U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos. 50-387/90-17
50-388/90-17

License No. NPF-14
NPF-22

Licensee: Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Facility Name: Susquehanna Steam Electric Station

Inspection At: Corporate Office, Allentown, Pennsylvania
Susquehanna Steam Electric Station Units 1 and 2
Berwick, Pennsylvania

Inspection Conducted: August 13, 1990 - September 7, 1990

Inspectors: _____ 9/13/90
G. S. Barber, Sr. Resident Inspector, 2A, Division of Reactor Projects — date

_____ 9/13/90
R. J. Paolino, Sr. Reactor Engineer, Plant Systems Section, EB, DRS — date

Approved by: _____ 9/13/90
C. J. Anderson, Chief, Plant Systems Section, Engineering Branch, DRS — date

Inspection Summary:

Areas Inspected: A special announced inspection was conducted to review the implementation of certain aspects of the licensee's environmental qualification (EQ) program. Specifically, the qualification of polyurethane seals in ITT NH90 dampers used in the SGTS and DX Systems and the disposition Limitorque motor operated valve EQ discrepancies were reviewed.

Results: Three potential violations were identified. One potential violation involved failure of the licensee to follow established procedures for identifying nonconforming conditions and completing the required EDR and NCR forms in a timely manner. The second potential violation involves operating with components not environmentally qualified to operate in the environment in which they must function. The third potential violation involves the lack of prompt corrective actions related to Nonconformance Reports for numerous limitorque motor actuators with suspect harsh environmental qualification.

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DETAILS

1.0 Persons Contacted

1.1 Pennsylvania Power Light Company

*A. P. Derkacs, Senior Project Engineer
*T. A. Gorman, Supervising Engineer
*J. M. Kenny, Licensing Group Supervisor
*G. J. Kuczynnski, Technical Supervisor
A. M. Male, Manager, NPE
*H. G. Stanley, Superintendent of Plant
W. W. Williams, Licensing Engineer
C. A. Myers, Manager, Nuclear Projects

1.2 U.S. Nuclear Regulatory Commission

C. J. Anderson, Section Chief, Plant Systems Section, DRS/EB
*G. S. Barber, Senior Resident Inspector
W. R. Butler, Project Director, NRR
J. P. Durr, Chief, Engineering Branch, DRS
M. W. Hodges, Director, DRS
*R. J. Paolino, Senior Reactor Engineer, PSS/EB/DRS
J. P. Stair, Resident Inspector
P. D. Swetland, Chief, Reactor Projects Section 2A/DRP
J. T. Wiggins, Deputy Director, DRP

* Present at exit meeting.

2.0 Introduction

The NRC conducted a special inspection to review the acceptability of the environmental qualification (EQ) of certain damper seals and O-rings. In addition, long-standing concerns regarding the adequacy of the licensee's EQ program were also reviewed. Abbreviations are used throughout the text. A listing of these abbreviations is provided in Attachment 1.

The inspection also included a review of the licensee's followup activities regarding elevated reactor building temperature harsh environmental qualification deficiencies identified by the licensee in October 1988 and the licensee's activities regarding Nonconformance Reports involving deficiencies in the harsh environmental qualification of certain electrical equipment.

3.0 Overview

The licensee contacted the Senior Resident Inspector on July 23, 1990 to report that polyurethane damper actuator seals and BUNA-N O-rings for Division 1 SGTS and DX dampers were being replaced. The licensee stated that they had ongoing concerns regarding the continued acceptability of these seals and O-rings in meeting their EQ program. The seals had been replaced as a prudent measure, in parallel with an ongoing EQ evaluation being conducted to evaluate their acceptability. On July 24, 1990 after this evaluation, the licensee concluded that the remaining Division II seals and O-rings were inoperable. The appropriate Technical Specification Limiting Condition of Operation was entered and the licensee completed replacing the unqualified seals and O-rings on July 25. The deficiency was reported per 10CFR 50.72 on July 24, 1990.

The specific deficiency concerned the use of polyurethane (poly) seals and BUNA-N O-rings in ITT NH90 series dampers for the SGTS and DX systems. There were four dampers affected, two for each system. The polyurethane seals were originally environmentally qualified in 1976 with followup testing conducted in 1982 and 1983. This EQ testing provided the basis for the licensee's qualification of the poly seals and BUNA-N O-rings.

4.0 Background

4.1 ITT NH90 Series Damper Seals

In late 1986, the licensee began transferring data from the EQ binders to the Susquehanna Equipment Inventory System (SEIS) Computerized Listing. The SEIS was to contain a computerized version of the EQ master list. The EQ master list's incorporation into the SEIS would allow rapid computerized searches and sorts of the EQ database. The SEIS listing incorporated both the maximum expected temperatures in rooms containing EQ equipment and the temperature at which the qualification tests were run. This SEIS listing was developed in late 1986. During the same period, the plant contacted Nuclear Plant Engineering (NPE) to request an extension on the qualified life for the poly damper seals to reduce their replacement interval. Nuclear Plant Engineering reviewed the EQ binder and noted that the original qualification tests were performed at 130 degrees F. This testing resulted in a short life that could be extended to approximately 10.9 years if the qualification temperature were lowered to 104 degrees F. The plant staff found this new temperature acceptable. As a result, NPE changed the EQ binder to reflect the extended new EQ life at 104 degrees F. However, they neglected to change the maximum qualified temperature in the SEIS listing. This error went undetected by NPE personnel at the time because it occurred when the SEIS listing was originally being developed.

This deficiency was later discovered when one of the EQ binders for the NH series 90 actuators was returned to NPE after a contractor reviewed the seal material. The contractor noted problems with the seal material and changed the specified material to Viton. The binder previously required poly. Nuclear Plant Engineering reviewed the basis for the change and noted the change was specified because the COTTAP computer program predicted temperatures above the 104 degrees F listed in the EQ binder. A comparison with the 130 degree F listing in the SEIS showed the discrepancy. As a result, NCRs 90-0153 and 90-0154 were written to document the discrepancy. These NCRs stated that the poly seal's qualified life was less than one year at the elevated temperatures, whereas, the Viton seals could be qualified for at least seven years. The plant decided to replace one train of poly seals with Viton while the NCRs were being processed. The replacements were complete on July 21, 1990. This deficiency was determined reportable on July 24, 1990 because the seals in the remaining train of the SGTS and DX were believed to have been in service for longer than one year. The appropriate LCO was entered and the seals were replaced with seals made of the Viton material.

4.2 Elevated Reactor Building Temperatures

As a result of problems discovered during the licensee's 10CFR 50 Appendix R fire safe shutdown analysis, the licensee identified the need for the development of a transient model to reanalyze reactor building post accident air temperatures. This post accident transient temperature analysis performed using the COTTAP computer program confirmed that under worst case conditions some safety related equipment in the reactor building would be exposed to temperatures greater than those assumed for existing environmental qualification.

The COTTAP computer model accounts for heat loads resulting from outside walls, adjacent rooms piping systems, mechanical equipment, electrical equipment and heat removal using the HVAC systems. The results of the analysis were documented in Calculation No. M-RAF-024. The calculation determined that certain heat loads must be eliminated from the reactor building in order to limit the temperature rise to acceptable values considering harsh environmental qualification limits. However, these loads would not have to be eliminated until 24 hours following a postulated design base accident. As a result, the licensee developed a procedure, Emergency Plan Implementing Procedures (EP-IP-055), which requires shedding of certain loads in the Reactor Building of the unit involved in the postulated accident as mechanical cooling is restored. The action to reduce specific heat loads would not be made until 24 hours after the postulated accident occurred. The licensee identified the affected EQ binders and equipment that would be subject to elevated reactor building temperatures. The results of this analysis were documented on October 20, 1988 in NCRs for each unit. A report pursuant to 10 CFR 50.9 was approved on December 12, 1988 and reported to the NRC the following day.

4.3 Binder Upgrade

The NRC initially reviewed the licensee's compliance with 10 CFR 50.49 as part of the original plant licensing review. The review was primarily based on initial program submittals and JCOs for those items for which qualification was not completed. This review was documented in SER Supplements 3, 4, 5, and 6.

An evaluation of the EQ program was performed by the NRC office of NRR during May of 1982. This evaluation led the NRC to conclude that, subject to completion of several confirmatory items, compliance with 10 CFR 50.49 had been demonstrated. After the 1982 evaluation, inspections were scheduled to verify adequate implementation of the EQ program. Combined Inspection Report 50-387/86-25 and 50-388/86-28 documented the licensee's efforts to upgrade their EQ binders. This effort was necessary because of the way the binders varied in their quality, content and organization. Some of the binders were very difficult to use. At that time, the licensee stated that they would continue their binder upgrade program that had begun shortly before the inspection. The licensee's plans to complete the binder upgrade program was documented in Inspection Report 50-387/86-25.

During this inspection, the inspector noted that the licensee did not complete the upgrade program in 1987, as expected. Only 12 of 88 binders had been upgraded. Upgrading of the licensee's EQ binders is discussed further in Section 8.0 of this report.

4.4 Overall EQ Concerns

In addition to the polyurethane (poly) damper seal EQ concern, the NRC continued to have long-standing concerns regarding specific aspects of the licensee's EQ program. First, the licensee stated plans to continue to upgrade their EQ binders with an estimated completion date of 1987 and appeared to have failed to do so. Second, the licensee, because of a noted inadequacy during an Appendix R review, developed a revised reactor building temperature profile which required the reverification of the operability of EQ components contained in the RB. This reverification was flawed because it was based on unverified input data (SECS). Thus, the NRC decided to conduct a management meeting to discuss these issues. The management meeting is discussed in Section 5.0 of this report. Lastly, a concern was identified regarding the licensee's handling of NCRs involving EQ issues. This issue is discussed further in Section 7.0.

5.0 Management Meeting

The NRC held a management meeting on August 10, 1990 to discuss specific EQ concerns. Attachment 2 contains the list of Attendees. Attachment 3 contains a copy of the licensee's presentation.

The licensee discussed the scope and history of the EQ deficiencies identified as a result of the reactor building temperature issue which was raised in 1988. The meeting also addressed the error made in the EQ master index which resulted in the licensee's failure to identify NH-90 series actuators as not being qualified for their expected post accident conditions. Three other items were also reviewed: status of the EQ binder upgrade, control of the master index, and identification and timely reporting of the NH-90 actuator issue. The NRC concluded that the licensee provided reasonable assurance that the currently installed Viton seals satisfy EQ requirements. The licensee agreed to provide a schedule for their EQ binder upgrade program. In addition, the licensee confirmed that the SEIS equipment qualification temperatures were deleted from the SEIS list to prevent personnel from retrieving questionable data that could lead to further EQ problems.

Although the licensee provided some assurance that activities were completed safely and in accordance with requirements, the NRC determined that a special EQ inspection was warranted. This inspection was developed to review the specific activities surrounding the SGTS/DX Unit Damper Seals issue.

6.0 SGTS/DX Unit Damper Seals

On June 29, 1990 the licensee received a letter from Tenera (contractor) that transmitted EQ binder EQB-31 containing an upgraded format for the qualification of ITT NH-90 series damper actuators. The cover letter of the Tenera letter identified a potential problem in the qualification of the polyurethane seals used for NH90 series dampers. It stated that the qualified life of these seals was 0.33 yrs. without considering the effects of a DBA LOCA. The letter recommended that the licensee switch to Viton seals. On July 2, 1990 the licensee initiated a review of the binder which noted the recommended switch to Viton seals. The licensee's review emphasized that the Viton seal had not been qualified but missed emphasizing the important qualification issue of the polyurethane seals. On July 11, 1990 NPE notified the plant that poly seals were installed on NH90 series damper actuators. On July 17, 1990 NPE notified the plant that problems might exist with the use of poly seals. Nuclear Plant Engineering initiated work to qualify the Viton seals and issued an Replacement Item Equivalency (RIE) to use Viton seals on July 20, 1990. Division I seal replacement was begun for the SGTS and the DX units. The licensee completed their replacement of the Division I unqualified seals with Viton seals on July 22, 1990. The licensee had not yet completed their operability evaluation on the poly seals at that time. The replacement of the Division I seals was a considered prudent measure.

When the licensee completed their evaluation of the poly seals on July 24, 1990 they declared the SGTS system inoperable and entered the appropriate TS LCO. By July 25, 1990 the Division II seals were replaced with Viton seals. NCRs 90-0153 and 90-0154 were written on July 24, 1990 to identify and document the nonconforming condition. The Division II

seals were replaced. These NCRs were appended to SOOR 1-90-203 which documented that the SGTS and DX units did not meet environmental qualification requirements and hence would have prevented these systems from fulfilling their safety functions. EDR G00060 was written to analyze and determine the polyurethane life in response to the Tenera (contractor) letter.

Since the seals had exceeded their qualified life, the licensee declared them inoperable. TS 3.6.5.3 requires that both SGTS trains be operable. Since the SGTS recirculation plenum suction dampers were not environmentally qualified, the SGTS was inoperable and had been inoperable for an extended period. In addition, the support function of the emergency switchgear room cooling function was degraded. This was an apparent violation of TS 3.6.5.3 and an apparent environmental qualification violation (50-387/50-388/19-17-01).

In addition to the SGTS inoperability, the inspector reviewed activities concerned with the identification and documentation of these EQ deficiencies. The inspector noted that the licensee received the Tenera letter on June 29, 1990, yet failed to write an EDR or an NCR until July 24, 1990. PP&L Quality Assurance Manual, OPS-5 requires that the licensee's deficiency control system promptly report and correct conditions that are adverse to quality. EPM-QA-122 requires the prompt identification and documentation of engineering discrepancies. Thus, the licensee failed to identify and process this deficiency in accordance with OPS-5 and EPM-QA-122. This is a potential violation (50-387/50-388/90-17-02).

7.0 EQ Related NCRs

During the inspection several EQ related NCRs were reviewed. The inspector noted a discrepancy in NCR No. 88-0659 involving the use of BUNA-N O-rings in the H_2/O_2 analyzer sample pumps (2V219A/B) in place of neoprene O-rings.

The category 1 qualifications for this system were based on using sample pumps with neoprene O-rings and silicon O-rings. BUNA-N O-rings were not tested or analyzed for use under post accident conditions. However, the licensee determined that the operating conditions were not significantly stressful to the O-ring material in its use as a static seal and the design radiation dose was not expected to deteriorate for the short term operation required of the system. Replacement of the nonconforming O-rings in the installed pumps with new O-rings required to maintain the category 1 qualification is to take place within six months. After the six month period, all spare pumps are to be built or rebuilt with Category 1 qualified O-rings. To date, only the pumps in Unit 2 have been replaced. Unit 1 H_2/O_2 pumps are scheduled for replacement at the next refueling outage (Fall 1990).

NCR 88-0661 identified the installation of Viton seal kits in ITT damper/ valve actuators as being nonconforming with respect to the licensee's EQ program. The original installation specified polyurethane seals. The inspector reviewed the conditional release and noted that it was properly processed with an adequate engineering evaluation. The inspector also noted that the qualification report reference in the NCR to determine the Viton Seals "qualifiability" was the same report used to "qualify" the seals on July 21, 1990. Since the qualification report existed at the time the NCR was written, there was no basis for the conditional release. The continued use of conditional releases in place of NCR closeout is viewed as an undesirable practice.

Two NCRs reviewed pertained to the lack of qualification for numerous limitorque motor actuators. Of specific concern were NCR's 88-0181 and 88-0520 which were originated on March 24, 1988 and July 11, 1988 respectively.

NCR 88-0181 identifies the concern that 21 motor actuators in each unit are equipped with Reliance dc motors which were not subjected to Limitorque qualification testing. The qualification testing related to these motor actuators was performed on Porter/Peeless dc motors which have not been clearly shown to be similar to the Reliance dc motors installed at Susquehanna. Although the evaluation for NCR 88-0181 identified a similarity analysis performed by Wyle Labs for the Shoreham nuclear plant which compared Reliance 125 Vdc and Reliance 480 Vac motors, PP&L has not shown its applicability to their 250 Vdc Reliance motors.

NCR 88-0520 identifies the concern that 31 motor actuators in each unit are operated with 250 Vdc control power which is twice as much as the 125 Vdc control power used in the limitorque qualification testing of these actuators. The 250 Vdc control power is routed though the motor actuator limit and torque switches which have exposed terminal connections. These connections would be subject to insulation breakdown due to moisture intrusion resulting from the accident environment. Although PP&L provided some evidence of "qualifiability" in its evaluation of this NCR, the evaluation was found to be weak in that

1. It relied partially on a test report (F-C3271) that included no pre-accident aging or radiation.

2. Low resistance readings have been recorded for fibrite torque switches, even at 120 Vdc.

3. The Limitorque motor actuator is not a sealed device and some moisture intrusion is expected.

In conclusion, although both NCR's 88-0181 and 88-0520 have been evaluated, a time period of over two years has elapsed without resolution of these issues. Although the licensee's interim position was that the Limitorque operators were "qualifiable," it was not apparent to the inspector that the installed Limitorque configurations would be finally qualified. If

the valves are not qualified, the safety significance of the problem would be high because the subject valves control many safety components needed to mitigate the consequences of an accident. During a conference call between NRC and PP&L on September 11, 1990, the licensee stated that the final qualification determination for the Limitorque operators would be completed by October 31, 1990. The inspector noted that NRC Generic Letters 88-07 and 86-15 specify the need for prompt corrective actions following the identification of suspect EQ deficiencies. The lack of licensee prompt corrective actions for the suspect Limitorque EQ deficiencies is a potential violation (50-387/50-388/90-17-03).

8.0 Current EQ Binders Status

The inspector examined nine EQ binders to verify the licensee's revision of the binders to include data affecting qualified life and Maintenance/ Surveillance replacement schedule of EQ components as a result of the increased Post-Accident DBA temperatures. The inspector noted that Binder Change Notices (BCNs) were issued for each binder upgrading component qualification to the revised Post-Accident DBA temperatures for the Reactor Building. Where the increased temperature affected the qualified life of the component, the maintenance and surveillance requirements were revised to reflect the effects of the higher Post-Accident DBA temperatures.

Of the nine EQ binders examined, two EQ binders (EQDF 33 and 34) involved movement of the critical components to a mild environment. Thirteen components were involved (10 from EQDF 33 and 3 from EQDF 34). Eleven have been verified as having been moved to cabinets in a mild environment. The two remaining items are scheduled to be moved to mild areas by December 31, 1990. A justification for Interim Operation was developed to establish equipment operability pending final resolution of this issue (Reference Meeting Notes of March 3, 1989).

The licensee has developed its own EQ Binder prototype (EQPL-E13) as the basis for upgrading all EQ Binders. The PP&L EQ Binder Prototype is auditable with information that is easily tracked with support information that is easily accessed when required.

During the August 10, 1990 management meeting, the licensee committed to providing the NRC with a plan and scheduled completion date for upgrading all EQ Binders in accordance with the PP&L EQ Binder Prototype. This item is unresolved pending NRC review of licensee proposed schedule for Completing the EQ Binder Up-Grade (50-387/50-388/90-17-04).

9.0 Loss of SGTS/DX Unit - Safety Consequence Assessment

The licensee was asked to assess the safety consequences of the seal failures during a DBA and to specifically assess the repair efforts needed and the exposures to individuals during damage control operations to mitigate the effects of a loss of these dampers.

Both the DX Unit and SGTS systems were evaluated by the licensee. The function of the SGTS dampers (PDD-07554A/B) is to modulate airflow from the secondary containment to maintain a negative pressure of -1/4" wg upon receipt of a secondary containment isolation signal. During the initial drawdown phase, these dampers open fully due to the loss of the negative pressure within the secondary containment. Following the drawdown period (less than 92 seconds per Tech Specs), the secondary containment pressure has returned to the required -1/4 wg. Responding to the pressure changes, PDD-07554A/B modulate at some intermediate position to hold -1/4" wg.

The DX Unit valves (HV-27203A/B) modulate ESW system cooling water flow to the DX condenser to remove heat from the system. The DX units function to remove heat from air supplied to the Unit 2 4 kV switchgear rooms following a LOCA (i.e., when reactor building chilled water is not available). Valves HV-27203A/B throttle ESW system flow to maintain the condenser at a constant pressure.

EDR G00060 documented the nonconforming condition with regard to qualification of ITT General Control NH90 series actuators. The engineering assessment included in the EDR indicated that the failure mode of the actuator with polyurethane and viton seals is such that hydraulic fluid would leak past the seals and eventually cause the actuator to drive to an end position (via the spring pressure). In the case of SGTS Dampers PDD-0755A/B, this end position closes the damper. Since this is a common problem to both the A and B dampers, both dampers are postulated to fail closed. In the case of DX Unit valves HV-27203A/B, this end position opens the valve. Since this is a common problem to both the A and B valves, both valves are postulated to fail open. This engineering assessment states that the qualification test data does not provide a basis to calculate how long into the post-LOCA period the valves would function. However, based on engineering judgement, the licensee's NPE organization believes the valves to both systems would continue to operate for at least a few days and possibly up to 30 days. The actual time would depend upon the valves operating history prior to the event.

Based on this assessment, the postulated valve failure would most likely not occur until several days following an accident. Failure of the DX Unit valves to the full open position would result in overcooling of the DX unit condenser and a trip of the system on low suction pressure. Given the fact the failure would probably occur days into the event, the heat load within the switchgear rooms would be lower since many ECCS loads would probably have been shutdown. Also, it is possible normal ventilation to the reactor building (including the emergency switchgear) may have already been restored (ref. EP-IP-055). This emergency procedure directs personnel to restore normal reactor building ventilation following an event if no source term release has occurred. If a source term release has occurred, the DX units must remain in service to provide switchgear cooling. Thus, if a loss of the DX units occurs it would be up to emergency response personnel to determine the

appropriate actions based on the nature of the event and plant conditions at the time. Such actions could include manually positioning the actuators locally to maintain correct DX condenser pressure (handcranks exist on HV-27203A&B), manually throttling other valves on the ESW system piping to or from the DX units to maintain correct DX condenser pressure, or replacing the valve actuators. These actions would be dependent on the post-accident dose rates in the reactor building where the valve actuators and ESW throttle valves are located. Manual positioning of the DX unit actuators or manual throttling of ESW system valves would take less than 1 hour. Replacement of the valve actuator would take about 4 hours. Assuming worst case conditions in the reactor building (i.e., LOCA with failure of ECCS resulting in 100 percent noble gas and 50 percent iodine release), the following exposures per individual based on 45 minute stay time have been calculated (45 minute stay time based on SCBA use and skin exposure):

Whole Body	4.5 Rem
Lens of Eye	20.4 Rem/0 Rem depending on eye protection
Thyroid	0.0 Rem
Skin	150.0 Rem

All of the above exposures are within the emergency exposure limits for equipment saving action. The assessment for SGTS is essentially similar with the same safety consequences. The major differences are that the postulated damper failure would occur after the initial SGTS drawdown function was completed. The major concern would be one of long term maintenance of secondary containment pressure using SGTS. Given the fact that the failure would probably occur days into the event, it is possible that SGTS may have already been secured and the normal reactor building HVAC systems restarted (Ref. EP-IP-055). This emergency procedure directs personnel to restore normal reactor building ventilation following an event if no source term release occurred. If a source term release has occurred, the SGTS system must remain in service. Failure to modulate dampers PDD-07554A/B closed would result in a loss of secondary containment negative pressure.

The inspector reviewed the consequences and noted that the assumptions used by the licensee are extremely conservative since the plant's ECCS is designed to mitigate the effects of a DBA with no fuel damage. The use of 100 percent noble gas and 50 percent iodine from the fuel as a source term, in conjunction with one percent per day primary to secondary containment leakage resulted in a 6 R/hr dose rate from noble gas, <6 R/hr dose rate from iodine and a 2500 R/hr dose rate from beta radiation inside the reactor building. The licensee's use of double PCs with fire turnout gear would effectively shield a large portion of the limiting beta radiation. Additionally, the inspector noted that the actual damage control efforts could be accomplished in a 10 to 20 minute time frame since it would involve disconnecting the actuator from the damper and wiring it in an acceptable position. Therefore, the calculated exposures are considered extremely conservative with respect to the expected exposure.

10.0 Conclusion

The licensee was made aware of a potential EQ deficiency related to the ITT NH 90 series damper actuator seals on June 29, 1990. Action was taken by the licensee to further evaluate this apparent deficiency but it was not done in accordance with the licensee's procedural framework. An inoperability determination was made 25 days after the initial notification by the vendor which was considered acceptable. However, prompt identification and processing of the original discrepancy may have resulted in more prompt resolution.

Nonconformance reports have been written to document potential EQ deficiencies. These NCRs are not limited to just the polyurethane damper seals but involve the environmental "qualifiability" of other systems and components. Inspector review noted frequent use of Justifications for Interim Operation (JIOs) on EQ deficiencies in lieu of prompt resolution of the qualification issue. The licensee's failure to bring about prompt corrective actions for identified EQ deficiencies in accordance with Generic Letters 88-07 and 86-15 is a concern requiring licensee management attention.

11.0 Unresolved Items

Unresolved items are matters which require more information in order to ascertain whether they are acceptable items, or violations. An unresolved item identified during this inspection is discussed in Details paragraph 8.0.

12.0 Exit Meeting

The inspector met with licensee and licensee representatives during the inspection at the site and the corporate office and at the conclusion of the inspection in a conference call to discuss the scope of the inspection and inspection findings.

At no time during the inspection was written material provided to the licensee. This report does not contain information subject to 10CFR 2.790 restrictions.

ATTACHMENT 1

Abbreviation List

AD	- Administrative Procedure
ADS	- Automatic Depressurization System
ANSI	- American Nuclear Standards Institute
CAC	- Containment Atmosphere Control
CFR	- Code of Federal Regulations
CREOASS	- Control Room Emergency Outside Air Supply System
DG	- Diesel Generator
DX	- Direct Expansion
ECCS	- Emergency Core Cooling System
EDR	- Engineering Discrepancy Report
EP	- Emergency Preparedness
EPA	- Electrical Protection Assembly
EQ	- Environmental Qualification
ESF	- Engineered Safety Features
ESW	- Engineering Service Water
EWR	- Engineering Work Request
FO	- Fuel Oil
FSAR	- Final Safety Analysis Report
ILRT	- Integrated Leak Rate Test
LCO	- Limiting Condition for Operation
LER	- Licensee Event Report
LLRT	- Local Leak Rate Test
LOCA	- Loss of Coolant Accident
LOOP	- Loss of Offsite Power
NCR	- Non Conformance Report
NDI	- Nuclear Department Instruction
NPE	- Nuclear Plant Engineering
NRC	- Nuclear Regulatory Commission
OI	- Open Item
PC	- Protective Clothing
PCIS	- Primary Containment Isolation System
PMR	- Plant Modification Request
QA	- Quality Assurance
RCIC	- Reactor Core Isolation Cooling
RG	- Regulatory Guide
RHR	- Residual Heat Removal
RHRSW	- Residual Heat Removal Service Water
RPS	- Reactor Protection System
RWCU	- Reactor Water Cleanup
SEIS	- Susquehanna Equipment Inventory System
SGTS	- Standby Gas Treatment System
SI	- Surveillance Procedure, Instrumentation and Control
SO	- Surveillance Procedure, Operations
SOOR	- Significant Operating Occurrence Report
TS	- Technical Specifications

ATTACHMENT 2

Environmental Qualification Management Meeting

August 10, 1990

Name	Position/Company
Jim Stair	Resident Inspector/NRC
William W. Williams	Licensing Engineer - PP&L
George J. Kuczynski	Technical Supervisor - SSES - PP&L
James M. Kenny	Licensing Group Supervisor
Gene Stanley	Superintendent of Plant
Thomas A. Gorman	Supervising Engineer - PP&L
Alan P. Derkacas	Senior Project Engineer - PP&L
Jacque Durr	Chief, Engineering Branch, RI
Cliff Anderson	Section Chief, Plant System Section, RI
Wayne Hodges	Director, DRS, Region I
Mohan C. Thadani	Project Manager, NRR, NRC
Walter R. Butler	Project Director, NRR
James T. Wiggins	Deputy Director, DRP, Region I
Paul D. Swetland	Chief, Reactor Projects Section 2A
Ralph Paolino	Senior Reactor Engineer, PSS/EB/DRS
Chuck Meyers	Manager - Nuclear Projects, PP&L
Al Male	Manager - NPE, PP&L