U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS. 50-387/95-13
50-388/95-13

LICENSEE: Pennsylvania Power and Light Company (PP&L)

FACILITY: Susquehanna Steam Electric Station (SSES)
Berwick, Pennsylvania 18101

DATES: May 30 - June 9, 1995

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SUMMARY: The inservice test program at SSES was effectively supervised, well documented, and properly implemented. Strong corporate engineering support to the plant was evident in resolving potentially significant safety issues such as the effect of liquid containment bypass leakage on off-site radiological dose limits. The new condition report program involved senior management early in the corrective action process, improving the quality and timeliness of problem resolution and communication between the site and corporate offices. Condition reports initiated at PP&L were dispositioned properly and communicated to SSES in a timely manner. An excellent industry event review program was noted, although more attention to administrative requirements for extension of action due dates is warranted. Effective oversight of contracted engineering services was accomplished through integration of these services into PP&L's programs. The high quality of audit findings and observations in this area was noteworthy. The Engineering Review Committee, an ongoing self-assessment initiative, significantly contributed plant safety through review of design issues and engineering programs. Strong performance was noted in the area of core reload safety analysis. Management support was evident in the initiatives implemented in response to Generic Letter 94-01 regarding core stability.

Two unresolved items were closed: URI 50-387,388/95-01-02, HPCI Suction Transfer Logic; and URI 50-387,388/95-01-03, Post-Scram Uncontrolled Rod Withdrawal. Two violations of NRC requirements, involving premature implementation of an inservice test relief request and an undocumented quality assurance audit, were not cited due to low safety significance and prompt, appropriate corrective action.

DETAILS

1.0 INSPECTION SCOPE

The purpose of this inspection was to assess the effectiveness of the licensee's engineering organization in support of safe operation of the Susquehanna Steam Electric Station. The inspection consisted of reviews of the Inservice Test Program, engineering involvement in resolving technical problems and design deficiencies, oversight of contracted engineering services, and the conduct of core reload safety analyses. Particular attention was paid to the roles of senior management and the quality assurance organization in ensuring safe plant operation. NRC Temporary Instruction 2515/114 (Inservice Test Programs) and NRC Inspection Procedures 37550 (Engineering) and 61710 (Refueling Activities) were used as guidance for this inspection.

2.0 INSPECTION FINDINGS

2.1 Inservice Test Program

The purpose of inservice testing (IST) is to assess the operational readiness of pumps and valves, to detect degradation that might affect component operability, and to maintain safety margins with provisions for increased surveillance and corrective action. The requirements for IST are contained in plant Technical Specification (TS) 4.0.5, which requires testing in accordance with 10 CFR 50.55a, "Codes and Standards," and Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code).

The licensee currently is implementing the second 10-year interval of the IST program as described in program submittals to the NRC dated June 30 and December 29, 1974. The submittals included requests for relief from several Code requirements, and incorporated Pennsylvania Power and Light Company's (PP&L) responses to the regulatory positions contained in Generic Letter (GL) 89-04, "Guidance On Developing Acceptable Inservice Test Programs." An NRC letter, dated April 26, 1995, forwarded the safety evaluation report (SER) which provided the results of the NRC staff's review of the Susquehanna IST program. In this inspection, components primarily in the residual heat removal service water, control rod drive, high pressure coolant injection, and core spray systems were reviewed. Plant risk, relief requests, and program anomalies identified in the SER were considered in the selection of these systems.

General IST Program Review

The inspectors verified that program responsibilities were assigned and that requirements were being implemented effectively. Licensee personnel were found to be well versed in IST program requirements. Adequate administrative controls were in place to: (1) schedule and track test performance, (2) ensure appropriate acceptance criteria were met, (3) ensure that reference values were reverified or reestablished following component repair or replacement, and (4) analyze and trend test data.

The Susquehanna IST program is described in administrative procedure NDAP-QA-0423, "Station Pump and Valve Testing Program." Testing is performed pursuant to Section XI of the Code (1989 Edition), which incorporates by reference Parts 6 (OM-6) and 10 (OM-10) of ASME/ANSI OMa-1988 for pumps and valves, respectively, and Part 1 (OM-1) of ASME/ANSI OM-1987 for pressure relief devices. Design information and test requirements for specific components, relief requests, and test deferral justifications are contained in detailed program plans, and supporting technical details are contained in sub-tier procedures, calculations, and engineering specifications. The inspectors found the program documents to be particularly well organized, cross-referenced, and readily available.

At the time of the inspection, procedure NDAP-QA-0423 had been revised considerably to incorporate the new requirements of OM-6 and OM-10. However, some holdovers from the previously applicable Code edition remained to be changed, and steps applicable to relief requests which were disapproved by the NRC in the SER had not been deleted or modified. The licensee stated that a revision was being prepared to conform to the provisions of the SER.

The inspectors verified that the scope requirements of OM-10 were met through review of system drawings and the Susquehanna Final Safety Analysis Report. The inspectors noted that two check valves located between the residual heat removal (RHR) service water and RHR systems were not in the program. The licensee explained that the open function of the valves was beyond the plant design basis, and that the isolation function (to prevent diversion of RHR from the reactor coolant system) was performed by redundant motor-operated valves (MOVs) 112F078A and 112FL78B located in the same lines. The inspectors noted that the MOVs were in the IST program. In addition (in response to a quality assurance surveillance recommendation) the licensee had established new preventive maintenance activities to exercise the check valves every 18 months. The inspectors also questioned deletion from the program of the emergency service water supply and return isolation valves to/from emergency diesel generators A through D. The licensee provided the safety evaluation and relevant portions of design change package 88-3016I, which removed the automatic and remote-manual safety functions of these valves. The inspectors concluded that this adequately justified removal of valves from the program, and that administrative controls effectively ensured that the IST program was maintained current with changes in plant design.

IST Procedures

For the selected components, completed surveillances for the past year were reviewed to confirm that instructions were sound technically, that components were tested in accordance with the requirements of the Code and the regulatory positions of GL 89-04, that appropriate quantitative or qualitative acceptance criteria were established, and that procedures were reviewed and approved per licensee requirements. The procedures were found to be of good quality, and clear, step-by-step instructions were provided in the event that a component failed a test.

The inspectors observed several instances in which new valve acceptance criteria based on reference stoke times had not yet been incorporated into procedures. However, these values had been developed and were available in procedure NSEP-AD-401, "Inservice Testing (IST) Program Stroke Time Limitations For Power-Operated Valves." Procedure revisions were being prepared and issued as-needed prior to performance of scheduled tests, and the inspectors found no cases in which incorrect acceptance criteria had been utilized.

Pump Testing

Surveillance procedures and performance records for the high pressure coolant injection (HPCI), emergency service water, residual heat removal service water, and core spray pumps were reviewed against OM-6 requirements for IST of pumps. The test frequencies, quantities measured, and allowable ranges were consistent with those specified in OM-6, Tables 2 and 3. Test instruments were verified through review of calibration data sheets to meet the accuracy requirements of OM-6, Table 1 and section 4.6. The licensee properly dispositioned test results which entered "alert" (if applicable) or "required action" ranges and maintained a computerized database to trend pump performance.

The inspector noted that procedure SO-252-002, "Quarterly HPCI Flow Verification," which was performed on March 8, 1995, specified a one minute stabilization period at the required flow rate prior to taking performance data. This is contrary to OM-6, section 5.6, which requires a two minute stabilization period. The licensee documented this deviation from the Code in a relief request submitted to the NRC in 1994. The inspector's concern regarded implementation of relief requests prior to NRC approval.

10 CFR 50.55a requires IST of components in accordance with the requirements of the Code to the extent practical within the limitations of design, geometry, materials of construction. Where a test requirement is determined by the licensee to be impractical, the basis of the determination must be demonstrated to the satisfaction of the NRC. Because of impracticality, a licensee may test applicable components by the method proposed in a relief request until an NRC evaluation is completed. However, relief requests which do not relate to impractical requirements, but rather, propose alternatives to Code requirements, are not to be implemented prior to NRC approval. In its relief request, the licensee stated that a two minute stabilization period was impractical primarily due to heatup of the suppression pool caused by the HPCI turbine exhaust. However, the inspector noted that during the first 10-year interval, the licensee had complied with the five minute period specified by Article IWP-3500 of Section XI. In addition, due to the physical constraints involved, the licensee typically met or exceeded the two minute requirement when performing the surveillance. The licensee promptly revised the surveillance procedure to comply with the Code when the NRC denied the requested relief in April 1995.

The inspector concluded that compliance with the two minute stabilization period had been practical, and that the licensee's implementation of an alternative test method prior to NRC approval was a violation of 10 CFR 50.55a. However, this failure constitutes a violation of minor significance and is being treated as a noncited violation, consistent with Section IV of the enforcement policy (60 FR 34381, dated June 30, 1995).

Valve Testing

Test methods and frequency, acceptance criteria, and corrective actions for several types of valves in the selected systems were reviewed, and except as noted below, were found to meet the provisions of OM-10. The containment isolation valves listed in TS Table 3.6.3-1 for both units were included in the IST program as Category A or A/C valves. As documented in the programs, these valves are leak rate tested in accordance with OM-10, section 4.2.2 and 10 CFR 50, Appendix J. Also, the pressure isolation valves listed in TS Table 3.4.3.2-1 also were incorporated appropriately into the IST programs at both units.

Procedures and test records of power-operated valves were assessed against the stroke time criteria of OM-10, sections 4.2.1.4 and 4.2.1.8, and position 6 of GL 89-04. A minor discrepancy was noted concerning a quarterly test of air-operated scram discharge volume vent and drain valves which was performed in April 1995. The valves satisfied limiting stroke time acceptance criteria, but failed the criteria based on reference values. The inspector noted that the ensuing corrective action record was technically correct, but was not approved by the plant operations review committee as specified in procedure. NDAP-QA-0423. The inspector found the administrative discrepancy to be an isolated instance of lack of attention to detail.

OM-10, section 4.3.2.2, provides for deferral of quarterly check valve exercise tests to cold shutdowns and/or refueling outages. The inspectors reviewed several cold shutdown and refueling outage justifications and verified that most could not be exercised practically during power operation. In the new IST program, the licensee added tests of the emergency diesel generator starting air receiver inlet check valves every refueling outage (vice every three months). The licensee justified the test deferral on the basis of impracticality; viz. lack of installed instrumentation or provision to install temporary instruments, and degradation of diesel start capability during test performance. In the SER, the NRC staff questioned the licensee's justification. The inspectors walked down the system using a draft surveillance procedure and concluded that the test was practical (albeit laborious) to perform on a quarterly basis. The licensee agreed to implement the test at the frequency specified by OM-10 pending NRC approval of a relief request.

The inspectors found that the licensee was implementing the requirements of OM-1 for testing of main steam safety/relief valves (SRVs) properly. However, in the case of Class 2 and 3 relief valves, the licensee's practices appeared not to conform fully to the provisions of OM-1, section 3.2.2. That section requires that tests prior to maintenance or set pressure adjustment, or both, be performed in the following sequence: visual examination, seat tightness test, set pressure determination. While test data sheets indicated satisfactory performance of seat tightness tests, the wording and order of steps in procedure MT-GM-005, "Safety/Relief Valve Setting," indicated that the leakage tests were performed following verification of the lift setting rather than "as found." Clarification 4.3.9 of NUREG 1482, "Guidelines For Inservice Testing At Nuclear Power Plants," states that seat tightness is to be determined before determining the set pressure only if practicable. However, it was not apparent to the inspectors that the licensee justified its practice on the basis of impracticality. The licensee agreed to review and upgrade its procedures to incorporate this provision of OM-1.

Control Room Observations

The inspectors observed the performance of procedure SO-149-002, "Quarterly RHR System Flow Verification," by control room personnel. The operators properly adhered to the procedure and performed the test in a professional manner.

The inspectors also reviewed the contents of the licensee's "hot box," which is used for quick distribution of information to the operations staff. The system is designed to ensure review of information which cannot be deferred to the next shift training session. Personnel are required to read and sign for the information in the box, which included procedure changes and the status of condition reports. The inspectors verified that the material in the box was reviewed by the operators in a timely manner.

Reliability of Containment Instrument Gas System Check Valves

The new IST program added exercise tests of 22 containment instrument gas (CIG) system check valves associated with the main steam safety relief valve (SRV) accumulators. Each of the 16 SRVs has a small accumulator and check valve (F036). The six SRVs which also perform an automatic depressurization (ADS) function have an additional, larger accumulator and check valve (F040) sized to open and hold open the SRV against post-accident drywell pressure with the reactor completely depressurized. The accumulators are intended to ensure that during a small break loss of coolant accident (SBLOCA) reactor pressure can be reduced so that low pressure emergency core cooling systems can inject makeup water into the reactor vessel should the HPCI system be unavailable. The check valves close to isolate the accumulators from a break in the upstream CIG piping.

Three F036 valves did not meet the acceptance criterion for verification of proper closure during testing at Unit 1 in April 1995. Also, during inspection of the F040 valves, valve F040M was found to have loose and damaged internal parts. Following repairs, all 22 of the check valves were tested satisfactorily. The licensee initiated a condition report (CR) to investigate prior performance of the valves, and to assess the safety significance of the failures. The CR stated that similar valves at unit 2 would be tested and repaired as necessary in November 1995, during the next refueling outage. The inspector noted that loose and missing internal parts previously had been identified as a problem at Unit 1; in the mid-1980's, seven check valves were found with loose seats and missing stem nuts and lockwashers. The licensee concluded that the condition had minimal safety impact since each ADS valve would have at least one repaired check valve.

The inspector considered CR to be deficient in several respects. The report did not differentiate the functions of the differently sized SRV accumulators. Thus the existence of at least one repaired check valve did not assure that the ADS function was unimpaired. In addition, the inspector was not persuaded that the licensee had determined the root cause of the failures, and was concerned by the absence of a technical basis (ie. maintenance or test records) for concluding that the ADS valves at Unit 2 were not similarly affected. The licensee initiated another CR specifically to evaluate ADS operability at Unit 2. The licensee found that the problems at Unit 1 had occurred during construction, and that the issue had been assigned to the architect-engineer for resolution at Unit 2. A satisfactory preoperational test of one set of ADS check valves was found, but no conclusive evidence could be located that the other check valves had been installed correctly.

The licensee's operability determination relied chiefly on system design considerations. The CIG system has two redundant trains, each provided with safety-related backup high pressure nitrogen bottles. Both supply headers inside the containment were designed and installed to withstand seismic and LOCA loadings. Finally, failure of both a CIG header and the HPCI system was considered to be beyond the plant design basis. Thus the check valves were not needed to assure the operability of the ADS valves. The inspector concluded that the operability determination was acceptable, but noted that it did not address the potential for consequential failure (through pipe whip or jet impingement) of the CIG piping during a LOCA. Subsequently, the licensee reviewed high energy line break studies and confirmed that the CIG system was not subject to consequential failure. The inspector concluded that there was adequate assurance of ADS system operability at Unit 2.

The inspector concluded that the resolution to the initial CR had been uncharacteristically weak for this licensee. However, the licensee responded promptly to the Unit 2 operability concern, and the inspector noted that the second CR was prioritized appropriately, dispositioned adequately, and reviewed and discussed by the proper levels of licensee management.

(Closed) URI 95-01-03, Post-Scram Uncontrolled Rod Withdrawal - both units

This item involved the potential for unplanned control rod withdrawal following a reactor scram during a loss of coolant accident (LOCA). In Engineering Deficiency Report (EDR) No. 94-001, the licensee postulated that leakage past a nonsafety-related check valve (V114) in the scram discharge line to the scram discharge volume might result in withdrawal of multiple unlatched control rods. The potential for the occurrence was aggravated by lack of periodic testing or maintenance of the V114 valves. The licensee contacted General Electric Company (GE) to perform a hydraulic analysis of control rod drive system response during a postulated LOCA. In a proprietary report, GE concluded that the combination of conditions required to actuate the collet piston of an unlatched control rod were not credible.

The inspectors reviewed the EDR and GE's evaluation to assess the licensee's corrective actions and the rigor of the licensee's review of the GE report. The licensee generated a list of 13 technical questions and comments in its review of the draft GE report, which varied from challenging basic assumptions to soliciting maintenance guidance. The items raised by the licensee evidenced a detailed and critical evaluation process. The licensee reclassified the V114 check valve ball as safety-related for procurement purposes, and revised its maintenance practices to inspect the check valve during maintenance of the associated control rod drive outlet scram valve. The inspectors also noted that the V114 valves were added to the new IST program, with test requirements consistent with Position 7 of GL 89-04. The inspector also noted that the IST reduces the potential for backleakage by flushing the check valve ball and seat.

The inspectors concluded that the licensee demonstrated an excellent safety perspective in identifying and vigorously following up this potentially significant safety issue.

Program Audit

The licensee audited the Inservice Inspection and Testing programs in March 1994. The auditors found that the IST program and implementing procedures properly addressed ASME Code requirements. The inspector reviewed Audit Report #94-011 and concluded that the scope, findings, and observations of the audit included a good mix of compliance and performance-based attributes. A finding regarding management review of temporary changes to surveillance procedures was resolved quickly and satisfactorily. The licensee's use of technical specialists from outside of the company contributed to the independence of the assessment and was considered to be a strength.

2.2 Engineering Programs

Resolution of Technical Problems

Several safety significant plant deficiencies and problems were reviewed based on a screening of condition reports (CRs), engineering deficiency reports (EDRs), and NRC inspection items. The performance of the engineering organization and technical adequacy of problem solutions were assessed by review of relevant documents and discussions with corporate engineers.

Condition Reports

The licensee implemented a new CR program in March 1995 to streamline the process of resolving technical deficiencies and operational problems. The former plethora of corrective action programs was replaced with one comprehensive system. Provisions also were made to improve communication between corporate offices and the site, to involve management in the process earlier and more effectively, and to assure that corrective actions and actions to prevent recurrence were assigned and performed commensurate with safety-significance. The inspectors reviewed eight CRs which were initiated at the corporate offices since the inception of the program and verified that the potential problems were promptly reported to the site and dispositioned in a timely fashion.

The inspectors found that the licensee assigned appropriate safety significance levels to the CRs. Through a corrective action team, management involvement early in the process ensured that potential generic and programmatic issues were addressed by appropriate corrective actions in addition to the immediate technical concerns. The inspectors concluded that the new program effectively addressed communications weaknesses which had existed in previous corrective action processes.

Engineering Deficiency Reports

Since the elimination of EDRs, the licensee is working assiduously to eliminate the EDR backlog. The inspectors reviewed two EDRs to assess the quality of the resolutions to previously identified safety issues.

EDR G00040 evaluated the potential consequences to offsite radiological doses of water bypass leakage past the primary and secondary containments via the emergency core cooling system keep-full system check valves. The licensee determined that a cumulative keep-full system leakage rate limit of 0.55 gallons per minute was required to maintain offsite doses within the limits of 10 CFR 100. The licensee verifies total leakage below the limit by testing the check valves every refueling outage. The inspector noted that the cumulative design leakage rates of the check valves was far below the established limit, and that test results typically were less than the design leakage rates.

Because of the potential for water bypass leakage past the secondary containment through the remainder of the systems at Susquehanna, the potential existed that offsite dose limits still could be exceeded. Therefore, the licensee developed a comprehensive corrective action plan, under EDR 94-11, to address the concern. The inspectors reviewed calculation EC-059-1014, "Evaluation of Secondary Containment Bypass Leakage," to evaluate the quality of the licensee's plan.

The licensee concluded that water bypass leakage was not addressed explicitly in the regulatory documents applicable to Susquehanna, and that postulation of the leakage is beyond the design and licensing basis of the plant. The conclusion was well supported by reference to the Standard Review Plan (NUREG-0800, Section 6.2.3), Branch Technical Position CSB 6-3, the Susquehanna Safety Evaluation Report (NUREG-0776), Regulatory Guide 1.3, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident For Boiling Water Reactors," and other NRC guidance documents.

Notwithstanding the above, the licensee developed general engineering guidelines by which to evaluate water bypass leakage including: (a) identification of credible leakage pathways, (b) realistic determination of total leakage rate, (c) quantification of the dose contribution from the total leakage. Regarding the last consideration, the inspector noted that the licensee utilized a realistic (vice design basis) approach to the analysis, which permitted crediting reductions in offsite dose due to plate-out, decay, and dilution effects, a flashing fraction for radioactive iodine in the water, and the expectation that no fuel damage will occur during a design basis loss of coolant accident. The inspector considered the licensee's approach to be reasonable.

After verifying that the FSAR tables identified all potential leakage pathways, the licensee performed an evaluation of the containment penetrations which took into consideration actual leakage test results. The licensee concluded that the contribution to offsite dose from gaseous leakage far exceeded and bounded the effect of water leakage.

The inspectors found that the EDR resolutions were of high quality, and evidenced an excellent safety perspective. The exhaustive evaluation of this issue exemplified the licensee's willingness to expend considerable engineering resources to resolve potentially significant safety issues.

(Closed) URI 95-01-02, HPCI Suction Transfer - both units

This item involved EDR 94-046, which was initiated by the licensee to resolve an apparent conflict between HPCI system design and operator actions dictated by plant emergency operating procedures pursuant to Boiling Water Reactor Owner's Group guidelines. The EDR questioned the ability of the HPCI system to fulfill its safety function during a small break LOCA coincident with a loss of offsite power. By original design, HPCI pump suction transfers automatically from the condensate storage tank (CST) to the suppression pool (SP) on either low ST level or high SP level. If permitted to occur, however,

high SP temperature may overheat the HPCI lube oil cooling system and result in system failure. If the feature is bypassed per procedure, increasing SP level potential could flood the HPCI turbine exhaust line if the turbine trips. Subsequent automatic restart of the turbine could the result in water hammer and damage to the exhaust line rupture disks.

The licensee originally based HPCI system operability on the assumed capability to control SP level via letdown to the radioactive waste system, and downgraded the significance level of the EDR. Four months later, after completing formal calculations, the licensee determined that letdown flow rate was not sufficient to mitigate the increase in SP level. However, this discovery was not communicated promptly to the site, resulting a delay in reevaluating HPCI system operability. The inspector noted that the licensee had no guidelines in place regarding the technical basis needed to support an interim operability determination.

NRC review of HPCI system operability was documented in Susquehanna Inspection Report 95-02. Regarding programmatic corrective actions, the inspectors observed that procedure NDAP-QA-0702, "Condition Reports," contains provision for management review of significance level changes through participation in daily corrective action team meetings. The licensee also has under review a procedure containing guidelines for interim operability determinations which closely parallel the regulatory positions of Generic Letter 91-18. The inspectors concluded that these actions adequately addressed this item.

Industry Event Review Program

The inspectors reviewed the licensee's industry event review program (IERP) which is described in procedure NDAP-QA-0725. The program was established to collect industry information, review and classify it for safety significance, and develop corrective or enhancement actions. The inspectors sampled licensee evaluations of 20 NRC Information Notices, several GE technical letters and various Part 21 reports. The inspectors noted that the program successfully gathers and disseminates information both formally and through informal newsletters. Good quality responses to the items reviewed were noted. An apparent weakness was identified concerning control of the timeliness of action item responses. According to the IERP coordinator, approximately 50 items are open at any given time, of which about 30 are overdue, The inspectors found several instances in which extensions were not requested, as specified by the program procedure, and a few cases in which the number of permitted extensions had been exceeded. Notwithstanding, the inspectors considered the program to be excellent.

Oversight of Vendor Engineering

The inspector reviewed the licensee's oversight of contractor engineering services and verified that the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria For Nuclear Power Plants," and the PP&L Operational Quality Assurance Program were satisfied. Through a variety of formal mechanisms, the licensee exercises strong oversight and control over vendor engineering work.

The licensee has no unique procedures for the control of contracted engineering services. Instead, these activities are integrated fully into the PP&L organization, and contractors are trained in and work to the licensee's engineering procedures. Design reviews are performed in the same manner as those conducted in-house, and the accountability for the product resides with the licensee's design modification group. Appropriate licensee quality assurance program requirements were specified the procurement documents (General Services Agreements) for the services. The inspector reviewed two design changes relating to safety-related motor-operated valve actuators which were performed by Gilbert/Commonwealth, Inc. The design inputs, analyses, verifications, and reviews were complete and clearly documented. The impact of the changes on the plant systems was assessed fully in the 10 CFR 50.59 safety evaluations. In all respects, the modification packages were equivalent to those produced by the licensee itself.

The inspector reviewed documentation associated with the services provided by Alden Research Laboratory, Inc. in connection with experiments on emergency core cooling system (ECCS) strainer blockage. Alden was contracted to construct a hydraulic model of the Susquehanna suppression pool and ECCS strainers and to evaluate pump performance under simulated post-LOCA conditions. The results of the experiments formed the basis for removal of fibrous insulation from piping located in the primary containments at Susquehanna. The inspector noted that the services were procured as nonsafety-related. However, since the results of the study were used in safety-related applications, the licensee's quality assurance department performed several source verification inspections to ensure that the procedures, test plans, computer software, and test instruments used by Alden met appropriate quality standards. The inspector concluded that the licensee exercised excellent oversight of this safety-significant activity.

The licensee also evaluates contractor performance after the fact under an incentive program which includes an assessment of procedure compliance, use of design inputs, and completeness and technical quality of design change packages, documentation, and safety evaluations. The inspector reviewed performance evaluations for modifications involving elimination of HPCI pump suction automatic transfer and replacement of residual heat removal pump motor coolers, and concluded that the licensee performs vigorous and critical reviews of contractor performance.

The inspector reviewed PP&L Audit Report #94-122, completed in March 1995. The 2-month audit evaluated the design modification work performed by Gilbert/Commonwealth, Inc. and Stone and Webster Engineering Corporation under the auspices of the licensee's design modification group. The report concluded that the contractors generally comply with the licensee's administrative and quality requirements. The inspector found that the licensee performed a rigorous and critical assessment as evidenced by the number and content of findings and observations. The safety significance of the findings was relatively low, but illustrated the licensee's high expectations concerning strict adherence to procedures.

Engineering Review Committee

The Engineering Review Committee (ERC) is a senior level management self-assessment initiative intended to oversee the quality of engineering department performance. It accomplishes this function by performing in-depth reviews of procedures and processes, and by conducting investigations of significant operational problems having their root causes in design engineering inadequacies. The inspector assessed ERC performance through discussions with engineers and review of meeting minutes, audit reports, and engineering deficiency reports. For example, during its review of a recent audit on the ASME Section XI repair/replacement program, the ERC took note of an audit observation that instances of implementing ASME code cases and relief requests prior to NRC approval had been identified. The committee directed Nuclear Licensing to address this subject in the near future. Design issues such as environmental qualification of the high pressure coolant injection system, unqualified main steam isolation valve actuator cylinder seals, and unplanned post-LOCA control rod withdrawal also were discussed. The inspector found that the ERC focused on safe operation of Susquehanna, and provided an excellent vehicle for communicating and reinforcing high performance standards throughout the engineering organization.

2.3 Refueling Activities

The purpose of this inspection was to review PP&L's core reload performance analysis. The licensee is required to review the accident analyses described in the final safety analysis report (FSAR) each time the core design is changed. Core reload analysis is important to safety because it assures that a new core provides adequate margin to limit core damage during postulated accidents. The analysis supported the operation of Unit 1 through cycle nine. The information in the reload analysis was used by the licensee as a basis for a 10 CFR 50.59 safety evaluation which determined whether an unreviewed safety question existed.

Reload Safety Evaluation Report

The inspector reviewed core reload safety calculations performed by GE, Siemans Power Corporation (SPC), and the licensee for Unit 1, cycle nine in support of the core performance analysis report. This report contains descriptions and analysis results pertaining to the fuel mechanical design, thermal-hydraulic, nuclear physics, and operating safety aspects of the reload cycle. The inspector verified that the calculations were performed in accordance with nuclear fuel engineering (NFE) technical instructions and nuclear department procedure NFP-QA-009, "Reload Design and Analysis Program." The following calculations were selected for review:

NFE-1-09-013, "UIC9 Feedwater Controller Failure Analysis": Feedwater controller failure results in a rapid increase in feedwater flow which causes a turbine trip on high reactor water level. The results of this analysis change the critical power ratios which are used to determine the minimum critical power ratio (MCPR) operating limit.

Three modes of equipment operability were analyzed by the licensee: (1) main turbine bypass and end-of-cycle recirculation pump trip (EOC-RPT), (2) main turbine bypass inoperable and EOC-RPT inoperable, and (3) main turbine bypass operable and EOC-RPT inoperable. The inspector reviewed system model datasets along with assumptions and cases analyzed for these modes.

All three cases were performed at various initial powers and maximum core flow. For the "Bypass and EOC-RPT Operable Analyses" case, two different scram curves were utilized: (1) the TS 3.1.3.3 scram curve, and (2) the modified best estimate scram curve. The latter was selected because it is expected to conservatively bound actual cycle nine scram speeds. The inspector reviewed the results and found that delta critical power ratio (DCPR) using the most conservative scram curve was below the TS limit for various core powers.

The inspector also compared the Unit 1 cycle nine 9 feedwater controller failure results to the Unit 1 cycle eight and Unit 2 cycle seven analysis results for the turbine bypass and EOC-RPT operable cases. The difference between the cycle eight and cycle nine limits were attributable to an additional time delay on the high-level trip and the high core flow used in the cycle nine analyses. The difference between the Unit 2 cycle seven and Unit 1 cycle nine limits at low core power are attributable to the conservative time delay for the high-level trip in the Unit 2 analyses. Based on the above, the time delay would have a small affect on higher power cases since it affects the DCPR from the "cold water" part of the transient. The change in CPR for the feedwater controller failure event was due to the cold water transient at low core power conditions.

NFE-1-09-010, "Unit 1 Cycle 9 Overpressurization Analysis": The objective of this calculation was to document the limiting ASME overpressurization event and to demonstrate that the maximum resulting pressures are less than or equal to 100% of the vessel pressure boundary design pressures.

The inspector verified that main steam isolation valve (MSIV) closure was the worst case for the overpressurization analysis. The MSIV closure event was analyzed with RETRAN computer code MOD12 of the system model. The RETRAN input deck for the 102% power and maximum core flow was used along with the system parameters which optimized the overpressurization analysis. These parameters were technical specification MSIV closure time, high neutron flux trip setpoint, high reactor pressure, and maximum average scram times. The analysis showed that maximum calculated reactor vessel pressure due to MSIV closure was 1338.5 psig, which corresponded to a margin of 36.5 psi to the TS design criteria of 1375 psig. Based on the above, the calculated reactor dome pressure corresponding to peak vessel pressure was 1322.2 psig. The current TS safety limit of 1325 psig, based on dome pressure and a 50 psid vessel differential pressure, therefore was satisfied.

F-204, "Core Shutdown Margin Analysis," Core shutdown margin calculations were performed to assess whether the basic criterion for reactivity control was met. In its most reactive condition (cold, Xenon-free) the reactor core must be subcritical with the highest worth control rod fully withdrawn and all other control rods fully inserted. TS 3/4.1.1 requires that core shutdown margin must be at least 0.38% ΔK/K. Core shutdown margin depends on fuel bundle and core design and is a function of core exposure. Therefore, core shutdown margin must be evaluated throughout the expected operating cycle to assure adequate margin to the TS limit.

The inspector reviewed the Unit 1 cycle nine reload design report and interviewed the licensee's engineering staff regarding the core shutdown margin analysis. The calculated margin at any point in cycle nine exceeds the minimum 0.38% ΔK/K TS requirement. The result of the analysis showed a shutdown margin as a function of cycle exposure from 0.0 to 12.30 GWD/MTU. During the present cycle, the minimum shutdown margin of 1.25% ΔK/K occurs at 9.820 GWD/MTU, which is well above the TS limit.

In August 1994, PP&L revised the reload analysis for Unit 1 cycle nine. The revision differs from the original due to a change in energy requirements for the cycle as a result of the delayed shutdown during cycle eight. The shutdown margin calculation was reanalyzed based on cycle eight core exposure, and calculated a shutdown margin for cycle exposure from 0.0 to 1.970 GWD/MTU. PP&L stated that, based on engineering judgement, a calculation for the shutdown margin from 2.0 to 12.30 GWD/MTU was not needed. Although PP&L did not have any documentation to support this engineering judgement, the inspector concluded that the reload analysis was reasonable. The lack of documentation for the analysis also was identified by licensee management and was being corrected at the time of the inspection. The inspector had no further questions regarding this analysis.

Core Stability

The inspector reviewed PP&L's long-term solution to GL 94-01 regarding interim operating recommendations for instabilities in BWRs. In response to the GL, PP&L modified its operating procedures and operator training to be consistent with BWR owners' group guidelines for stability for BWRs. The guidelines incorporated a redefined and expanded stability region to strengthen the prevention of oscillations. The revised stability region provided enhanced protection against unacceptable power oscillation for Unit 1 cycle nine.

In addition to the above, PP&L elected to proceed with a long-term solution, which introduces new plant hardware/software to provide early detection of oscillation, and to initiate an appropriate mitigating action. This "Long-term Solution Stability System (LTSSS)" features the Option III concept description in GE NEDO-31960. Furthermore, recommended TS changes will be provided as part of the program. These changes will be incorporated at both Susquehanna units in the fourth quarter of 1996 and the second quarter of 1997, respectively. The inspector concluded that the licensee's response to the GL was acceptable.

Quality Assurance

The inspector reviewed two audit reports related to refueling engineering, and evaluated the effectiveness of the licensee's quality assurance (QA) program in identifying weaknesses and assessing the effectiveness of engineering activities. Audit Report #92-085, "Audit of Fuel Management; Fueling and Refueling; Special Nuclear Material and Source Control Program," performed in December 1992, evaluated the program for control fuel management, refueling activities, and safety-related and nonsafety-related engineering activities. The audit verified a well-established program which met regulatory requirements and licensing commitments, and which was effectively implemented by knowledgeable personnel. The audit team made eight observations and recommendations. One audit finding was identified relating to fuel inspection records; a fuel channel serial number did not correspond to the channel available for the core reload. A review of the reload videotape revealed the correct number, and the inspection record and fuel accountability records were corrected. All other observations were administrative in nature.

Audit Report #94-025, "NQA Audit of Power Uprate Project Implementations," was performed to verify the effectiveness of engineering support associated with the power uprate project. QA surveillance QASR-94-041 was performed in support of this audit. No findings were identified; however, five observations or recommendations were issued. These observations related to documentation and review of revisions to the test program as described in the license amendment submittal to the NRC; a calculation referencing an unreviewed and unapproved GE document; and lack of a formal communication mechanism between system engineering and the nuclear fuel group when making changes to design outputs which have potential reactivity control impacts. The inspector reviewed the responses to the observations and found them to be appropriate.

According to nuclear department operating procedure NQAP-QA-400, an annual audit is scheduled to evaluate the programs for training and qualification of operating staff; assessment of corrective actions; and refueling and maintenance activities. The last QA audit on refueling activities and staff training qualification was completed on December 8, 1992. In June 1994, another audit on refueling and special nuclear material was completed, and an exit meeting was conducted on June 27, 1994. Most of the audit deficiencies were corrected by the licensee soon after the audit. However, the inspector found that the licensee did not issue an audit report. This is contrary to 10 CFR 50. Appendix B, Criterion XVIII, which requires that audit results shall be documented and reviewed by management, and is a violation of TS 6.5.2.10.c, which requires audit reports to be forwarded to the Senior Vice President - Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit. When apprised the finding, the licensee immediately directed that the audit be reperformed. This failure constitutes a violation of minor significance and is being treated as a non-cited violation, consistent with Section IV of the enforcement policy (60 FR 34381, June 30, 1995).

Staff Training and Qualification

The inspector reviewed the qualification records of the PP&L nuclear fuel engineering (NFE) staff and noted that they were well trained and knowledgeable of reload safety analyses. All technical staff were required to complete the engineering support program per Nuclear Training Procedure NTP-QA-64.8. In addition to the above program, each of the nuclear fuel engineering staff was required to comply with the NFP-QA-007 qualification. This procedure establishes the qualification requirements for NFE personnel to perform safety-related reload analysis for Susquehanna. The inspector reviewed NFE personnel qualification and certification records; all of the engineering staff have at least a Bachelor's degree in Science and 10 years of nuclear industry experience. The inspector concluded that the nuclear fuel engineering staff met the training and qualification guidelines of ANSI 18.1-1979.

3.0 MANAGEMENT OVERSIGHT

The inspectors concluded that the Susquehanna IST program was well documented and implemented, reflecting a knowledgeable staff and effective management support. Management involvement at the site and corporate offices contributed to timely resolution of an ADS operability concern, and evidenced an appropriate regard for safe operation. The new CR system involved management in the resolution of plant problems at an early stage, and appeared to address previous deficiencies in the area of corrective action. Effective oversight was reflected in EDR backlog reduction efforts, control of vendor engineering activities, and ERC engineering performance reviews. Support for core reload engineering was evident was evidenced by high quality safety analyses. Failure to document the result of a QA audit was considered to be an isolated occurrence.

4.0 MEETINGS

The scope and purpose of the inspection were discussed at entrance meetings conducted on May 30 and June 5, 1995. During the course of the inspection, the findings were discussed periodically with licensee representatives. An exit was conducted on June 9, 1995, at which time the preliminary findings were summarized and conclusions were presented. In acknowledging the findings and conclusions, the licensee took exception to the inspector's position regarding implementation of an IST relief request prior to NRC approval. Some proprietary information was reviewed as part of this inspection (relative to EDR 94-001), however, the details of this information were not included as part of the written inspection report.

The persons listed below were the principle participants in the exit meeting:

Pennsylvania Power & Light Co.

G. Jones, Vice President Nuclear Engineering
G. Miller, Manager Nuclear Technology
H. Palmer, Manager Nuclear Systems Engineering
G. Kuczynski, Manager Nuclear Plant Services
W. Burchill, Manager Nuclear Assessment
T. Dalpiaz, Manager Nuclear Maintenance

Nuclear Regulatory Commission

R. Blough, Acting Deputy Director, Division of Reactor Safety
B. McDermott, Resident Inspector