. Robinson, Nuclear Technician
abc G. Roggio, PM SW Project
b J. Rowey, Project Engineer
F. Saraceni, Electrical Systems Engineer
T. Shome, Civil/Structural Lead Designer
W. Schultz, Manager QA & Audits
W. Straubmuller, Project Manager
R. Swartzwelder, Senior Licensing Engineer
D. Tauber, Quality Control Supervisor
bc F. Thompson, Supervisor Nuclear Licensing
D. Thompson, Field Superintendent, Combustion Engineering
W. Tomanek, Senior Design Engineer, General Physics
b H. Trenka, Project Manager
L. Trow, Principal Engineer, Atometrics Co.
J. Vorderbueggen, PE Project Director, General Physics
M. Wita, Station QA Engineer
T. Worrell, Station QA Engineer

United States Nuclear Regulatory Commission (U.S. NRC)

abc R. Borchardt, Senior Resident Inspector, Salem
ab K. Gibson, Resident Inspector, Salem
b P. Swetland, Chief, Reactor Projects Section No. 2B

The inspectors also contacted other administrative, operational, technical and contractor personnel during the inspection.

a Denotes attendance at the entrance meeting on October 17, 1988
b Denotes attendance at the exit meeting on October 21, 1988
c Denotes attendance at the exit meeting on October 28, 1988

ATTACHMENT B

Reference Documents

1.0 Organization/Administrative Procedures

Procedure Number	Revision	Title
OA-AP.ZZ-0001(Q)	0	E&PB Organization
OA-AP.ZZ-0002(Q)	0	Engineering Manual System
NA-AP.ZZ-0008(Q)	0	Administrative Control of Design and Configuration Change
NA-AP.ZZ-0001(Q)	0	Preparation and Use of Procedures
DE-AP.ZZ-0001(Q)	0	Design Bases/Input
DE-AP.ZZ-0003(Q)	0	Modification Walkdown Program
DE-AP.ZZ-0007(Q)	0	Speciality Review
DE-AP.ZZ-0008(Q)	0	10 CFR 50.59 Reviews and Safety
DE-AP.XX-0009(Q)	0	Peer Review
DE-AP.ZZ-0010(Q)	0	Design Verification
DE-AP.ZZ-0048(Q)	0	Control of Calibrated Measuring and Test Equipment
GM8-MSP-001	3	E&PB Manual
GM8-MSP-003	0	Indoctrination and Training
GM8-EMP-004	1	Design Drawing Control
GM8-EMP-005	2	Design Calculations
GM8-EMP-009	2	Operational Design Change Control
OA-PJ.ZZ-0011(Z)	0	Matrix Organization-A Project Overview

2.0 Engineering and Work Control Procedures

Procedure Number	Revision	Title
GM8-EMP-007	0	Document Identification
GM8-EMP-008	1	NRC Bulletin, Information Notices and INPO SOERs
GM8-EMP-010	2	Safety Evaluations & Field Directives
DE-AP.ZZ-0017(Q)	0	Modification Concerns and Resolutions
DE-AP.ZZ-0018(Q)	1	Engineering Deficiency Control

Procedure Number	Revision	Title
DE-CS.ZZ-0013(Q)	0	Contractor Use of M&TE
DE-CS.ZZ-0014(Z)	0	E&PB Contractor Electrical Installation Verification Procedures
M13-IVP-501	0	Installation Verification Procedure, Insulation Resistance, Continuity and Integrity Checks
M3K	3	Electrical Cable Installation/Pulling
S-C-E000-EFD-0438	0	Technical Requirements for Construction of Electrical Installations
S-C-EC00-EFD-0384	0	Acceptance Criteria for Crimp and Formed Wire Hook Terminations
Specification 401-P301D		Stone and Webster Specification for Shop Fabricated Piping

3.0 Structural References

Document Number	Report/Revision	Title
S-C-S000-SDM-0582-1	5-6-88	Design Memorandum S-C-S000-SDN Engineering Department Annual Inspection of IE Bulletin 80-11 Masonry Walls
Computech Engineering Report No S000-VDC-O-0197	1-30-88	Control Facility Building/Walkway and Truckbay - Assessment of Structural Integrity/Qualification of Masonry Wall
N/A	11-28-80	PSE&G Report on Re-evaluation of Masonry Walls for Salem Generating Station Unit 1

Document Number	Report/Revision	Title
N/A	N/A	PSE&G Repair Procedure for Cracked Masonry Wall on Reference Line No. 14
N/A	12-7-87	PSE&G Masonry Wall Evaluation (Wall 2-4) in Reference to IE Bulletin 80-11
N/A	N/A	CYGNA Energy Services - Calculation Package P-2110 Multiple Support Self Weight Excitation
N/A	5-18-87	CYGNA Energy Services - Calculation for Pipe Stiffness
PSE&G Stress Directive No. 18	3-20-87	Pipe Support Evaluation "Identification and Control of Design Activities between participating Design Disciplines Salem No. 1 and 2 Units"
PSE&G Report No. S-C-MPOO-VDC-0133-0	N/A	Analysis and Testing of U-bolt Anchor Assemblies
PSE&G Stress Directive No. 17	Rev. 1	Criteria for Evaluation of Directive No. 17 Locally Induced Stress in U-bolt Anchors and Welded Attachments
PSE&G Design Modification DCR 2SC-2003	6-2-88	Installation of 2" Diameter Recirculation Line #23 Auxiliary Feedwater Pump
PSE&G Field Directive S-C-VAR-NFD-0460	8-24-88 Rev. 4	Torque Verification Program for Safety Related Piping U-bolt Anchor Assemblies

Document Number	Report/Revision	Title
Sargent and Lundy Engineers Report EMD-064314	12-87	Technical Justification for Using Infinitely Rigid Support Models in Piping Design Basis Analysis
Franklin Research Center Report F-6070-001	3-14-85	Analysis and Testing of U-bolt Anchor Assemblies

4.0 Non-Destructive Examination Procedures/References

Document Number	Revision	Title
M9-ISV-01S	Rev. 1	Solvent-Removable Liquid Penetrant Color Contrast Examination (SWRI-NDT-200-1/68, Rev. 4)
M9-ISV-02S	Rev. 0	Visible Water-Washable Liquid Penetrant Examinations (SWRI-200-3/7)
M9-ISV-03S	Rev. 0	Dry Powder Magnetic Particle Examination (SWRI-NDT-300-1/26, Rev. 4)
M9-ISV-05S	Rev. 0	Manual Ultrasonic Examination of Pressure Piping Welds (SWRI-NDT-600-3/62, Rev. 5)
M9-ISV-15S	Rev. 0	Visual Examination of Nuclear Power Plant Components by Direct or Remote Viewing (SWRI-NDT-900-1/51, Rev. 1)
M9-11P-01C	Rev. 0	Review and Acceptance of NDE Data Result Records of ISI Long Term Plan Examinations
AP-9	Rev. 14	Work Control Program

Document Number	Revision	Title
S-2-VARX-MFD-0517	Rev. 0	Ultrasonic Thickness Examination of Piping Systems with High Rate Probability of Erosion - Salem Generating Station, Unit No. 2

5.0 Drawings

Document Number	Revision	Title
201093A-8706	3-11-87 Rev. 27	Salem Nuclear Generating Station No. 1 & No. 2 Units Auxiliary Building, Section X-X, Sheet 2
207076A-8798	10-17-86 Rev. 17	Salem Nuclear Generating Station No. 1 Unit Auxiliary Building Floor Plan Elevation 64'-0" Architectural
245702A-1682	4-30-84 Rev. 2	Salem Nuclear Generating Station Controlled Facilities Building Walkway and Truck-Bay Roof Plan Wall Sections & Details
201061A-8705	10-25-83 Rev. 20	Salem Nuclear Generating Station No. 1 & 2 Unit Auxiliary Building Section F-F
207082A-8798	1-28-87 Rev. 18	Salem Nuclear Generating Station No. 1 Unit Auxiliary Building Floor Plan Elevation 122'-0"

6.0 QA Surveillance Reports (SR)

SR 88-0639; SR for Installation of Reference Junction Boxes for the 78' Elevation Penetration.

SR 88-0647; SR for Combustion Engineering Welder Certifications.

SR 88-0649; SR for Review of Data Sheets and Test Equipment Logs to Verify Combustion Engineering, M&TE Program Compliance with PSE&G M&TE Program.

SR 88-0665; SR for 5" Core Bore in Control Room Equipment Room Floor.

SR 88-0688; SR for Review of DCR 2EC-1915A, DCP No. 2.

SR 88-0733; SR for Review of DCR 2EC-1915A, DCP No. 2.

SR 88-0852; SR for Removal of Flux Thimbles Nos. 23, 31, 42, and 49.

SR 88-1093; SR for Assembly and Installation of FTTC Hoist Frame in Seal Table Room.

7.0 Work Orders

WO 880511051; Erect Flux Thimble Frame and Hoist in the Seal Table Room to Support the FTTC Installation.

WO 881002055; Repair Penetration Seal #F-15612-112.

8.0 Other Reference Documents

Fire Protection Permit #88-654; Permit for Penetration Seal #F-15612-112 Impairment.

MCR-2EC-1915-5; Modification Concern/Resolution for Cut Rebar in 5" Core Bore.

ANSI B30.11-1980; Monorails and Underhang Cranes.

NRC Information Notice IE-84-55; Seal Table Leaks at PWRs.

S-C-R300-CDM-486-0; Design Memorandum on Bottom Entry In-Core Instrumentation System, Core Exit Thermocouple Upgrade for NUREG-0737.

S-C-R300-CDM-0490-0; Design Memorandum on Core Exit Thermocouple Backup Display - Upgrade of NUREG-0737.

S-C-R200-MSE-274; Design Memorandum on Flux Thimble Ejection and Seal Table Leak; Review of Westinghouse and NRC Documents.

Civil Engineering Directive No. 1, Rev. 0; Instructions for Drilling Holes and Core Bores in Concrete.

S-C-R200-MSE-0322; Safety Evaluation of the Flux Mapping System; Potential for Interaction of the System with the Seal Table Due to Seismic Loads.

ATTACHMENT C

Specific Concerns

The inspection team used the following evaluation criteria for assessing management activities relative to the inspection findings and concerns:

Involvement: Active management participation to ensure that engineering design, analysis, and work packages are adequately prepared, reviewed, and approved; including active participation in review of results of ongoing work.

Control: Active management participation during the execution phases of work to ensure that administrative controls exist and are fully implemented both in work performance and in deficiency resolution.

Attention to Detail: Sufficient oversight to ensure that adequate detail is considered to properly prepare engineering and work documents and to provide for adequate and timely resolution of deficient conditions.

The specific concerns identified during the inspection are tabulated below:

Inspection Report Paragraph	Concern	Involvement	Control	Attention to Detail
3.1.b	50.59 Review not properly executed.	X	X	X
	Double nut installed, contrary to seismic design specified.		X	X
	Adequacy of restrictions for dual-unit operators		X	X
3.2.b	Debris found in safety related cabinets (including cigarette butt)		X	X
	Rear doors are permanently open to nuclear instrumentation cabinets	X	X	X
3.3.b.4	50.59 review presents incorrect conclusion after rebar was cut	X	X	X
	50.59 review failed to consider trolley assembly	X		X

Inspection Report Paragraph	Concern	Involvement	Control	Attention to Detail
3.3.b.6	Work instructions and design packages received no QA review for technical adequacy	X		X
3.4.b	Incorrect checkoff on Design Change Request, Exhibit 7, question 6, of Procedure DE-AP.ZZ-0007(Q)	X		X
3.4.b	DE-AP.ZZ-0001(Q), Exhibit 3D, operability questions 14, 16, and 17 as checked removed the operations interface with the modification design on the front end	X		X
	Lack of detailed guidance for engineers doing design work	X	X	X
	"Controlotron" (ultrasonic flow measuring) electronic cabinet installed in a potential hydrazine environment. Operating personnel may be exposed to the hazardous environment	X	X	
	50.59 review did not consider the consequences of a malfunction of a different type	X	X	X
	Valve No. 146 not listed in the Tagging Request and Information System	X	X	X
3.5.b	Sampling plan was not expanded for weld defects and root defects may have potential for initiating crevice corrosion	X		X

Inspection Report Paragraph	Concern	Involvement	Control	Attention to Detail
3.8.b	No acceptance criteria for craft or QC personnel for assessing damage to emergency diesel generator cable insulation. No QC hold point to witness meggering of the cable	X	X	X
	One Measuring & Test Equipment controlled megger was issued to unauthorized person contrary to procedure	X	X	
3.9.b.	DCR 2EC-2193 was accomplished without a properly executed 50.59 review	X	X	X
	Torque wrench of incorrect size was used contrary to procedure. Torque wrench failed post use calibration test and lead seal was missing	X	X	X
	Measuring & Test Equipment controlled stop watch was found to be missing for a day		X	
3.9.c	50.59 review failed to examine potential consequences of the allowable system operation, indicating inadequacies in the review process	X		X
4.0.c	Plant piping erosion/corrosion prevention and control program needs improved definition	X	X	X
5.0.c	Engineering and Plant Betterment implementation of management controls has not been fully effective	X	X	X