U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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Licensee:	GPU Nuclear Incorporated 1 Upper Pond Road Parsippany, New Jersey 07054
Facility Name:	Oyster Creek Nuclear Generating Station
Location:	Forked River, New Jersey
Inspection Period:	September 14, 1998 - October 25, 1998
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EXECUTIVE SUMMARY

Oyster Creek Nuclear Generating Station
Report No. 98-09

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

Plant Operations

- Operators properly performed a normal reactor shutdown and transition to shutdown cooling to commence refueling outage 17R. (Section O1.2)

- Operators conducted refueling activities safely while handling several equipment challenges, water clarity problems, and delays to the critical path schedule. (Section O1.3)

- Operations responded appropriately to a loss of shutdown cooling. Operations management did not clearly communicate and coordinate a subsequent operating procedure change implemented to improve shutdown cooling system availability while shutdown. (Section O2.1)

- Senior reactor operators demonstrated a good technical specification awareness in effectively tracking and implementing limiting conditions for operation action statement requirements during the refueling shutdown. (Section O4.1)

- Operations demonstrated good attention to detail and a standard for excellence in effectively controlling the removal, tagging, and restoring of numerous systems, involving thousands of tags, in an error-free manner. (Section M2.1)

Maintenance

- Logistical support (planning and scheduling) provided a thoroughly planned, risk-informed, and properly scoped refueling outage plan. (Section M2.1)

- Maintenance provided good support to operations in the conduct of refueling equipment repairs, however, the material condition of the refueling equipment presented several challenges during refueling. (Section M2.1)

- Plant personnel demonstrated good foreign material exclusion control. (Section M2.1)

- Senior management demonstrated safety-conscious decision making, a commitment to worker safety, an active involvement and plant presence, and a concerted effort to remove schedule pressure. (Section M2.1)

- Maintenance personnel demonstrated good work practices, conducted activities in accordance with approved procedures, demonstrated a good questioning attitude, properly documented work, and appropriately engaged the corrective action process. (Section M2.1)

- The Safety Department provided high-quality oversight of outage activities and contributed to a safe work environment. (Section M2.1)

- Nuclear Safety Assessment developed and implemented a safety-focused and comprehensive outage inspection plan. (Section M2.1)

- Nuclear Safety Assessment and Quality Verification provided around-the-clock outage coverage, made important contributions to ensure nuclear safety and quality was maintained, and appropriately documented concerns via the corrective action process. (Section M2.1)

- Maintenance demonstrated good work practices and proper radiological controls, and maintained the work package documentation accurate and up to date during the emergency core cooling system suction strainer replacement. (Section M2.2)

- Quality Verification conducted near-continuous oversight of the emergency core cooling system suction strainer replacement and, in general, ensured excellent foreign material exclusion control in the torus. (Section M2.2)

- Maintenance personnel caused an unplanned actuation of the core spray system 1 due to a procedure weakness. Maintenance supervision did not demonstrate a good questioning attitude while implementing the procedure. Maintenance management took prompt and appropriate corrective actions. (Section M3.1)

Engineering

- Engineering did not thoroughly evaluate an April 1998 shutdown cooling system functional failure to improve shutdown cooling system availability while shutdown. (Section O2.1)

- Engineering provided a high-quality and well developed design change package for the emergency core cooling system suction strainer replacement. (Section M2.2)

- Engineering demonstrated good awareness and performed an appropriate evaluation of oxidation and marking on the outside of several fuel channels. (Section E2.1)

- Engineering demonstrated good initiative in conducting small bore piping walkdowns and evaluations. Engineering appropriately documented and initiated corrective actions for identified deficiencies. (Section E2.2)

- Engineering led plant staff's efforts to hydrostatically test and inspect the isolation condenser tube bundles. Engineering appropriately documented identified deficiencies, promptly addressed reportability, and initiated actions to replace leaking tube bundles. (Section E2.3)

- Engineering demonstrated a good questioning attitude in the conduct of a thorough root cause evaluation for leaking electromatic relief valve pilot valves. Oyster Creek engineering identified an industry concern regarding pilot valve manufacturing tolerances and initiated appropriate actions to replace installed pilot valves and to inform the nuclear industry of the discrepancy. (Section E4.1)

Plant Support

- Radiation Protection technicians effectively controlled radiological conditions and access during the emergency core cooling system suction strainer replacement. (Section M2.2)

- Radiation Protection established an acceptable radiation protection program to support refueling outage 17R as evidenced by appropriate briefings, postings, and ALARA controls in the reactor, drywell, and turbine buildings. (Section R1.2)

- The outage work force demonstrated a weakness in the areas of foreign material exclusion and radiological housekeeping. A number of small loose articles were found near the downcomers in the drywell, while poor housekeeping practices in high dose rate areas could lead to additional occupational exposures during outage close-out. (Section R1.2)

- Radiation Protection management and training staff took appropriate actions to assess the integrity of the Oyster Creek radiation control technician training program and to alert other utilities of a potential Northeast Utilities examination bank security compromise. (Section R5.1)

- Security maintenance technicians promptly addressed several minor equipment deficiencies and contributed to good material condition and high reliability of the security equipment. (Section S1.1)

- A security guard did not properly conduct vehicle escort duties as he left a running vehicle unattended in the protected area. Security force members did not properly document the occurrence. Once notified by the inspector, security management took prompt and appropriate corrective actions. (Section S4.1)

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Report Details

Summary of Plant Status

The period began with the unit at 97% power in coastdown to refueling outage 17R. On September 15, operators removed the intermediate pressure feedwater heaters from service and then increased power to 100%. Coastdown to refueling continued. On September 24, operators reduced power to 70% for planned maintenance on the 'A' feedwater pump and 'A' condensate pump. At 9:04 p.m. on September 25, operators commenced a reactor shutdown in order to refuel. At 2:32 a.m. on September 26, operators conducted a manual scram from 10% power in accordance with their reactor shutdown procedure. At 11:11 a.m. on September 26, operators placed the unit in cold shutdown (<212° F). The unit remained in the cold shutdown condition for the remainder of the period.

I. OPERATIONS

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors conducted frequent reviews of ongoing plant activities and operations using the guidance in NRC inspection procedure 71707. The inspectors observed plant activities and conducted routine plant tours to assess equipment conditions, indications of operator work-arounds, procedural adherence and compliance with regulatory requirements.

Operators conducted control room activities in a professional manner with staffing levels above those required by Technical Specifications. The inspectors verified operator knowledge of ongoing plant activities, the reason for any lit annunciators, safety system alignment status, and existing fire watches. The inspectors also routinely performed independent verification from the control room indications and in the plant that safety system alignment was appropriate for the plant's current operational mode.

O1.2 Normal Reactor Shutdown for Refueling Outage 17R

a. Inspection Scope (71707)

To assess operations performance, the inspectors made periodic observations of the normal reactor shutdown and transition to shutdown cooling to commence refueling outage 17R.

b. Observations and Findings

Operators began reducing power on September 25 and took the main turbine generator off line at 1:06 a.m. on September 26. The operators performed a controlled plant cooldown in accordance with plant procedures. They also properly controlled the transition to the shutdown cooling system. The plant staff prepared an outage risk assessment to provide guidelines to ensure reliability and availability of plant systems and components for decay heat removal and inventory control.

Operators and outage management effectively implemented the outage risk management plan to control plant conditions.

c. Conclusion

Operators properly performed a normal shutdown and transition to shutdown cooling to commence refueling outage 17R.

O1.3 Refueling Activities

a. Inspection Scope (71707, 60710)

The inspectors reviewed the procedures for refueling operations and observed related activities to assess operator performance during refueling.

b. Observations and Findings

Operators and technicians adequately performed refueling bridge testing prior to conducting refueling activities. The inspectors observed proper use of procedures and good communications while technicians operated the reactor building crane and the refueling bridge. Operators displayed good attention and safety focus in identifying problems involving the bridge coordinate indicator, mechanical binding of the bridge mast, mast full extension, and water clarity. Despite the challenges, operators appropriately delayed refueling activities to document the problems and initiate corrective actions. Station management remained actively involved and supported the operators' efforts to correct the problems, even though the delays directly impacted the outage critical path schedule. Reactor engineering effectively supported operations to ensure conservative reactivity shutdown margins and adherence to the fuel movement plan.

c. Conclusion

Operators conducted refueling activities safely despite several equipment challenges, water clarity problems, and delays to the critical path schedule.

O2 Operational Status of Facilities and Equipment

O2.1 Loss of Shutdown Cooling (SDC)

a. Inspection Scope (71707, 37551)

During the refueling outage, operators experienced a loss of SDC. The inspector assessed operator response, corrective actions, and maintenance rule implications following the loss of SDC. The inspector reviewed NUREG 1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, to aid in risk assessment.

b. Observations and Findings

Event

At 1:19 a.m. on September 29, 1998, with the plant in the cold shutdown condition, the SDC valves closed due to a failed 'B' recirculation loop suction temperature thermocouple. This resulted in a loss of SDC (designed to isolate if temperature exceeds 350°F) and a plant heatup from 117°F to 131°F.

Licensee Response

Operators received an alarm in the control room and responded promptly. Operators noted that the 'B' recirculation loop suction temperature thermocouple failed high (reading 882°F), used an accurate alternate reactor coolant temperature indication (since SDC isolated and only the 'B' recirculation pump was in service), and initiated action to bypass the faulty temperature indication to restore SDC. At 2:04 a.m., instrumentation and control (I&C) technicians installed an electrical jumper bypassing the SDC isolation interlock for the 'B' loop and operators restored the SDC system to service. Operators noted that reactor coolant temperature had increased to 131°F prior to SDC restoration. The 'B' loop temperature indication eventually returned to normal.

The group shift supervisor (GSS) made a timely and appropriate Four-Hour NRC Notification (EN 34847) in accordance with 10 CFR 50.72(b)(2)(iii)(B) and informed the resident inspector. The shift technical advisor initiated CAP 1287. The GSS considered the SDC isolation a functional failure of the SDC system. The multi-disciplined Management Review Team concurred with the GSS and determined that CAP 1287 was significant and requested a root cause evaluation.

Immediately following the SDC isolation, the GSS contacted the Drywell Coordinator and inquired about on-going work in the vicinity of the 'B' loop thermocouple. The Drywell Coordinator investigated the area, observed damage to the thermocouple's conduit elbow, and noted that no one had been in the area and no work had been scheduled in the area. On October 17, I&C technicians repaired the thermocouple connection under job order 524534. Technicians reported that the conduit appeared to have been stepped on. They assumed that the damage occurred while setting up the drywell for the 17R outage. Technicians noted that a lack of full thread engagement at the conduit elbow contributed to the failure. Maintenance's root cause evaluator determined the cause to be less than adequate work practices due to workers bumping or steeping on the conduit and due to lack of full thread engagement.

The SDC system engineer evaluated the SDC isolation design basis and proposed a change to procedure 305, *Shutdown Cooling System Operation*, to bypass the SDC isolation interlocks when reactor coolant temperature is less than 212°F. Operations approved the procedure change on October 8.

Design Basis Review

The inspector reviewed the Updated Final Safety Analysis Report (UFSAR) and noted that the SDC system is designed to isolate above 350°F and receives an isolation signal based on a high temperature condition at any one of the five recirculation loop suctions. The inspector reviewed the electrical schematic drawings for the reactor protection system and determined that trip control relays operated properly to isolate the shutdown cooling system.

Maintenance Rule Implications

The inspector reviewed the event and recalled a similar SDC isolation in April 1998. The inspector revisited CAP 585 (dated 4/28/98) and noted that the 'B' recirculation loop suction temperature thermocouple failed high, resulted in a SDC isolation signal, and the GSS considered it a functional failure. Shortly thereafter, the 'B' loop temperature indication returned to normal. The April SDC isolation did not receive a lot of visibility as the plant was at power and the SDC system was not required to be operable at the time. Since the SDC is a risk significant system, GPUN's maintenance rule program required placement of the SDC system in (a)(1) status if the expert panel considered the failure a maintenance preventable functional failure (MPFF). The SDC system engineer stated that the SDC system remained in (a)(2) status and that he did not believe that a recirculation loop thermocouple failure would count as a functional failure against his system. The inspector discussed maintenance rule implications regarding the April and September SDC isolations with the Maintenance Rule Coordinator and the SDC system engineer.

In July 1998, I&C used job order 524534 to troubleshoot the 'B' loop temperature thermocouple. Technicians found no indication of a problem in the circuit outside of the drywell. Maintenance left the job order open to allow installation of a temporary modification (to bypass the isolation interlock) if needed and to allow additional troubleshooting in the drywell when opened. The expert panel could not determine if the failure was maintenance preventable; thus, the system engineer was not required to place his system in (a)(1) status and establish a corrective action plan. However, the inspector noted that engineering did not thoroughly evaluate the need to improve SDC system availability below 212°F until after the loss of SDC in September 1998. In addition, maintenance did not place a high priority on job order 524534 as they scheduled it five days into the outage. The inspector concluded, based on all the available information, that the same degraded condition (a loose electrical connection at the 'B' loop thermocouple conduit elbow) caused both SDC isolation signals (4/28/98 and 9/29/98) and more aggressive actions may have prevented the second, more risk significant, isolation. In response to the system engineer's comment regarding which system (SDC or reactor recirculation) gets charged with the functional failure, the Maintenance Rule Coordinator stated that he would review their program requirements to ensure appropriate functional failure accounting.

Risk Assessment

The Oyster Creek Individual Plant Examination (IPE) did not address shutdown risk or loss of SDC. Following the September 1998 loss of SDC, GPUN probability risk assessment analysts did not perform any quantitative conditional core damage frequency calculations to evaluate the relative risk of the event.

At the time of the SDC isolation, plant status affected the risk as follows:

- 70 hours since reactor shutdown from 70% power (a high decay heat load significantly reduces the time available for SDC recovery before boiling or core uncovery)
- 117°F initial reactor coolant temperature (low in band - farther from boiling condition)
- actual heatup rate : 0.45°F/minute (approximately 3 hours to boil assuming no other operator action)
- water level: 250" above top of active fuel (if the refueling cavity was filled - flooded up with the gates removed - the time available for SDC recovery would be increased)
- reactor head installed but detensioned (events that occur when the head is removed are typically less significant than those that occur with the head on, since water inventory makeup combined with core region boiling provides residual heat removal)
- primary containment relaxed, secondary containment established with one operable train of standby gas treatment (reduction in defense-in-depth relative to offsite release)
- reactor water cleanup available (contributed to lower heatup rate)
- electrical power - No.1 EDG, S1A startup transformer, 'C' vital 4160V bus, 'A' non-vital 4160V bus out of service; the A1,A2,A3 480V buses were cross-tied to the 'B' 480V buses (redundancy reduced, typical of outage conditions, cross-tied buses allowed 2 reactor building closed cooling water pumps and 2 service water pumps to remain available)
- isolation condensers drained, main condenser unavailable (one less means of decay heat removal available)
- combustion turbines and station blackout transformer available (redundant electrical power if needed)
- core spray system in reduced availability - core spray pumps 'A' and 'D' out of service (reduction in available makeup sources)
- contingency decay heat removal paths available and specified in plant configuration risk assessment guide: (1) reactor water cleanup system and letdown to the hotwell, and (2) use of electromatic relief valves and torus cooling - main steam line plugs not installed yet, torus open but still available as suction strainer replacement had not commenced (availability of diverse systems that can operate independently of the SDC system reduces the risk associated with losing SDC)

Overall, the relatively low reactor water temperature, prompt operator response, availability of reactor water cleanup, and the proper operation of the SDC system following bypass of the isolation signal resulted in an increased margin to boiling. In the event of boiling, contingency decay heat removal paths and adequate makeup water sources existed to further reduce the risk of core damage.

Communication and Coordination

On October 18, the inspector noted that operators had not implemented the SDC procedure change (approved on 10/8/98) which directed operators to bypass the SDC isolation interlocks when less than 212°F. The GSS and several operators appeared unfamiliar with the recent procedure change and the reason for it. The operating shifts did not receive any apparent guidance from operations management regarding bypass of the SDC interlocks. The inspector discussed the status of the interlocks and the new procedural guidance with operations management. Management stated that they had intended to implement the new guidance but were not aware that the new revision had made it to the control room. On October 20, operators bypassed the SDC interlocks in accordance with procedure 305, Step 4.3.19.

c. Conclusions

Operations responded appropriately to a loss of shutdown cooling. Engineering did not thoroughly evaluate an April 1998 shutdown cooling system functional failure to preclude this occurrence. Operations management did not clearly communicate and coordinate a subsequent operating procedure change implemented to improve shutdown cooling system availability while shutdown.

O4 Operator Knowledge and Performance

O4.1 Technical Specification Awareness While Shutdown (71707)

During the refueling outage, the inspector reviewed operating logs and conducted equipment and panel walkdowns throughout the plant to verify that operator's conducted activities in accordance with Technical Specifications (TS). Operators routinely made appropriate log entries concerning changes in equipment and plant status. Senior reactor operators (SROs) maintained the Limiting Conditions for Operation (LCO) Log up-to-date, remained cognizant of TS-required equipment status, and ensured proper implementation of LCO action statement requirements. SROs ensured that operators met all TS-required conditions prior to moving control rods or performing core alterations. On several occasions, when issues surfaced that raised operability concerns regarding electrical power sources, SROs promptly declared the source inoperable and took appropriate actions to place the plant in a safe and stable condition. The inspector concluded that SROs demonstrated a good TS awareness in effectively tracking and implementing TS LCO action statement requirements during the refueling shutdown.

O8 Miscellaneous Operations Issues

O8.1 Institute of Nuclear Power Operations (INPO) Report Review (71707)

In May 1998, INPO conducted a Training Accreditation Evaluation Review for non-licensed operators, reactor operators, senior reactor operators, shift managers, and shift technical advisors. In June 1998, an INPO team conducted an evaluation of site activities to make an overall determination of plant safety, to evaluate management systems and controls, and to identify areas needing improvement. The licensee received the final reports in August 1998. The inspectors and the DRP Branch Chief reviewed the reports during the inspection period. The inspectors determined that the INPO assessment was generally consistent with the most recent NRC assessment of the licensee's performance.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 Maintenance Activities (62707)

The inspectors observed selected maintenance activities on both safety-related and non safety-related equipment to ascertain that the licensee conducted these activities in accordance with approved procedures, Technical Specifications, and appropriate industrial codes and standards. The inspectors observed all or portions of the following job orders (JO):

- JO 525560 'B' MG Set Generator and Motor Re-installation
- JO 528137 Fabricate Contingency Replacement Assembly for V-16-0001
- JO 525937 Re-assemble MSIV (V-1-0008)
- JO 523036 Diesel Generator 18-month Inspection
- JO 522338 DC Hi-Pot Test of the No. 1 EDG Power Feed Cable
- JO 522480 Calibrate Emergency Diesel Generator Electrical Equipment and Instruments
- JO 528257 Replace the Fuel Grapple Air Cylinders and Inspect/Replace Hoses and Fittings
- JO 522544 Containment Isolation Valve - Instrument Air System Inspection
- JO 522628 Calibrate Fuel Zone Instruments

- JO 522566 Feedwater Control Instrumentation Calibration
- JO 525805 125VDC Power Panel DC-F Clean and Inspect
- JO 528273 Troubleshoot and Repair the Refueling Platform

Maintenance personnel obtained approval for work and conducted activities in accordance with approved job orders and applicable technical manuals and instructions. Personnel appeared knowledgeable of the activities and observed appropriate safety precautions and radiological practices.

M1.2 Surveillance Activities (61726)

The inspectors performed technical procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. They verified that the surveillance tests were performed in accordance with Technical Specifications, approved procedures, and NRC regulations. The inspectors reviewed all or portions of the following surveillance tests:

- 205.62 Refueling Bridge Check-Off
- 610.4.003 Core Spray Valve Operability and IST
- 612.4.001 Standby Liquid Control Pump and Valve Operability and IST
- 620.4.004 Source Range Monitor Test and Calibration (Front Panel Test)
- 636.4.003 Diesel Generator Load Test
- 651.4.001 Standby Gas Treatment System Test
- 654.4.003 Control Room HVAC System Operability Test
- 656.4.001 Refueling Interlock Circuit Surveillance
- 676.4.001 Drywell Equipment and Floor Drain Sump Isolation Valve Operability and IST
- 680.4.007 Safety Related Equipment Verification
- 680.4.010 Local Shutdown Panel LSP - 1B32 Functional Test
- 604.4.023 Drywell/Torus Vent and Purge Isolation on Drywell High Radiation Functional Test
- 617.4.014 Alternate Rod Insertion Logic Test

- 634.2.001 Main Station Battery Discharge and Low Voltage Relay Annunciator Test

- 634.2.007 Main Station Batteries Service Test

Personnel used the appropriate procedure, obtained prior approval, and completed applicable prerequisites. Personnel used properly calibrated test instrumentation, observed good radiological practices, satisfied Technical Specification requirements, and properly documented test results. Qualified technicians conducted the tests and appeared knowledgeable about the test procedure.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Outage Activities (General Overview)

a. Inspection Scope (60710, 62707)

The inspectors attended outage management meetings, reviewed corrective action reports, observed outage maintenance, and conducted frequent plant walkdowns to assess the control of outage activities.

b. Observations and Findings

Preplanning

Logistical Support (planning and scheduling) developed a thorough refueling outage plan. Station management met frequently for months prior to the outage to ensure proper outage work scope. The management team loaded the maintenance backlog into the plan along with a majority of the existing work-arounds, control room deficiencies, and temporary modifications. Logistical Support used a risk-informed approach to schedule outage activities. The schedule reflected appropriate consideration of defense-in-depth, redundancy, and shutdown risk.

On-Line Preparation

Prior to the outage, maintenance services erected numerous scaffolds and pre-staged various materials and equipment throughout the plant. Maintenance properly controlled these activities and operations demonstrated good ownership and configuration control. In one isolated occurrence, scaffold erectors inadvertently bumped a sensitive control air instrument resulting in a nitrogen compressor failure alarm and an automatic swap-over to air in the drywell. Drywell pneumatic systems functioned as designed, operators responded promptly to restore the nitrogen lineup, maintenance initiated Corrective Action Process (CAP) 1217, and maintenance management stopped the work and counseled the involved workers.

Risk Assessment

The outage management team developed a comprehensive risk plan. Management tailored the plan specifically to address the various plant configurations, from a risk perspective, expected during the 17R outage. The risk assessment plan addressed decay heat removal, reactivity control, containment control, electrical power availability, and inventory control. The plan identified key activities, equipment out of service, and available contingency courses of action in each area. Plant personnel appeared knowledgeable of the plan and effectively implemented it.

Foreign Material Exclusion (FME) Control

Station management demonstrated a commitment to quality in hiring an FME Coordinator in July 1998 and providing additional contract FME monitors during the outage. The FME Coordinator conducted FME training for over 500 outage workers prior to the outage and provided close oversight of outage activities, especially on the refuel floor. Plant personnel demonstrated very good FME control in the torus (see Section M2.2) and on the refuel floor, however, they did not demonstrate the same attention to detail in the drywell (see Section R1.2). Overall, plant personnel used good FME controls in the conduct of outage work and promptly initiated corrective actions for identified concerns.

Management Oversight

Senior managers demonstrated a good safety-focus while managing refueling outage activities. Management conducted frequent plant tours, including weekends and back shifts, and provided close oversight of outage work. At the outage status meetings, senior management repeatedly emphasized the importance of maintaining nuclear safety and personnel safety. Management fully supported the staff in this regard and effectively removed unnecessary schedule pressure, even though the staff's questioning attitude and cautious approach delayed the critical path schedule on several occasions.

Corrective Action Process

Plant personnel continued to identify conditions adverse to quality and initiate CAPs. The number of CAPs per day more than doubled due to the volume of job orders and the fact that plant personnel were out looking for problems. The multi-disciplined Management Review Team met daily to discuss the CAPs and appropriately addressed operability, reportability, and significance level. The Corrective Action Coordinator appropriately marked and tracked certain CAPs as "restart required."

Tagging

Operations demonstrated good attention to detail and a standard for excellence in effectively controlling the removal, tagging, and restoring of numerous systems, involving thousands of tags, in an error-free manner. Inspectors independently verified hundreds of tags in the plant and randomly verified several completed tagging requests. All tags were properly placed and components were properly positioned. Tagging errors did not contribute to any outage event, unexpected water transfer, equipment malfunction, or personal injury. A review of the CAP database revealed no reported tagging errors.

Personnel Safety

Plant personnel demonstrated good OSHA work practices and experienced no "lost time" or "restricted duty" injuries, and a dramatic reduction in minor injuries compared to previous outages (as of 10/19/98, Safety totaled 21 minor injuries in 17R compared with 67 in 16R). The Safety Department contributed significantly to this accomplishment through pre-outage orientation training, increased safety engineer staffing and field observations (24-hour, 7-day coverage), Plan of the Day Meeting safety discussions, job order package safety hold points, and periodic Safety Grams. In addition, senior plant management frequently emphasized personnel safety and clearly communicated their expectations concerning worker safety during outage status meetings and plant tours.

Conduct of Maintenance

In general, maintenance personnel demonstrated good work practices, conducted activities in accordance with approved procedures, demonstrated a good questioning attitude, and properly documented work. In several cases, where workers fell short of management's expectations in these areas, maintenance personnel appropriately documented the deficiency on a CAP report and engaged the corrective action process. Maintenance technicians used properly calibrated instruments and current procedures. Supervisors provided close oversight of on-going work and ensured job orders were scheduled and approved prior to the start of work.

Quality Assurance and Quality Verification

Nuclear Safety Assessment developed and implemented a comprehensive outage inspection plan. The plan detailed over 30 refueling outage activities, listed the responsible assessors, and provided a safety-focused inspection scope. Nuclear Safety Assessment and Quality Verification provided around-the-clock outage coverage, made important contributions (such as identifying concerns with prerequisites for reactor head removal, refueling procedure usage, risk assessment guidance implementation, and radiological safety) to ensure nuclear safety and quality was maintained, and appropriately documented concerns via the corrective action process.

Refueling Equipment

Operations and maintenance satisfactorily completed the required refueling equipment surveillances prior to the outage. During refueling activities, plant personnel demonstrated a good questioning attitude, identified and initiated corrective actions for deficiencies, and promptly stopped refueling activities while maintenance and engineering evaluated the deficiencies. Maintenance provided good support to operations in conducting timely and lasting repairs to the refueling equipment.

However, engineering and maintenance did not adequately improve refueling equipment reliability prior to outage 17R to preclude challenges to operators during the outage. In particular, following four Maintenance Preventable Functional Failures (MPFFs) during the 16R outage, the maintenance rule expert panel placed the fuel handling equipment system in (a)(1) status on April 5, 1997. The expert panel planned to re-evaluate the system status following the 17R outage. Corrective actions included upgrading the vendor manuals, upgrading maintenance procedures, and scheduling appropriate work.

Although no repetitive MPFFs occurred in the 17R outage, engineering and maintenance did not effectively upgrade maintenance procedures nor schedule all of the appropriate work. For example, mechanical binding in the bridge mast interfered with full extension of the 3rd segment of the mast (CAP 1315). Maintenance disassembled the mast to correct the condition. Maintenance identified that the vendor recommended preventive maintenance interval (which involved disassembling, cleaning, and inspecting the bridge mast prior to each outage) was not in place prior to entering the outage. In addition, a degraded wire rope cable on a retractable spring hoist, used with the channel handling boom, was identified in 1995 but maintenance never implemented the job order. The inspector determined that these licensee-identified deficiencies, along with several others identified during the outage, do not constitute violations of maintenance rule requirements, however, they do indicate that maintenance did not effectively maintain the material condition of the refueling equipment.

c. <u>Conclusions</u>

Logistical Support (planning and scheduling) provided a thoroughly planned, risk-informed, and properly scoped refueling outage plan. Maintenance personnel demonstrated good work practices, conducted activities in accordance with approved procedures, demonstrated a good questioning attitude, properly documented work, and appropriately engaged the corrective action process. Maintenance provided good support to operations in the conduct of refueling equipment repairs, however, the material condition of the refueling equipment presented several challenges during refueling. Plant personnel demonstrated good foreign material exclusion control.

Operations demonstrated good attention to detail and a standard for excellence in effectively controlling the removal, tagging, and restoring of numerous systems, involving thousands of tags, in an error-free manner. Senior management demonstrated safety-conscious decision making, a commitment to worker safety, an active involvement and plant presence, and a concerted effort to remove schedule pressure.

The Safety Department provided high-quality oversight of outage activities and contributed to a safe work environment. Nuclear Safety Assessment developed and implemented a safety-focused and comprehensive outage inspection plan. Nuclear Safety Assessment and Quality Verification provided around-the-clock outage coverage, made important contributions to ensure nuclear safety and quality was maintained, and appropriately documented concerns via the corrective action process.

M2.2 Emergency Core Cooling System (ECCS) Suction Strainer Replacement

a. Inspection Scope (37551, 62707)

During the 17R refueling outage, maintenance implemented modification No. OC-MD-F227-001, replacement of the *Emergency Core Cooling* (ECCS) *Strainers*. The licensee replaced their ECCS strainers in response to NRC Bulletin 96-03, *Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors*. The inspectors reviewed the ECCS suction strainer replacement modification package, associated 10 CFR 50.59 safety evaluation, in-field documentation, and post maintenance test (PMT) results. The inspectors conducted frequent torus area observations, including a final torus closeout inspection, to assess radiological controls, work practices, and FME control. Inspectors observed diver activities via underwater cameras viewed at a diver control point in the reactor building. The inspectors observed various stages of strainer assembly, including portions of Quality Verification's (QV) bolt torque verifications.

b. Observations and Findings

The purpose of the strainer replacement was to significantly increase the post loss of coolant accident (LOCA) debris loading capability of the ECCS strainers. The modification basically involved unbolting and removing the existing strainers and installing and bolting-up of the new strainers. The modification also replaced four torus catwalk support struts that interfered with the new strainers.

Engineering properly planned and performed thorough evaluations in preparation for the modification. Engineering performed an acceptable 10 CFR 50.59 safety evaluation to support the modification. Engineering identified an unreviewed safety question (USQ) due to the use of containment overpressure (1.25 psig) for the calculation of ECCS net position suction head (NPSH). Engineering noted that the original NPSH calculation did not take credit for containment overpressure; therefore, a change to the Oyster Creek Nuclear Generating Station licensing basis was required. Oyster Creek licensing submitted the appropriate documentation to the NRR Project Manager for NRC review. The licensee's use of containment overpressure of 1.25 psig was not an NRC restart concern. The design change package was well-written, contained appropriate QV hold points and FME controls, and specified an appropriate performance based PMT. The package required few revisions during the strainer replacement work and technicians experienced little or no problems in implementing the modification package as written.

During the conduct of the strainer replacements, maintenance technicians demonstrated good work practices, attention to detail, proper radiological controls, heightened FME awareness, and accurate and up-to-date work documentation. For example, technicians promptly implemented appropriate FME controls to cover the ECCS suction nozzle after removal of the old strainer and prior to installation of the new strainer. Workers appropriately used the corrective action process to document deficiencies (CAPs 1285, 1293, 1297, 1307, 1309 and 1403). Radiological Control technicians effectively controlled radiological conditions in the torus work area and consistently reinforced good ALARA practices.

Quality Verification personnel conducted near-continuous oversight of the replacement activities. The QV specialists provided safety-focused independent quality control during the strainer replacements. In general, QV ensured excellent FME control in the torus. Quality Verification's commitment to quality was particularly evident during the divers' final underwater FME inspection. The QV specialist clearly and directly communicated to the on-site General Electric (GE) project manager that no time constraint existed on the final dive - it would take as long as it took to conduct a thorough underwater inspection. In response to QV's expectations, divers conducted a comprehensive and well-documented inspection and demonstrated excellent attention to detail.

Following maintenance's and QV's final torus closeout inspection, the inspector conducted an internal torus walkdown. In general, the torus catwalk, water surface, and surrounding internal areas were very clean and free of debris. However, the inspector did amass two handfuls of small debris (tie wraps, short pieces of wire and duct tape, one 8" length of rope, and several small screws) collected from the torus catwalk. The majority of this small debris appeared old and not associated with the strainer replacements, however, it did indicate that maintenance and QV did not demonstrate good attention to detail during their final walkdown inspection.

As a PMT, operations ran the 'C' containment spray pump and engineering recorded applicable containment spray system operating parameters to verify that the clean strainer pressure drop was bounded by the GE design specification. Engineering compared the old strainer pressure drop (recorded on 9/23/98) to the new strainer pressure drop (recorded on 10/15/98) and noted a slight increase of 0.2 psig. Engineering determined that this pressure drop was bounded by the GE analysis and documented satisfactory completion of the PMT. Engineering attributed the slight change in suction strainer differential pressure to the more torturous flow path taken through the new strainers.

c. <u>Conclusions</u>

Engineering provided a high-quality and well developed design change package for the emergency core cooling system suction strainer replacement. Maintenance demonstrated good work practices, proper radiological controls, and maintained the work package documentation accurate and up to date. Radiation Protection technicians effectively controlled radiological conditions and access. Quality

Verification conducted almost continuous oversight of strainer replacements and, in general, ensured excellent foreign material exclusion control in the torus.

M3 Maintenance Procedures and Documentation

M3.1 Unplanned Actuation of an Engineered Safety Feature

a. Inspection Scope (61726)

The inspectors observed operators' recovery from an unplanned actuation of an engineered safety feature. The inspectors reviewed the root cause evaluation, procedure change and performance of portions of the surveillance test.

b. Observations and Findings

On September 16, 1998, technicians performed procedure 610.3.205, *Core Spray System 2 Instrument Channel Calibration and Test.* During the performance of the surveillance, technicians demonstrated a questioning attitude as they noted a voltage at a point that they had expected to find no voltage. Operations and maintenance personnel decided to stop the test, complete the current section of the surveillance, and resolve the problem. While securing from the surveillance procedure, technicians returned a cross-channel normal-inhibit switch to normal. With the switch in normal, connection of an ohmmeter across two terminals acted as an electrical jumper across the cross-channel relays and provided an actuation signal for the core spray system 1, which was not being tested.

The core spray system 1 pumps started as designed after receiving the actuation signal from the core spray system 2 logic. Control room staff responded appropriately to assess the condition and restore the core spray system 1 to standby status. Core spray system 1 injection valves were not affected during this event, and remained closed and operable. Core spray system 1 was available to provide core cooling if needed. Therefore, there was no resultant impact on nuclear safety or safe plant operation.

Maintenance completely secured fro... the surveillance and initiated CAP 1204. Operations returned core spray system 2 to standby readiness. Maintenance management requested a thorough critique of the event prior to allowing any further surveillance testing involving the core spray system. The group shift supervisor promptly made a Four-Hour NRC Notification for the inadvertent actuation of an engineered safety feature.

Maintenance's root cause evaluators determined that core spray system 1 started because the surveillance procedure did not specify in enough detail that technicians should remove their ohmmeter after ea h step. This was the first time technicians performed the surveillance since a procedure wr ter had revised the procedure in July 1998 to incorporate requirements of NRC Generic Letter 96-01, *Testing of Safety-Related Logic Circuits*. On September 22, maintenance revised the procedure to include improved guidance concerning ohmmeter connections.

Following this revision, technicians successfully completed the core spray surveillance test. The Maintenance Director planned to submit a Nuclear Network information report on the event. In addition, maintenance planned to complete a review of other procedures revised to incorporate Generic Letter 96-01 guidance for similar inadequacies by April 1999.

The inspector attended the critique and noted that during the surveillance test, because the procedure step was new and did not clearly state whether the ohmmeter should remain connected or not, a technician questioned the connection of the ohmmeter. The technician discussed this concern with his supervisor. The supervisor decided to leave the ohmmeter installed since they would recheck readings across the same two terminal points repeatedly throughout the procedure. The procedure did not contain enough detail concerning the ohmmeter, and the supervisor decided to leave the meter installed to reduce the challenges to human performance due to repeatedly removing and reinstalling test leads in the panel. The procedure writer did not intend for the ohmmeter to remain installed; however, the maintenance supervisor did not demonstrate a good questioning attitude and did not contact the procedure writer for clarification.

c. Conclusion

Maintenance personnel caused an unplanned actuation of the core spray system 1 due to a procedure weakness. Maintenance supervision did not demonstrate a good questioning attitude while implementing the procedure. Maintenance management took prompt and appropriate corrective actions.

M8 Miscellaneous Maintenance Issues (92902, 90712)

M8.1 (Closed) Violation 50-219/97-03-01b: Failure to Determine an Appropriate Isolation Boundary for Maintenance. Inspectors discussed this item in NRC Integrated Inspection Report 50-219/98-02 Section M8.2. The item remained open pending licensee corrective actions to address the involvement of appropriate personnel in the rescheduling of maintenance tasks. Planning supervision established a requirement that engineering must perform a 10 CFR 50.59 review for all maintenance tasks performed in any plant mode other than that which was originally prescribed. Planning stated that this requirement would be incorporated in a future revision to Procedure 2400-ADM-1220.18, *Preventive Maintenance*. The inspector determined that the licensee took appropriate actions to prevent recurrence. This item is closed.

M8.2 (Closed) Licensee Event Report 98-12: Actuation of an Engineered Safety Feature by a Test Instrument Due to an Inadequate Procedure. Inspectors discussed this event in Section M3.1 of this inspection report. Inspector in-office review of LER 98-12 determined that the licensee's report met the 10 CFR 50.73 reporting requirements. This LER is closed.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 Evaluation of Fuel Channel indications

a. Inspection Scope (37551)

The inspectors reviewed engineering's actions and evaluation of fuel channel indications on fuel assemblies removed from the core during refueling outage 17R. The inspectors evaluated engineering's activities to determine their effectiveness on reviewing issues that could affect plant operations.

b. Observations and Findings

Video inspections as part of the 17R refueling outage revealed indications of an oxidation buildup on the outside of four fuel channels. Engineering was concerned that some of the indications appeared to be cracks and holes on the channel surface. Engineering initiated CAP 1338 to address the condition. Engineering formed a team that included in-house engineers, corporate engineering, other utilities which use the same channels, the channel manufacturer, the fuel supplier, a procurement engineering firm and a zircaloy expert to evaluate the condition. Engineering inspected an additional fifteen fuel channels of similar manufacture and exposure (three to four operating cycles) to evaluate the extent of the condition.

Team members performed a closer visual inspection, used a probe to check the metal, and performed an eddy current examination. They determined that the potential holes were due to markings from the roller balls from the control rod blades, and were not actual holes. The crack-like appearances had sufficient base metal and were determined to be fretting marks. Oxidation of the channels occurred due to the interaction of zircaloy and stainless steel. Engineering review of the data determined that similar patterns of wear and oxidation have been seen in other boiling water reactors.

Engineering advised operators to shuffle one channel to the periphery for use during the next operating cycle. Operators stored the remaining channels in the spent fuel pool and did not use them when refueling the reactor. Engineering concluded that the oxidation and markings were not a safety concern and do not affect safe operation of the plant. The inspectors reviewed the engineering evaluation and determined that engineering took appropriate and conservative actions.

c. Conclusion

Engineering demonstrated good awareness and performed an appropriate evaluation of oxidation and marking on the outside of several fuel channels.

E2.2 Small Bore Piping Design Issues

a. Inspection Scope (37551, 90712)

As part of ongoing activities to characterize nuclear safety-related (NSR) small bore lines at Oyster Creek, structural engineering identified several cases in which NSR small bore lines were not in compliance with the Oyster Creek design basis. The inspector assessed the licensee's actions in response to this condition and reviewed LER 98-11.

b. Observations and Findings

In June 1997, a plant deviation report (455-97) raised a concern as to whether NSR small bore piping at Oyster Creek met B31.1-1955 seismic and thermal requirements as described in the Oyster Creek Updated Final Safety Analysis Report (UFSAR). In response to the deviation report, GPUN developed a program to evaluate whether applicable NSR small bore lines were in compliance with the design basis.

Engineering performed a small bore piping walkdown of in-scope systems and used a design basis screening criteria to disposition lines in the plant. Engineering dispositioned lines that met the screening criteria as having met the code requirements. Engineering committed to analyze all lines that did not meet the screening criteria using the configuration documented during the walkdowns. Engineering conducted walkdowns of approximately 63 lines, accessible during power operation, during May and June of 1998. Approximately 2/3 of these lines met the screening criteria and required no further action. Engineering analyzed the remaining lines and noted the following discrepancies:

- On September 14 and 15, 1998, engineering identified apparent B31.1 design deficiencies involving small bore piping on the 'A' isolation condenser condensate return drain line and a containment spray system drain line. The licensee made Four-Hour NRC Notifications for each of these conditions (EN 34770 and EN 34773, respectively.) Subsequent engineering calculations determined that these small bore lines met the B31.1 design basis requirements.

- On September 28, 1998, engineering determined that the configuration of two ½" lines in the SDC system did not meet B31.1 requirements based on preliminary calculations. Engineering determined that the first clamp supports on each of these lines were located too close to an 8" large bore SDC line to accommodate the large vertical anchor motion at the 8"line.

- On September 29, 1998, engineering determined that the configuration of a ½" tubing line on core spray system I did not meet B31.1 requirements based on preliminary calculations. The thermal and seismic motions of the 14" large bore core spray line, could cause the ½" tubing to interfere with a structural concrete beam resulting in an over-stress condition on some of the ½" tubing fittings.

Although the configuration of the SDC and core spray small bore piping did not meet B31.1 code requirements, engineering determined that the lines were operable based on the guidelines found in Generic Letter 91-18, Revision 1. The inspector reviewed engineering's calculated stresses and found engineering's operability determination acceptable. Maintenance modified the core spray and one of the SDC small bore lines during the 17R outage to correct the deficiencies (JO 527797 and JO 527798). Maintenance planned to modify the second SDC small bore line, along with a large bore modification, in the 1st quarter of 1999.

Failure to ensure that small bore piping met B31.1 design basis requirements is a violation of Appendix B Criterion III, Design Control. This violation constitutes an additional example of NCV 50-219/98-02-01(see NRC Integrated Inspection Report 50-219/98-02 Section E2.1) and is not being cited individually. Further corrective actions for this additional example are expected to be taken in conjunction with corrective actions for the previous NCV.

Engineering planned to conduct walkdowns on 18 additional lines, not accessible during power operations, during the 17R refueling outage. Engineering planned to address generic conclusions concerning all NSR small bore lines following analysis of the 17R piping walkdowns. The licensee planned to submit a supplement to LER 98-11 (see section E8.12) to address any additional piping deficiencies and generic conclusions.

c. Conclusions

Engineering demonstrated good initiative in conducting small bore piping walkdowns and evaluations. Engineering appropriately documented and initiated corrective actions for identified deficiencies.

E2.3 Isolation Condenser Tube Bundle Inspection and Replacement

a. Inspection Scope (37551, 62707)

In response to a September 1997 Nine Mile Point Unit 1 isolation condenser tube failure, the Oyster Creek isolation condenser system engineer established a comprehensive Isolation Condenser Action Plan to fully evaluate condenser tube condition (see NRC Integrated Inspection Report No. 50-219/97-09 Section E2.2). Part of the plan involved investigating non-destructive examination (NDE) inspection techniques. In response, engineering developed and implemented an isolation condenser hydrostatic test and inspection plan. The inspector assessed engineering's inspection plan, their preliminary findings, and their corrective actions.

b. Observations and Findings

Engineering coordinated with planning to prepare Modification Document OC-MD-H037-001, *Isolation Condenser Tube Bundle Testing and Replacement*. The modification package included appropriate FME controls and QV hold points, and adequately considered applicable design basis requirements. Engineering and operations collaborated to develop Special Procedure 98-004, *Isolation Condenser Hydrostatic Test and Inspection*, to satisfy the testing requirements of OC-MD-HO37-001.

On September 28, 1998, engineering identified a tube to tubesheet leak on the 'B' isolation condenser while conducting Special Procedure 98-004. Each of the two isolation condensers contains two tube bundles connected to the reactor coolant system. The 'B' north tube bundle had a leak of approximately 1 quart per minute. The 'B' south tube bundle had a leak of approximately 1 drop per minute. Both leaks were on top of the tube bundle. Engineering notified operations and initiated CAP 1998-1281. Operations made a timely Four-Hour NRC Notification in accordance with 10 CFR 50.72 (b)(2)(i) for a condition, found while shutdown, that involved degradation of a principal safety barrier. Hydrostatic testing of the 'A' isolation condenser tube bundles revealed no leakage.

As a contingency, prior to the outage, the licensee procured a set of tube bundles. Maintenance implemented JO 527757 to replace the 'B' isolation condenser tube bundles. Preliminary engineering analysis determined that transgranular stress corrosion cracking, coupled with cyclic thermal stresses, caused the cracks. Engineering gathered "as found" data from the failed bundles to re-evaluate their previous "leak-before-break" analytical approach and planned to address continued operability of the 'A' isolation condenser prior to restart (tracked via CAP 1998-1281-2).

c. Conclusions

Engineering led plant staff's efforts to hydrostatically test and inspect the isolation condenser tube bundles. Engineering appropriately documented identified deficiencies, promptly addressed reportability, and initiated actions to replace leaking tube bundles.

E4 Engineering Staff Knowledge and Performance

E4.1 Leaking Electromatic Relief Valve (EMRV) Pilot Valve Root Cause Determination

a. Inspection Scope (37551)

The inspector assessed engineering's root cause analysis of leaking EMRV pilot valves. The inspector reviewed Dresser Industries Report SV-426 and engineering's corrective actions in response to the SV-426 report findings.

b. Observations and Findings

In April 1998, maintenance identified steam wisping from 2 EMRV pilot valves during their 1000-psig drywell walkdown. Operations used CAP 1998-433 to document the condition and initiate corrective actions. The corrective action multi-disciplined Management Review Team determined that the CAP was significant and requested a root cause determination. Maintenance replaced the pilot valves (see

NRC Integrated Inspection Report 50-219/98-02 Section M4.2) and engineering sent the leaking valves to Wyle Laboratories where the original equipment manufacturer (Dresser Industries) could inspect the valves and determine the root cause.

On October 2, 1998, Dresser Industries provided Report SV-426, *Dimensional & Visual Evaluation of Pilot Valve Parts for 1525VX Relief Valve,* to GPUN. The report concluded that the EMRV pilot valve leakage was caused by 1) a thicker seat lip (.026" and .027" vice required thickness of .020") which reduced the flexibility of the seat and affected seat tightness and 2) incorrect position of the seat contact area caused by incorrectly manufactured seat angles (44° and 45° 10' vice required angle of 46°). The SV-426 report also identified that technicians failed to properly lap the valve seats and discs.

In response to the SV-426 findings, maintenance installed new pilot valve assemblies in all five EMRVs during the 17R outage. Engineering revised the QA/Technical requirements screen for the pilot valve assemblies to ensure future assemblies meet SV-426 requirements. Engineering initiated corrective action 1998-433-2 to revise maintenance procedures to incorporate SV-426 findings and action 1998-433-3 to revise the EMRV overhaul specification to ensure future rebuilds meet the SV-426 requirements.

Engineering management stated that Dresser Industries indicated that they (Dresser) intended to report the SV-426 findings in accordance with their 10 CFR 21, Reporting of Defects and Noncompliance, program requirements. Oyster Creek engineering planned to provide the information to industry peers through the Nuclear Network information system. In addition, engineering submitted a 10 CFR 21 package to GPUN licensing in case Dresser Industries did not produce their 10 CFR 21 report in a timely manner.

c. Conclusions

Engineering demonstrated a good questioning attitude in the conduct of a thorough root cause evaluation for leaking electromatic relief valve pilot valves. Oyster Creek engineering identified an industry concern regarding pilot valve manufacturing tolerances and initiated appropriate actions to replace installed pilot valves and to inform the nuclear industry of the discrepancy.

E8 Miscellaneous Engineering Issues (92903, 90712)

E8.1 (Closed) Unresolved Item 50-219/96-06-02: Use of Quartz for Fine Aggregate in HSM Roof. The NRC questioned the process by which GPUN and the vendor (VECTRA Technologies, Inc.) evaluated the use of quartz for fine aggregate in the horizontal storage module (HSM) concrete roofs associated with spent fuel dry storage. Specifically, the use of quartz for fine aggregate would have necessitated thermal expansion coefficient testing because it represented a change to the cask safety analysis report.

In response to the technical issue, GPUN planned to administratively limit the heat load of the dry-shielded storage canisters (DSC) so that the HSM general area and local concrete temperatures would not exceed specific temperature values. Subsequently, the subject of this open item, in conjunction with other NRC inspection findings of the vendor, raised broader concerns regarding the adequacy of VECTRA's quality assurance and design control programs. On January 24, 1997, VECTRA voluntarily issued a stop work order for fabrication of their HSMs and DSCs. Transnuclear-West, Inc. (TN-West), who had purchased VECTRA assets on November 21, 1997, committed to 44 corrective actions, the implementation of which had been the subject of NRC inspections. On May 6, 1998, the NRC concluded that TN-West had sufficiently implemented corrective actions to its quality assurance and design control programs to allow the resumption of limited fabrication activities.

In evaluating the concrete aggregate technical issue, the NRC completed vendor inspection 72-1004/97-209,and found that TN-West's resolution approach to modify the emissivity of the HSM heat shields to be consistent with the storage facility's design basis and the existing Certificate of Compliance. TN-West plans to submit its final resolution to GPUN for review and approval prior to implementation of any associated repair activities. NRC inspection 72-1004/97-209considered the technical issues resolved.

The inspector reviewed the associated documentation for the use of quartz for fine aggregate and discussed the technical issues and status with NRC, Office of Nuclear Material Safety and Safeguards, personnel. The inspector concluded that TN-West and GPUN have adequately resolved this issue. No violations of NRC requirements were identified. This item is closed.

E8.2 <u>(Closed) Unresolved Item 50-219/95-05-01</u>: Emergency Diesel Generator (EDG) Reliability Monitoring Program. This item was initially opened when it was identified that the licensee did not have a formal EDG reliability monitoring program to ensure that the assumptions made in the station blackout analysis would remain valid. The licensee subsequently formalized the program under Station Procedure 117.1, *Emergency Diesel Generators Reliability Program*, which was reviewed during NRC inspection 50-219/96-07. The item remained open following that inspection pending additional licensee experience and evidence of successful implementation of the program.

The inspector reviewed the EDG reliability data, reviewed a sample of quarterly diesel generator reliability and availability reports and discussed the program with the responsible system engineer. The inspector found the system engineer to be knowledgeable of the program and the EDG reliability indicators were being properly tracked and reliability calculations were current. Quarterly reports were thorough and provided a method for the data to be captured in the licensee document control system. The inspector concluded that the licensee program was adequate to ensure the station blackout analysis assumption relative to EDG reliability would remain valid. No violations of NRC requirements were identified. This item is closed.

E8.3 (Closed) Inspector Follow Item 50-219/95-22-01: Quality Assurance (QA) Monitoring of the Combustion Turbine (CT) Reliability Program. This item was opened when the inspector identified that Station Procedure 117.3, *Combustion Turbine Reliability,* required a quality assurance review of the CT reliability program. At that time the licensee's QA organization, the Nuclear Safety Assessment (NSA) Group, had not performed reviews in this area. Subsequent to the NRC inspection the licensee provided information regarding their plans to periodically monitor the CT reliability program.

The inspector discussed this item with a lead assessor in the NSA group and reviewed the results of audits that were performed subsequent to the initial NRC inspection in 1995. The NSA group performed a specific audit of the CT reliability program in June 1996 and also included CT reliability reviews in their monitoring program in January 1997 and December 1997. The inspector found the NSA assessments to be thorough as evidenced by the findings that were identified. The findings were documented in DRS and appropriate corrective actions were implemented. The inspector also noted that the CT reliability program has been included in the NSA biannual engineering audit and will also be included in the NSA monitoring program. The inspector concluded that the licensee actions are consistent with the procedural requirements. No violations of NRC requirements were identified. This item is closed.

E8.4 (Closed) Unresolved Item 50-219/96-07-03: Installation of AC fuses in DC powered control circuits. The licensee identified an instance where a fuse installed in a DC control circuit did not have a DC rating and this item was opened pending licensee determination if this condition existed in other DC applications. The licensee determined that they did not have an existing data base that would identify where AC rated fuses may have been used in DC circuits. As a result, the licensee performed field inspections of safety-related panels to assess the extent of the condition. The licensee found that most of the fuses were appropriately rated for DC applications. In the cases where they were not, a work order was generated to replace the AC rated fuses with an appropriate DC rated fuse. Since a 100% inspection of all fuses was not possible due to accessibility problems, the licensee also evaluated the potential effects of an AC fuse in a DC circuit. The evaluation concluded that the actions taken were adequate to address the condition based on several considerations including: 1) the fuses are only used in control circuits where available short circuit currents would be significantly less than in applications such as in distribution panels, 2) the fuses do not affect operability of equipment and would only potentially affect a component which has already experienced a failure, 3) there has not been a history of problems with fuse failures at the plant, and 4) the fuse control program requires engineering evaluation prior to installing a fuse other that an identical replacement.

The inspector reviewed the results of the walkdown and the fuse control program procedure, and discussed the issue with the responsible engineer. The inspector noted that the fuses in question were all installed in 125 volt DC circuits and the AC fuses that were identified all had voltage ratings of at least 250 volts. The inspector concluded that the licensee had adequately addressed this concern, including effects on circuit operability. No violations of NRC requirements were identified. This item is closed.

E8.5 (Closed) Unresolved Item 50-219/96-09-03: Control of Cable. This item identified that the licensee used the same stock numbers for medium voltage cables manufactured by three different manufacturers and that this could negate potential improvements resulting from the use of the most recent style cable.

The licensee used the same stock number because, although the manufacturing company name changed, the cable continued to be manufactured at the same facility and continued to meet the purchase requirements. To ensure the use of the preferred cable, the licensee procured new cable and removed all of the existing stock from the storeroom.

The inspector reviewed the licensee's evaluation and discussed the issue with the procurement engineer. The inspector concluded that licensee actions improved the program. However, the previous control of cable was adequate and no violations of NRC requirements were identified. This item is closed.

E8.6 (Closed) Unresolved Item 50-219/96-09-04: Voltage Drop Calculation. The licensee's battery sizing calculation indicated in the summary of results that the voltage drop calculation should be revised as a result of the calculated change in the battery load duty profile. However, the inspector noted that the calculation had not been updated 30 months later and concluded this was a possible weakness in the licensee's calculation control program.

Subsequent to that finding, an NRC inspection at the licensee's Three Mile Island facility (IR 50-289/96-201) also identified problems with the control of calculations. As a result, the licensee initiated a calculation improvement program to identify the cause of the problems and to make improvements. This program is applicable to both the Three Mile Island and Oyster Creek facilities.

Also, during NRC inspection 50-219/96-80 at Oyster Creek, the inspectors identified that no analysis had been performed to validate the voltage to the electromatic relief valves (EMRVs) would be sufficient to ensure operability. This condition would have been within the scope of the voltage drop calculation update which had not yet been completed. As a result of this finding, the licensee performed testing to determine the minimum required operating voltage for the EMRVs, implemented a modification to reduce the voltage drop and completed calculations to verify sufficient voltage would be provided to the EMRVs under all conditions.

The voltage drop calculation update was not complete at the time of this current inspection. However, the licensee indicated that the EMRVs would be the limiting components with respect to voltage drop and were confident that when complete, the calculation would validate that the voltage to all of the DC components would be sufficient to ensure operability.

The inspector discussed the specific voltage drop calculation with the responsible engineers and also discussed the calculation improvement program with the responsible project manager. The inspector concluded that based on actions already taken and those additional actions planned as a result of findings identified after inspection 96-09, this item is considered closed. Escalated enforcement action was taken by the NRC as a result of the findings in IR 50-219/98-80.

E8.7 (Closed) Unresolved Item 50-219/96-09-05: Medium Voltage Cable Testing Program. This item was open pending additional NRC evaluation of the licensee cable test program. Previous inspection had concluded that the cable test procedures were consistent with industry practice with the exception of conformance with field test voltages contained in IEEE 400-1991 (23kV) and AEIC CS6-87 (14 kV for cables greater than 5 years old). Also, the program and schedules were being revised such that the 6 year test interval goal was not being met.

The licensee program specifies a test of a new cable at 35 kV with a followup test two years later at 25 kV and then an additional test four years later at 10 kV. Periodic tests are performed at 10 kV to avoid the need to disconnect the cables so that the switchgear is not subjected to excessive test voltage.

The licensee cable testing schedule for the next refueling outage (17R) will result in all cables meeting the recommended test frequencies of the test program except for those associated with the startup and auxiliary transformers. Since these cables will exceed the program recommendations an engineering evaluation was performed and documented the bases for the test deferral.

The inspector reviewed the Medium Voltage Cable Test Program, dated August 11, 1997, discussed the program with the responsible engineer, and reviewed a sample of component history records associated with the cable program. The inspector found that the cable test program was clearly defined and, with minor differences, was consistent with industry practice. The component history records were comprehensive and deferred periodic tests were appropriately evaluated. The inspector concluded the licensee's program was appropriate and no violations of NRC requirements were identified. This item is closed.

The inspector also reviewed a related issue associated with the feeder circuit to Reactor Feedwater Pump '1A'. The circuit consists of two 500 MCM cables per phase and on May 7, 1998, the pump feeder breaker tripped due to a failure of one of the cables. At that time the licensee implemented a temporary modification which disconnected one cable per phase to allow continued operation of the pump without replacing the faulted cable. The licensee initially planned to replace the faulted cable during the current (17R) refueling outage but has since decided to defer the work until a subsequent outage. The licensee revised the temporary modification and associated safety evaluation to address operation of the pump with up to a maximum of 475 amps load with one 500 MCM cable per phase until the 18R refueling outage. Although this pump is not a safety related component, its failure would result in a plant transient that could potentially cause a plant trip which could then challenge safety systems. The inspector reviewed the revised safety evaluation and found that the licensee adequately addressed the condition and the operating procedures are to be revised to ensure the cables are not overloaded during normal plant operation.

E8.8 (Closed) Unresolved Item 50-219/97-06-10: Use of Degraded Voltage Relays Without Harmonic Filters. The licensee evaluated the effects of harmonic frequencies on the degraded voltage relays and concluded that although they are satisfactory for use-as-is, the system would be upgraded during the 18R refueling outage to include harmonic filters. This action is being tracked by the licensee under Licensing Action Request 97093.03 and Electronic Task Tracking System task No. 8346. The inspector found the licensee action to be appropriate and planned actions are being properly tracked. No violations of NRC requirements were identified. This item is closed.

E8.9 (Closed) Unresolved Item 50-219/97-06-11: Adequacy of Maintenance of Circuit Breakers to Ensure Fast Bus Transfer Analysis Remains Valid. The inspector reviewed the issue to evaluate whether there was sufficient maintenance and testing on bus supply circuit breakers to ensure that their operation had not degraded such that a slow bus transfer may have been the cause of a loss of offsite power event.

The inspector reviewed this issue with the system engineer and determined that the circuit breakers have received periodic overhauls that are performed by the circuit breaker manufacturer. Circuit breaker tripping and closing times are measured during the overhaul and a review of timing data indicates that the breaker operating times support the dead bus times that are assumed in the fast bus transfer analysis. The inspector concluded that the licensee control of circuit breaker maintenance and operating times were appropriate and no violation of NRC requirements were identified. This item is closed.

E8.10 (Closed) Violation 50-219/97421-03013: Inadequate Design Control Involving Startup Transformer Voltage Regulators. To prevent recurrence of a condition where the startup transformer voltage regulators cause an undervoltage condition on the 4160 volt buses following the loss of the preferred offsite power source the licensee implemented several actions. These actions included the establishment of an operating voltage band to account for the voltage regulators such that assumptions of the degraded grid analysis remains valid. The frequency of substation tour performed by the plant operators was increased from weekly to daily and additional parameters were monitored to improve the ability to detect abnormalities which could cause complications during a transient. The licensee had the degraded grid analysis reviewed by a consultant to obtain an independent assessment of the adequacy of the analysis. The licensee also plans to update the degraded grid analysis to include the induction regulators.

The inspector reviewed the licensee actions including a review of the control room log for monitoring regulator output voltage, the revised substation tour sheet and the independent consultant evaluation of the degraded grid analysis. The inspector also discussed the issue with control room operators and observed current voltage levels as indicated on the control panel. The inspector concluded the licensee has taken appropriate actions to address this issue. This item is closed.

E8.11 (Closed) Violation 50-219/97421-04014: Control Rod Drive (CRD) Pump Operability. During a transient both CRD pumps failed to restart following energization of the busses by the EDGs. The CRD pump circuit breakers initially closed but then tripped because the EDG output voltage had not yet increased to the pump undervoltage setpoint. The anti-pumping feature of the breaker prevented the breaker from reclosing. The licensee implemented a modification to install a sixty second time delay for pump start following a loss of offsite power. This time delay already existed for the sequenced loading following a loss of coolant accident. The licensee also reviewed the remaining EDG loading circuits and did not identify any additional problems. The inspector reviewed this issue and the circuit modification with the system engineer and found the corrective action to be appropriate. This item is closed.

E8.12 (Closed) Licensee Event Report 98-11: Three Small Bore Piping Lines did not meet Design Bases Seismic and/or Thermal Allowables. The inspectors discussed the event in Section E2.2 of this inspection report. Inspector in-office review of LER 98-11 determined that the report met the 10 CFR 50.73 reporting requirements. This LER is closed.

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 General Observations (71750)

During radiologically controlled area (RCA) tours the inspectors observed that technicians posted proper warning signs, personnel wore appropriate dosimetry, personnel conducted adequate radiological monitoring of personnel and materials leaving the RCA, and technicians maintained monitoring instrumentation functional and in calibration. Technicians maintained radiation work permits (RWPs) and survey status boards up-to-date and accurate, monitoring instruments were functional and in calibration. They observed activities in the RCA and verified that personnel complied with the requirements of applicable RWPs, and that workers remained aware of the radiological conditions in the area.

R1.2 Refueling Outage Radiological Controls

a. Inspection Scope (83750)

The inspector conducted an outage health physics inspection during refueling outage 17R. Areas of inspection focus were based on the following regulatory requirements from 10 CFR Part 20:

20.1101	Radiation protection program
20.1601	Control of access to high radiation areas
20.1602	Control of access to very high radiation areas
20.1902	Posting requirements
20.1904	Labeling containers
20.2103	Records of surveys

The inspection was conducted via direct observation of in-process work in the RCA, review of pertinent documents including surveys, RWPs and as low as is reasonably achievable (ALARA) reviews, and discussions with cognizant personnel.

b. Observations and Findings

For the 17R outage, an occupational exposure goal of 275 person-rem was established as part of the station 1998 goal of 325 person-rem. Through the first 14 days of the outage, personnel exposures were tracking lower than expected, at only 82% of the expected exposure based on the scope of work performed. The 275 person-rem is the lowest outage exposure goal ever established by the licensee. General area dose rates, especially in the drywell, were higher than at similarly designed facilities, due to the high source term in piping and components. Significant radiological work considered in the exposure goal included:
(1) repair/replacement of two reactor recirculation pump seals;
(2) repair/replacement of the safety relief valves; (3) repair of one inboard main steam isolation valve; (4) repair/replacement of three reactor water clean-up (RWCU) system valves located in the valve nest; (5) torus strainer repairs; and, (6) repair of the condenser demineralizer under vessel drains.

During the second week of the outage, fuel movement was ongoing, resulting in additional radiological control challenges, especially for work in the drywell. In order to minimize the consequences of a fuel drop accident, access above the 46' elevation within the drywell was restricted. All personnel entering the drywell during fuel movement were briefed on licensee procedure 6630-ADM-4110.13, Revision 10, *Drywell Occupancy and Evacuation During Fuel Handling Operation*. Exposure controls were generally effective, as installed shielding reduced the effective dose rates in the work areas reviewed, and total exposures were tracking at or below projections. During initial tours of the drywell, the inspector identified numerous instances of poor radiological housekeeping and potential foreign material exclusion (FME) concerns, especially on the 23' elevation. Numerous small tools and parts were left uncontained on the open deck grating in the drywell above the downcomers. Extensive clean-up efforts in high dose rate areas would be needed to close-out the drywell at the end of the outage if conditions were not addressed. The licensee, upon notification of these concerns, undertook an immediate general clean-up of this elevation. Senior station management held meetings with supervisors to emphasize the need for appropriate foreign material controls and radiological housekeeping practices. Additional tours conducted later in the inspection revealed significant housekeeping improvements on the 23' elevation. FME control on the 13' elevation (adjacent to the downcomers) and housekeeping in the condenser bay remained to be improved.

Tours of work in the general areas of the reactor building included work on the RWCU valve nest located beneath a shield plug on the 75' elevation, and work along the east hallway on the 51' elevation. Significant contamination was found inside the three RWCU valves, necessitating an aggressive decontamination regime. Materials removed during this evolution had dose rates upwards of 100 millirem per hour. The licensee effectively removed these materials out of the reactor building and to a shielded storage liner located outside in the yard area while minimizing personnel exposures. The tour of the work area on the 51' elevation indicated poor radiological housekeeping, similar to that found inside the drywell, in high dose rate areas.

Work inside the torus involved installation of new core spray suction strainers. Due in large part to previous torus work undertaken during other outages, general area dose rates of less than 2 millirem per hour were typically encountered during this activity. As a result, with the work near completion, total personnel exposures were less than 3 person-rem. FME practices in this area were effective.

A significant outage project involved replacement of the condensate demineralizer underdrains. Appropriate radiological work controls were in-place to minimize radiation exposures and to control the spread of contamination.

Work on the refuel floor was primarily focused on fuel movement and in-core inspections. Appropriate hot particle controls were in place whenever equipment was removed from the cavity. A number of additional work activities involving equipment inspection was also on-going on the refueling floor. Personnel traffic and job control in the area was well managed by the refueling manager and the radiation protection technicians.

A review of the program and process for entry into elevated areas of the RCA was also examined. In general, limitations are placed on working above seven feet off the floor level in the RCA. Postings instruct workers to contact radiation protection prior to accessing these elevated areas. The following procedures were reviewed as part of this inspection:

6630-ADM-4000.11, Rev 1, *Rules for Conduct of Radiological Work*

6630-ADM-4110.01, Rev 4, *Establishing and Posting Areas in the Radiologically Controlled Areas (RCA)*

6630-ADM-4200.01, Rev 1, *Radiological Surveys*

For example, entry into the turbine building operating floor crane requires entry to an area above seven feet. When accessing this equipment, contact is made with the Group Radiological Controls Supervisor who reviews available survey data for the area, including radiation dose rates and contamination levels, before allowing access. Records reviewed indicated that entries to the crane have been generally limited to periods of reactor shutdown, so that the radiation levels on the turbine building operating floor are less than 2 millirem per hour, and that contamination levels in this area, when no systems are breached, are consistently less than 1000 disintegrations per minute per 100 square centimeters.

c. Conclusions

Radiation Protection established an acceptable radiation protection program to support refueling outage 17R as evidenced by appropriate briefings, postings and ALARA controls in the reactor, drywell, and turbine buildings. The outage work force demonstrated a weakness in the areas of foreign material exclusion and radiological housekeeping. A number of small loose articles were found near the downcomers in the drywell, while poor housekeeping practices in high dose rate areas could lead to additional occupational exposures during outage close-out.

R5.1 Northeast Utilities Examination Bank Compromise

a. Inspection Scope (71750)

The inspectors evaluated the actions concerning a potential Northeast utilities examination bank security compromise.

b. Observations and Findings

On September 14, an instructor noticed a contract employee preparing for a radiation control technician screening examination. The instructor observed what appeared to be a copy of a Northeast Utilities examination. Oyster Creek uses the examination bank provided by Northeast Utilities to other utilities to aid in the screening of radiation control technicians. Radiation protection initiated CAP 1189 to address the potential problem. Further review of the material determined it was an active examination. Oyster Creek alerted Northeast Utilities of the potential examination bank security compromise. Radiation control management and training staff conducted an investigation to assess the integrity of the Oyster Creek radiation control technician training program. Oyster Creek uses a separate examination bank to certify its radiation control technicians and determined that the Oyster Creek program was not compromised.

c. Conclusion

Radiation control management and training staff acted accordingly to assess the integrity of the Oyster Creek radiation control technician training program and to alert other utilities of a potential Northeast Utilities examination bank security compromise.

R8 Miscellaneous RP&C Issues (92904)

R8.1 (Closed) Violation 50-219/97-11-04: Failure to Perform an Annual Land-use Survey. During a review of the land-use survey, it was determined that the location of all gardens greater than 50 square meters producing broadleaf vegetation out to a distance of 5km (3 miles) for each radial sector was not performed according to

Section 4.5.1.8 of the Offsite Dose Calculation Manual (ODCM). In response to NOV 50-219/97-11-04, dated April 10, 1998, GPUN attributed the violation to unclear written guidance in Section 4.5.1.8 in that a incorrectly placed footnote misrepresented the intent and meaning of the regulation. To correct the violation and to preclude future errors, the licensee conducted a safety analysis and revised the ODCM to clarify the regulatory guidance. The inspector reviewed the corrective actions and determined that the actions were reasonable. The violation is closed.

S1 Conduct of Security and Safeguards Activities

S1.1 General Observations (71750)

During routine tours, the inspectors noted that security controlled vital and protected area access in accordance with the security plan, properly manned security posts, locked or guarded protected area gates, and maintained isolation zones free of obstructions. The inspectors reviewed the security events log and noted that security force members seldomly had to compensate for degraded detection and monitoring equipment. Security maintenance technicians promptly addressed several minor equipment deficiencies and contributed to good material condition and high reliability of the security equipment.

S4 Security and Safeguards Staff Knowledge and Performance

S4.1 Vehicular Control in the Protected Area

a. Inspection Scope (71750)

The inspectors observed security force activities and assessed security's vehicle control.

b. Observations and Findings

At approximately 10:30 a.m. on October 14, 1998, the inspector observed a commercial trash truck approach the exit gate from inside the protected area. The driver left the unlocked running truck and stepped into the access building. The security escort followed close behind the driver, leaving the running truck unattended. The inspector recognized the abnormal condition, informed the Secondary Alarm Station (SAS) operator of the condition, and observed the running vehicle until a security escort re-appeared. The SAS operator notified the desk guard, who directed the security escort to return to his escort duties. The inspector observed that during the limited time (approximately five minutes) that the condition existed, no one attempted to use the truck for an unauthorized purpose. The condition appeared to be an isolated instance.

On October 15, the inspector discussed the occurrence with security management. Security management stated that the involved security force members did not document the occurrence on a Security Incident Report, did not initiate a CAP (corrective actions), and did not inform security management as expected.

Following this discussion, security management promptly documented the occurrence in CAP 1998-1459, ensured security force members properly completed an Incident Report, and counseled involved personnel.

Security attributed the failure to properly escort the vehicle to the guard's personnel error. The guard was a temporary; hired to support the 17R outage. Security management did not task the temporary guards with security post assignments or response duties and did not arm them. On October 19, the Security Manager issued a memo directing shift supervisors to ensure all temporary guards were aware of their assigned responsibilities and to provide additional training as needed. In addition, security management noted that an emergency medical condition caused the on-shift security supervisors' removal from site on the afternoon of October 14 and contributed to the inadequate documentation and reporting of the occurrence.

Security Procedure OSEC-IMP-1530.01, Revision 10, *Personnel Identification and Site Access Control*, Step 4.8.4. requires that when any vehicle is left unattended within the protected area the vehicle must be ignition locked and the keys removed. In addition, for non licensee-designated vehicles, left unattended within the protected area, the keys shall be in the possession of the Security Department. The guard left the non licensee-designated vehicle, unlocked and running, unattended in the protected area. This is a violation of procedure OSEC-IMP-1530.01. **(VIO 50-219/98-09-01)**

c. <u>Conclusions</u>

A security guard did not properly conduct vehicle escort duties as he left a running vehicle unattended in the protected area. Security force members did not properly document the occurrence. Once notified by the inspector, security management took prompt and appropriate corrective actions.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspectors provided a verbal summary of preliminary findings to senior licensee management on September 18, October 9, and October 30, 1998. During the inspection period, inspectors periodically discussed preliminary findings with licensee management. Inspectors did not provide any written inspection material to the licensee. The licensee did not indicate that any of the information presented at the exit meeting was proprietary.

ATTACHMENT 1
PARTIAL LIST OF PERSONS CONTACTED

Licensee (in alphabetical order)

R. Beck, Technical Training Instructor
G. Busch, Manager, Nuclear Safety & Licensing
W. Cooper, Radiological Engineering Manager
B. DeMerchant, Licensing Engineer
S. Levin, Director, Operations and Maintenance
D. McMillan, Director, Equipment Reliability
K. Mulligan, Plant Operations Director
J. Perry, Plant Maintenance Director
W. Quinlan, Station Services Manager
M. Roche, Director, Oyster Creek
D. Slear, Director, Configuration Control
M. Slobedien, Health Physics Manager
R. Tilton, Manager, Assessment
K. Wol Radiological Controls Field Operations Manager

NRC (in alphabetical order)

J. Furia, Senior Radiation Specialist
T. Hipschman, Resident Inspector
S. Pindale, Senior Resident Inspector, Hope Creek
L. Scholl, Senior Reactor Engineer
J. Schoppy, Senior Resident Inspector

ATTACHMENT 2
INSPECTION PROCEDURES USED

Procedure No.	Title
37551	Onsite Engineering
60710	Refueling Activities
61726	Surveillance Observation
62707	Maintenance Observation
71707	Plant Operations
71750	Plant Support
83750	Occupational Radiation Exposure
90712	Inoffice Review of Written Reports of Power Reactor Facilities
92902	Followup - Maintenance
92903	Followup - Engineering
92904	Followup - Plant Support

ATTACHMENT 3
ITEMS OPENED AND CLOSED

Opened\Closed

Number	Type	Description
50-219/98-09-01	VIO	Failure to properly escort vehicle within the protected area. (Section S4.1)

Closed

Number	Type	Description
50-219/95-05-01	URI	Emergency diesel generator reliability program. (Section E8.2)
50-219/95-22-01	IFI	NSA monitoring of the combustion turbine reliability program. (Section E8.3)
50-219/96-06-02	URI	Use of Quartz for Fine Aggregate in HSM Roof. (Section E8.1)
50-219/96-07-03	URI	Installation of AC fuses in DC powered control circuits. (Section E8.4)
50-219/96-09-03	URI	Control of cable. (Section E8.5)
50-219/96-09-04	URI	Voltage drop calculation. (Section E8.6)
50-219/96-09-05	URI	Medium voltage cable testing program. (Section E8.7)
50-219/97-03-01b	VIO	Failure to determine an appropriate isolation boundary. (Section M8.1)
50-219/97-06-10	URI	Use of degraded voltage relays without harmonic filter. (Section E8.8)
50-219/97-06-11	URI	Maintenance of circuit breakers used in fast transfer. (Section E8.9)
50-219/97-11-04	VIO	Failure to Perform an Annual Land-use Survey. (Section R8.1)
50-219/97421-03013	VIO	Inadequate design control.(Section E8.10)
50-219/97421-04014	VIO	Control rod drive pump operability (Section E8.11)
50-219/98-11	LER	Three small bore piping lines did not meet design bases seismic and/or thermal allowables. (Section E8.12)
50-219/98-12	LER	Actuation of an engineered safety feature by a test instrument due to an inadequate procedure. (Section M8.2)

ATTACHMENT 4
LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CT	Combustion Turbine
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
DSC	Dry-shielded Storage Canisters
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EMRV	Electromatic Relief Valve
FME	Foreign Material Exclusion
GE	General Electric
GPUN	General Public Utilities (GPU) Nuclear
GSS	Group Shift Supervisor
HSM	Horizontal Storage Module
I&C	Instrumentation and Control
INPO	Institute of Nuclear Power Operations
IPE	Individual Plant Examination
IST	In-Service Test
JO	Job Order
LCO	Limiting Conditions for Operation
LOCA	Loss of Coolant Accident
MPFF	Maintenance Preventable Functional Failure
NDE	Non-Destructive Examination
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSA	Nuclear Safety Assessment
NSR	Nuclear Safety-Related
OCNGS	Oyster Creek Nuclear Generating Station
ODCM	Offsite Dose Calculation Manual
PDR	Public Document Room
PMT	Post Maintenance Test
QA	Quality Assurance
QV	Quality Verification
RCA	Radiologically Controlled Area
RWCU	Reactor Water Clean-Up
RWP	Radiation Work Permit
RP&C	Radiological Protection and Chemistry
SAS	Secondary Alarm Station
SDC	Shutdown Cooling
SRO	Senior Reactor Operators
TN-West	Transnuclear-West, Inc.
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
USQ	Unreviewed Safety Question