U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS.: 50-272/94-04
50-311/94-04
50-354/94-02

LICENSE NOS.: DPR-70
DPR-75
NPF-57

LICENSEE: Public Service Electric & Gas Company
80 Park Plaza - 17C
Newark, New Jersey

FACILITY: Salem 1 & 2 and Hope Creek Generating Stations

INSPECTION AT: Hancocks Bridge, New Jersey

INSPECTION DATES: February 14 - 18, 1994

INSPECTORS: R. Bhatia, Reactor Engineer, EB, DRS
R. Skokowski, Reactor Engineer, EB, DRS

Leonard S. Cheung — 3/28/94
Leonard S. Cheung, Sr. Reactor Engineer — Date
Electrical Section, EB, DRS

APPROVED BY: for — 4/6/94
William H. Ruland, Chief — Date
Electrical Section, EB, DRS

Areas Inspected: Announced inspection to review: 1) Public Service Electric & Gas Company (PSE&G) environmental qualification (EQ) programs for Salem and Hope Creek plants, and 2) the status of two NRC unresolved items.

Results: PSE&G had an adequate program in place to assure that the EQ master lists were maintained. PSE&G's quality assurance (QA) audits of the EQ program were comprehensive with significant findings. Engineering's responses to the QA audit findings were generally complete with the exception of a few items as discussed in Sections 3.3 and 3.4 of this report. An NRC unresolved item regarding the EQ of the power range neutron detectors was identified, and two previously identified NRC unresolved items were reviewed and remained open.

DETAILS

1.0 PURPOSE

The purpose of this inspection was to review the Public Service Electric and Gas Company (PSE&G) environmental qualification (EQ) program for Salem Units 1 and 2 and Hope Creek plants. The inspection focused on the program's design bases criteria for a harsh environment, EQ master lists, environmental design conditions, and quality assurance (QA) audits of the EQ program. This inspection also reviewed the status of two NRC unresolved items regarding cable separation and the environmental qualification of certain safety-related valve position switches.

2.0 REVIEW OF ENVIRONMENTAL QUALIFICATION PROGRAM (40703)

The current Salem Station licensing bases for the EQ program consist of the following:

- Compliance with 10 CFR 50.49;
- NUREG-0588 Category I;
- Original equipment qualified to IEEE Standard 323-1971; and
- New/replacement equipment qualified to IEEE Standard 323-1974.

The Salem Station has two exceptions from the NUREG 0588, Category I, as documented in the Salem Equipment Qualification Review Report (EQRR), Revision 8.

The licensing bases for the Hope Creek EQ program are the same as those of the Salem Station with the exception that the original equipment at Hope Creek were also qualified to IEEE Standard 323-1974. Additionally, Hope Creek is committed to Section 4.4.1, "Aging" of IEEE 627-1980, "Design Qualification of Safety Systems Equipment Used in Nuclear Power Plants," for the qualification of active safety-related mechanical equipment located in harsh environments.

The review of the EQ program included: 1) the EQ program procedures; 2) the Equipment Qualification Master List (EQML); 3) the Environmental Design Criteria (EDC); and 4) the mechanical equipment qualification program for Hope Creek.

2.1 Environmental Qualification Program Procedures

The inspector reviewed the procedures associated with the EQ program to determine the adequacy of the program to control equipment environmental qualification. A list of the procedures reviewed is provided in Attachment 2 of this report. The Program Analysis Group (PAG) is primarily responsible for implementing the EQ program. The PAG is responsible for: 1) ensuring that the EQ program conforms to all applicable regulatory

requirements; 2) ensuring that the review of change packages (CPs) are preformed in accordance with the EQ program; and 3) ensuring that the EDC for Hope Creek, and the environmental conditions of the EQRR for Salem, are updated and issued in accordance with the applicable procedures. The PAG also interfaces with the other organizations to ensure that the EQ program is properly implemented. The inspection determined that the EQ program procedures were adequate to control equipment environmental qualification.

2.2 Equipment Qualification Master List

The EQ program applies to electrical equipment important to safety as described in 10 CFR 50.49. The Salem and Hope Creek equipment classification procedures describe the classifications of safety-related electrical equipment as either harsh, mild, or exempt.

The criteria, used by PSE&G, for determining if a safety-related electrical component located in a harsh environment can be exempt from harsh environment qualification requirements is based on the guidance provided in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1. The EQML is a control document listing all electrical equipment, which is important to safety and located in a harsh environment.

A 1991 QA audit of the EQ Program identified several discrepancies between the Salem Station EQML and the Managed Maintenance Information System (MMIS) with respect to the equipment required to be environmentally qualified for a harsh environment. As a result, an action plan was developed to address these discrepancies. The action plan included the following:

- Performing walkdowns to verify the equipment physical location and correct documentation as required;
- Preparing engineering evaluations justifying the exemption of equipment from the EQ program as required;
- Revising the equipment classification procedures and MMIS data base to include classification for Class 1E equipment located in harsh environments but exempt from the EQ program; and
- Developing and updating EQ documentation to reflect the results of the action plan.

The results of the walkdowns were documented in Engineering Evaluation S-C-ZZ-SEE-0801-0. The inspector reviewed this engineering evaluation and concluded that a majority of the discrepancies were due to errors within the MMIS. These errors included improperly identifying the equipment location, improperly classifying mechanical equipment as electrical equipment, and failure to update the MMIS for equipment that was removed through the modification process. The inspector determined that the resolution of the discrepancies

identified by the walkdown were appropriate. Additionally, the PAG has performed subsequent comparisons between the Salem EQML and the MMIS to ensure no further discrepancies exist. The PAG engineers stated that periodic comparisons between the EQML and the MMIS will be performed to ensure consistency between these documents.

The inspector examined a sample of the EQML, MMIS, environmental design criteria for both stations and the system component evaluation work (SCEW) sheets for Salem Units 1 and 2 and the equipment evaluation summary (EES) sheets for Hope Creek. The information contained within these documents was compared for a sample of systems and components, and no discrepancies were identified. Based on this review, the inspector concluded that the EQML, MMIS, environmental design criteria, the SCEW and EES sheets were adequate.

2.3 Environmental Design Criteria

Hope Creek Generating Station

The environmental conditions section of the Hope Creek Generating Station Environmental Qualification Summary Report (EQSR) is maintained for history only and has been replaced by the EDC, Calculation D7.5, "Hope Creek Generating Station Environmental Design Criteria," Revision 13. The EDC was reviewed by the inspector and found to provide adequate detail to allow equipment installed to be appropriately qualified for their environmental conditions.

Salem Generating Station

The Salem EQRR, Section III, entitled "Environmental Zones and Conditions," contained a table defining the environmental conditions (temperature, pressure, radiation, etc.) for each zone. In response to a quality assurance EQ audit finding, an EDC is being developed for the Salem Station. The engineering staff stated that the new EDC will provide more detailed information than is currently provided in the EQRR.

The inspector's review of the environmental conditions section of the EQRR found them to appropriately describe the environmental conditions at Salem with the exception of the concerns described in Sections 3.1 and 3.2 of this report.

2.4 Qualification of Mechanical Equipment

The inspector reviewed the Hope Creek mechanical equipment qualification (MEQ) program. The primary objective of this program is to ensure that active safety-related equipment in harsh environments can perform its safety function(s) with no failure mechanism that could lead to common mode failures under postulated conditions. This mechanical equipment is

listed in the Mechanical Equipment Qualification Master List (MEQML). The inspector reviewed the Hope Creek MEQ Program and found it adequate to meet the commitment stated in the UFSAR Section 3.11.2.6, "Qualification of NSSS and Non-NSSS Safety-Related Mechanical Equipment."

2.5 Conclusion

The review of the Salem and Hope Creek EQ programs determine that the programs were in accordance with 10 CFR 50.49. Based on the samples reviewed, the inspector confirmed that the required equipment is in the EQML. The procedures provided adequate guidelines for evaluating newly installed equipment for environmental qualification.

3.0 QUALITY ASSURANCE AUDIT OF EQ PROGRAM (40703)

The inspectors reviewed three quality assurance EQ program audit reports completed in 1989, 1991, and 1993 (89-101, 91-101, and 93-101). The licensee's QA audit team consisted of several QA auditors and an EQ specialist.

The first audit, 89-101, of the Nuclear Engineering Department (NED) was conducted from December 4, 1989 through January 4, 1990. The QA auditor concluded that the EQ program generally had been adequately implemented. However, significant weaknesses in the Salem EQ documentation was identified. The auditors identified several documentation discrepancies between the EQML, MMIS data base, field directives (FD's) and the nuclear engineering evaluations. As a result, one action request was issued to correct this overall documentation deficiency. The inspectors noted that the licensee had been upgrading the applicable procedures to assure the coordination among various groups to maintain EQ documentation. In addition, the licensee initiated a walkdown effort to reconcile discrepancies between the MMIS data base and other EQ documentation. The corrective actions for this finding had been completed and the action request was closed.

The second EQ audit, 91-101, took place between June 3, and June 27, 1991. The audit determined that the EQ program for the Hope Creek plant in general was adequate. However, the audit report identified concerns in controlling the MMIS data base changes and EQ classification. In addition, the EDC for Hope Creek Generating Station required a revision. The audit team identified several significant technical concerns associated with the Salem EQ program and issued ten QA action requests to NED to resolve these concerns. The Salem concerns included issues such as incorrect EQ classification, inconsistent EQ equipment data in the MMIS and the EQML, equipment located in harsh environment was listed as located in mild environment, and electrical equipment important to safety not being included in the EQML. In addition, the justifications for exemption from the EQ program were found to be deficient for several components.

The third EQ audit, 93-101, performed in 1993, consisted of an overall EQ program review and to close out earlier identified action items. The inspector reviewed the audit report and determined that some progress had been made in improving the Salem EQ program. However, several technical concerns identified in 1991 audit report needed additional management attention to expedite resolutions. The Hope Creek EQ program was found to be adequate and the EQ issues identified in the 1993 audit for Salem Units 1 and 2 are discussed in the following sections.

3.1 Inside Containment and Outside Containment Area Temperature Concerns (Salem 1 & 2)

The maximum containment temperature during normal plant operating conditions as specified in the Salem EQRR is 120°F. The Salem UFSAR, Section 3.11.1.3, also states that inside the containment, the normal operating temperature for all protective equipment will be maintained below the 120°F (containment bulk temperature), with the exception of the excore neutron detectors. The requirements stated in both documents were found to be consistent. However, the QA audit team was concerned that the temperature in the pressurizer penthouse area exceeded the 120°F limit. To resolve this issue, the licensee provided temperature monitoring and determined the temperature to be approximately 160°F. Even though the EQ qualification calculations may be conservative for bulk temperature, the EQRR does not clearly address the actual maximum temperature of those components located in the high temperature areas around the pressurizer penthouse. The licensee had evaluated all the components located in those areas and determined that all safety-related equipment would operate under higher temperature conditions.

The inspector noted that the licensee had a temperature monitoring program since 1988 to ensue that the appropriate temperature conditions were used to determine the operability and qualified life of equipment. The temperature monitoring program is still ongoing to determine temperatures in other plant areas both inside and outside the reactor containment. The licensee is tracking this as an open item under the nuclear department action tracking system. The inspector concluded that the actions taken by the licensee were appropriate.

3.2 Inside Containment and Outside Containment Area Radiation Concerns (Salem 1 & 2)

The Salem EQRR specified the radiation threshold for EQ requirements to be 1×10^4 rad total integrated dose (TID). The 1993 QA audit report states that certain electronic devices could be damaged when subject to a radiation dose below 1×10^4 rad. In response to this concern, the licensee performed a walkdown and identified two control panels for the Gamma-Metrics neutron detectors (in each Salem unit) that were located in areas that could subject to a TID up to 1×10^4 rad. These panels contain sensitive electronic devices that could be susceptible to radiation damage. The licensee issued in April 1993, a deficiency evaluation form (DEF-93-00120) to address this issue. The panels are located outside the reactor containment near the containment penetrations. During this inspection, the licensee

located a dosage calculation for a nearby reactor vessel level indication system (RVLIS) transmitter, which is about 3 feet closer to the containment wall than the control panels. The calculated TID for 40 years plus 180-day post loss-of-coolant accident (LOCA) condition is about 250 rads. The licensee stated that this radiation dosage would not have adverse affects to the electronic devices. The inspector concluded that current configuration of the Gamma-Metrics neutron detectors panels, with respect to the EQ for radiation dose, was acceptable.

For components located inside the reactor containment, the Salem EQRR specified a maximum TID of 5 x 10^7 rad gamma for equipment and 12 x 10^7 rad gamma for cables. The 1993 QA audit identified that certain areas inside the reactor containment may be subject to radiation dosages much higher than those specified. Typical examples are: 1) the area within 7.5 feet of the steam generator bottom; and 2) the reactor cavity area.

In response to the auditor's concern for the first example, the licensee reviewed the equipment location document and determined that there was no electrical equipment important to safety located within 7.5 feet of the steam generator bottom area. To prevent future qualification problems in this area, the licensee included this issue into their tracking system (item 272/94-04 for both temperature and radiation) that electrical equipment to be located in this area must be qualified to the higher radiation dosage. The licensee stated that this issue would be resolved when the new EDC is finalized.

For the second example, the QA audit identified the following items that were located in the reactor cavity area: Gamma-Metrics neutron detectors and associated connecting cables; Westinghouse power range neutron detectors and connecting cables; and incore thermocouples and connecting cables. These components are subject to much higher radiation dosages than the general containment TID. The EQ of these components are discussed in detail in Sections 3.3 through 3.5 of this inspection report.

3.3 Power Range Neutron Detectors (Salem 1 & 2)

The licensee had requested the NRC exempt the power range neutron detectors from the EQ program. This exemption was accepted by the NRC in a Safety Evaluation Report, dated June 8, 1984. The justification for this exception indicated that during an accident, the reactor will trip promptly by other safety equipment (containment pressure and steam flow).

The 1993 QA audit identified a concern related to the potential effect of the power range nuclear instrumentation on the rod control system during a small steam line break. During this event, the reactor may take approximately two minutes to trip after the accident. During this time, the power range detectors and associated cables are required to remain functional to ensure no outward rod motion. This event was not covered in the original exemption justification. The engineering staff stated that additional documented justification is required for this event. This item is unresolved pending NRC review of the resolution of this matter (NRC Unresolved Item 50-272/94-04-01 and 50-311/94-04-01).

The licensee determined that there was no immediate safety concern because the detectors and associated cables were tested for 16 hours at 300°F. In addition, the field cables used to connect the neutron detectors assembly to the containment penetration were mineral insulated cables, which was qualified for the containment environment. The inspector reviewed the applicable arrangement drawing of Salem Unit 2 (PSE&G Drawing No. 22038A8989) and found that the field cables were located outside of the reactor cavity high radiation area. Based on the above, the inspector concluded the immediate safety concern was adequately addressed by the licensee.

3.4 Environmental Qualification of Gamma-Metrics Neutron Detectors (Salem 1 & 2)

The Gamma-Metrics neutron detectors were provided to satisfy Regulatory Guide (RG) 1.97 commitments. These are classified as Category 1 equipment. The inspector reviewed a Stone & Webster letter, #SSP-1228, dated June 13, 1985, which provided radiation levels at different points of the reactor cavity. At the core mid-plane, the 40-year integrated dose is: gamma dose = 2.5×10^9 rad; total neutron fluence = 1×10^{19} nvt (neutron-velocity-time). The letter also indicated that for locations opposite to the core flat, the value is about 50% lower. The two neutron detectors are located opposite to the core flat.

A review of the system component evaluation work (SCEW) sheet, dated December 21, 1993, for this equipment indicated that the gamma dose of 2.5×10^9 rad and the total neutron fluence of 10^{19} nvt were specified. The EQ engineer stated that the specified value in the SCEW sheets did not include the 50% reduction in dose and neutron fluence for conservatism. The EQ engineer also stated that originally the general inside-containment dose of 5×10^7 was specified for these detectors, which was much lower than the current specified value. This was found to be incorrect as a result of the 1993 QA audit findings. The SCEW sheets were corrected to incorporate the high dose values. The neutron detector assemblies were tested and qualified to 3.2×10^9 rad and 1.0×10^{21} nvt; therefore, the change in the dose requirement did not affect the qualification status of the detectors.

The QA audit also identified that the 40-year integrated dose did not include the accident dose which could occur at the end of the 40-year period. Under this assumption, the neutron detectors are required to function 120 more days. This accident dose must be added to the 40-year integrated dose. PSE&G performed a calculation for the accident dose. The calculation indicated that the dosage increase due to the accident was 2.6×10^7 rad for gamma, and the neutron fluence was negligible. This additional dosage did not affect the qualification status of the neutron detectors. The inspector verified that this calculation result was incorporated into the EQ file.

The inspector concluded that PSE&G's resolution to this QA finding was adequate.

3.5 Environmental Qualification of Incore Thermocouples (Salem 1 & 2)

The incore thermocouples are used to provide core exit temperature indications for RG 1.97. RG 1.97, Revision 2, classified this parameter as a Category 1 item, requiring EQ for its equipment. The incore thermocouples, which are inserted into the bottom of the reactor vessel are subject to the radiation levels in the reactor cavity.

The incore thermocouples are type K thermocouples manufactured by Reuter-Stokes Company. Qualification of this equipment was documented in Salem EQ file #SC-MS-EQ48-001. PSE&G described the qualification method used in their 1984, RG 1.97, submittal to the NRC. The NRC reviewed and accepted the qualification of these components.

The QA audit report stated a concern that if the field cables and connectors for the thermocouples were located inside the reactor cavity, the specified radiation level for the environmental qualification may be insufficient. The inspector reviewed the equipment drawing and found that the connecting cables were located outside the reactor cavity. Mineral insulated cables were used connecting the thermocouples to the containment penetrations. Qualification of mineral insulated cables and connectors were documented in Salem EQ file #SC-MS-EQ26-001.

The inspector concluded that resolution for the QA finding was adequate.

3.6 PORV Auxiliary Air Supply Pressure Switches Concerns (Salem 1 & 2)

The QA auditors questioned the need for EQ of the power operated relief valves (PORV's) auxiliary air supply pressure switches (PD9859 and PD9860). These switches are United Electric Controls Company, Model No. 7-77, switches and associated solenoid operated valves (SV1198 and SV1199) are ASCO Model No. NP8316E35E valves. The existing MMIS data classified these devices as Class 1E, safety-related and seismically qualified. The PORV valves are air-activated and air is normally supplied by the plant instrument air system. In case of a loss of plant instrument air, the control air to the PORV is supplied from backup air accumulators. The function of the pressure switch (PD9859 &PD9860) is to connect the air accumulators to the air supply headers upon loss of pressure in the normal air supply headers due to containment isolation.

During 1991 EQ audit, the auditors determined that PORV's auxiliary air supply pressure switches were identified in the MMIS database as Class 1E, located and designated in harsh environments, but were not included in the EQ program. Since no technical evaluation existed to demonstrate the exemption claimed by engineering, an action request was generated to resolve this concern. The review of the PSE&G documentation indicated that the exemption of these devices from the EQ requirement was accepted by the NRC in a letter issued on June 8, 1984. The QA auditors believed that these pressure switches are required to perform safety functions, the QA auditors found this condition to have insufficient basis for exemption. Again, during the 1993 EQ audit, the QA auditors stated that the PORV's

are relied upon to perform a safety-related function for the mitigation of a steam generator tube rupture (SGTR) event and for low temperature overpressure protection (LTOP) of the reactor vessel during unit startup and shutdown. The current Salem design includes two PORV's, block valves and associated controls for these components. These components are identified as safety-related and are required to perform a safety function for the SGTR event and for LTOP. The auditors also noted that all other components were found to be qualified with the exception of pressure switches. Therefore, the QA auditors considered the basis for exemption inadequate and issued an escalated action request to resolve this concern.

During this inspection, the inspector noted that PSE&G engineering had completed an engineering evaluation (S-C-RC-CEC-0703-1) to justify the exemption of these components. The engineering evaluation indicated that the change of the environmental conditions (temperature, pressure, and radiation) associated with these two events (STGR and LTOP) were insignificant. Therefore, the EQ of the pressure switches was not required.

In addition, the licensee determined the pressure switches to be rugged, vibration/shock resistant, and designed for severe environment. The manufacturer's documentation indicated that these switches were designed for 160°F operation. The inspector reviewed the licensee's response to this issue and concluded that the evaluation was acceptable.

4.0 STATUS OF PREVIOUSLY IDENTIFIED INSPECTION ITEMS (FOR SALEM UNITS ONLY) (40703)

4.1 (Open) Unresolved Item (50-272/89-13-07 and 50-311/89-12-07)

Unresolved Item (50-272/89-13-07 and 50-311/89-12-07) pertained to inadequate cable separations and electrical isolation for various RG 1.97 instrumentation. This item was updated during the October 1992 inspection (Inspection No. 92-06). This unresolved item consists of six parts as follows:

a. Containment isolation valve positions indications cables for the nitrogen supply lines to the accumulators, need to be separated from non-1E circuitry.

b. The plant computer was used for Class 1E circuitry without being qualified.

c. Cable separation issue for low voltage and low energy cables for RG 1.97 instrumentation application.

d. Four circuit breakers were used for RG 1.97 Category 1 instrumentation and non Class 1E circuits.

e. RG 1.97, Category 1 indicator wiring and non-category 1 wiring from the same power source. Separation for this wiring is required.

f. At the time of the October 1992 inspection, no incore neutron monitor recorder was provided. PSE&G was considering either upgrading a non-1E recorder to be connected to a Category 1 instrument loop, or request NRR for an exemption.

The status of PSE&G's corrective actions for these six parts is as follows:

a. PSE&G issued two engineering change packages to separate the Class 1E cable (for containment isolation valve indications) from the non Class 1E cable. These two design changes (1EC-3225 for Salem Unit 1 and 2EC-3196 for Unit 2) were completed during the last refueling outages (January 1994 for Unit 1 and June 1993 for Unit 2). Therefore, this part is closed.

b. PSE&G selected to install isolation devices instead of qualifying the plant computer. These isolation devices are to be supplied by NUS Inc. Implementation of these modifications is scheduled for the next refueling outages for both units.

c. Cable separation for low voltage and low energy cables for RG 1.97 the instrumentation application is scheduled to be implemented during the next refueling outages for both units.

d. PSE&G decided to provide separate circuit breakers for Class 1E and non Class 1E circuits. Implementation of this sub item is scheduled for the next refueling outages for both units.

e. Implementation of this part will be the same as part d. above.

f. PSE&G decided to install a two-pen Leeds & Northrup recorder for each unit on Control Room Panel 2RPI for the RG 1.97 nuclear instrumentation Channel C. Implementation of these modifications is also scheduled for the next refueling outages for both units.

This item remains open pending the completion of parts b. through f.

4.2 (Open) Unresolved Item (50-272/91-30-01 and 50-311/91-30-01)

Unresolved Item (50-272/91-30-01 and 50-311/91-30-01) pertaining to the environmental qualification of 28 containment-isolation valve position switches (for each unit). PSE&G reclassified the locations for some of these valve position switches to be non-harsh environment, and provided engineering justification for the rest valve position switches that were located in a harsh environment. The engineering justification was determined to be acceptable. However, these exemptions were not documented in their submittal to the NRC. PSE&G originally scheduled to meet with the NRC to discuss this issue and all other RG 1.97 outstanding items in 1993, but was delayed due to emergent issues that developed in the Salem Station. PSE&G was scheduled to send a letter to the NRC by March 31, 1994,

to discuss Salem's position on the definition of RG 1.97 containment isolation valves. This item remains open pending PSE&G's submittal of a request for an EQ exemption for the valve position switches.

5.0 UNRESOLVED ITEMS

Unresolved items are matters about which additional information is necessary to determine whether they are acceptable, a deviation or a violation. One unresolved item was identified during this inspection and is discussed in Section 3.3 of this report.

6.0 EXIT MEETING

At the conclusion of the inspection on February 18, 1994, the inspector met with PSE&G representatives denoted in Attachment 1. At that time, the inspector summarized the scope and findings of the inspection. PSE&G representatives did not express any disagreement with the inspection findings. Also at the exit meeting, it was established that Mr. Frank Thomson would be the PSE&G technical contact for future NRC discussions regarding the issues covered by this report.

Attachments:
Persons Contacted
Documents Reviewed

ATTACHMENT 1

Persons Contacted

Public Service Electric and Gas Company

*	J. Bailey	Nuclear Engineering Science Manager
*	R. Beckwith	Station Licensing Engineer
*	A. Blum	Principal Engineer, PAG
*	M. Burnstein	Nuclear Electrical Engineering Manager
*	V. Fregonese	Senior Project Engineer, Nuclear Electrical Engineering
*	A. Giardino	QA Programs & Audits Manager
*	M. Gray	Station Licensing Engineer
*	B. Hall	Hope Creek Technical Manager
	L. Hayes	Electrical Engineering
*	L. Illy	Safety Review Engineer
*	C. Lambert	Nuclear Engineering Design Manager
*	E. Lawrence	QA Lead Auditor
*	R. Smith	EQ Engineer, PAG
	C. Stokes	I&C Engineer
*	F. Thomson	Licensing and Regulations Manager
	J. Vender	Senior Engineer, RCM

U.S. Nuclear Regulatory Commission

T. Fish	Resident Inspector
C. Marschall	Senior Resident Inspector
W. Ruland	Electrical Section, Section Chief

* Denotes those present at the exit meeting on April 30, 1993.

ATTACHMENT 2

Procedures Reviewed

Calculation D7.5, Hope Creek Generating Station Environmental Design Criteria, Revision 13.

EQRR-0001, Salem Generating Station Environmental Qualification Review Report, Revision 8.

EQSR-001, Environmental Qualification Summary Report for Hope Creek Generating Station.

H-1C-EQML-0001, Hope Creek Generating Station Equipment Qualification Master List (EQML), Revision 0.

HC.DE-AP.ZZ-0031(Q), Component Functional Classification for Hope Creek Generating Station, Revision 0.

HC.DE-PS.ZZ-0002(Q), Environmental Equipment Qualification Program, Revision 0B.

HC.DE-PS.ZZ-0003(Q), Mechanical Equipment Qualification Program, Revision 0.

NC.NA-AP.ZZ-0062(Q), Environmental Qualification Program, Revision 1.

ND.DE-TS.ZZ-3054(Q), Hope Creek Generating Station Environmental Design Standard, Revision 0.

S-C-EQML-0001, Salem Generating Station Equipment Qualification Master List (EQML), Revision 1.

S-C-ZZ-SEE-0801-0, Engineering Evaluation to Document the Walkdowns and Engineering Evaluations Performed to Resolve QA Action Request M29-91-048-1M, Task 9, Revision 0.

SC.DE-AP.ZZ-0019(Q), Design Classification of Structure, Systems, Components for Salem Generating Station, Revision 3.