April 18, 2000

Mr. Robert G. Byram
Senior Vice President - Nuclear
PPL, Inc.
Susquehanna Steam Electric Station
2 North Ninth Street
Allentown, Pennsylvania 18101

SUBJECT: **NRC INTEGRATED INSPECTION REPORT 05000387/2000-002 and 05000388/2000-002**

Dear Mr. Byram:

On April 1, 2000, the NRC completed an inspection at the Susquehanna Steam Electric Station (SSES) Unit 1 & 2 reactor facilities. The enclosed report covered routine activities by the resident inspectors and an announced follow-up inspection of your fire protection program. We concluded that your staff safely operated the facility during this period.

Based on the results of this inspection the NRC has determined that two violations of NRC requirements occurred. These Severity Level IV violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999; (64 FR 61142)). The first NCV involved an inadequate operating procedure which did not allow the supplemental decay heat removal system to be manually restarted following a system trip. The second NCV involved a failure to implement design requirements for primary containment isolation valves in the traversing incore probe system.

If you contest the violation or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region I, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

In accordance with 10CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be placed in the NRC Public Document Room.

A reply to this letter is not required, but should you have any questions regarding this please contact me at 610-337-5233.

Sincerely,

/RA/

Curtis J. Cowgill, Chief
Projects Branch 4
Division of Reactor Projects

Docket Nos: 05000387, 05000388
License Nos: NPF-14, NPF-22

Enclosure: Inspection Report 05000387/2000-002, 05000388/2000-002

cc w/encl:
B. L. Shriver, Vice President - Nuclear Site Operations
G. T. Jones, Vice President - Nuclear Engineering and Support
R. Ceravolo, General Manager - SSES
R. M. Peal, Manager, Nuclear Training
G. D. Miller, General Manager - Nuclear Assurance
R. R. Wehry, Nuclear Licensing - SSES
M. M. Golden, Manager - Nuclear Security
P. Nederostek, Nuclear Services Manager, General Electric
W. H. Lowthert, Manager, Nuclear Plant Services
A. M. Male, Manager, Quality Assurance
H. D. Woodeshick, Special Assistant to the President
G. DallaPalu, PP&L Nuclear Records
R. W. Osborne, Vice President, Supply & Engineering
  Allegheny Electric Cooperative, Inc.
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REGION I

Docket Nos:	05000387, 05000388
License Nos:	NPF-14, NPF-22
Report No.	05000387/2000-002, 05000388/2000-002
Licensee:	PPL, Inc. 2 North Ninth Street Allentown, PA 19101
Facility:	Susquehanna Steam Electric Station
Location:	P.O. Box 35 Berwick, PA 18603-0035
Dates:	February 13, 2000 through April 1, 2000
Inspectors:	S. Hansell, Senior Resident Inspector J. Richmond, Resident Inspector A. Blamey, Resident Inspector K. Young, Reactor Engineer G. Smith, Senior Physical Security Inspector
Approved by:	Curtis J. Cowgill, Chief Projects Branch 4 Division of Reactor Projects

EXECUTIVE SUMMARY

Susquehanna Steam Electric Station (SSES), Units 1 & 2
NRC Inspection Report 05000387/2000-002, 05000388/2000-002

This inspection included aspects of PPL's operations, maintenance, engineering, and plant support at SSES. The report covers a six-week period of routine resident inspection activities and an announced follow-up inspection of your fire protection program by a regional specialist.

Operations

- On March 25, 2000, when the supplemental decay heat removal system was needed to remove spent fuel pool decay heat, the system operating procedure did not provide adequate guidance to restart the supplemental decay heat removal system when it tripped on system low pressure. The failure to establish adequate procedures for this condition is a Severity Level IV violation, which is being treated as a Non-Cited Violation, consistent with section VII.B.1.a of the NRC Enforcement Policy. This violation was documented in PPL's corrective action program as Condition Report 243926. (section O4.2)

- PPL identified that a postulated failure of a non-safety related circuit could result in opening the traversing incore probe system primary containment isolation valves during a loss of coolant accident. PPL determined that this design deficiency did not meet NRC requirements. PPL's interim corrective actions were adequate. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with section VII.B.1.a of the NRC Enforcement Policy. This violation was documented in PPL's corrective action program as Condition Report 97-4106. (section O8.1)

Plant Support

- The inspectors concluded that PPL had performed a detailed evaluation to determine what actions should be taken by the operators to maintain downcomer water level above the top of active fuel for a postulated Appendix "R" fire. PPL had not initiated any changes to procedures or other plant documentation pending NRC resolution of the automatic depressurization system/core spray industry generic issue for boiling water reactors. (section F8.1)

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Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 operated at or near 100% power through the inspection period until March 18. On March 18, Unit 1 was shut down to begin a refueling outage.

SSES Unit 2 operated at 100% power through the inspection period.

I. Operations

O1 Conduct of Operations [1]

O1.1 Unit Operations and Operator Activities (71707)

The inspectors determined routine operator activities were satisfactorily established, communicated, and conservatively performed in accordance with SSES procedures. Control room logs accurately reflected plant activities. During tours of the main control room, the inspectors observed good turn-over briefings and formal communications.

O2 Operational Status of Facilities and Equipment

O2.1 Operational Safety System Alignment (71707)

During routine plant tours, the proper alignment and operability of various safety systems, engineered safety features, and on-site power sources were verified. Partial walkdowns were performed for the emergency diesel generators, safety related batteries, and Unit 1 primary containment.

O2.2 Separation of Instrument Tap on Emergency Service Water System Pipes (71707)

On February 13, an instrument tap (3/8 inch instrument tube with isolation valve, approximately six inches long) was found separated from an emergency service water (ESW) pipe on the “C” emergency diesel generator (EDG). On March 6, a similar instrument tap on the "B" EDG was found separated from an ESW pipe. In each event, no ESW leakage occurred because the instrument tap was blocked with corrosion products. PPL determined that the EDGs were degraded but operable, and that there was no common cause failure. Temporary repairs, not susceptible to the same type of failure, were completed within 48 hours of the second event. PPL’s actions were commensurate with the safety significancy of the event. This item was documented in PPL’s corrective action program as Condition Report 237813. No violation of NRC requirements was identified.

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[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

O4 Operator Knowledge and Performance

O4.1 Unit 1 Reactor Shutdown for Refueling Outage

a. Inspection Scope (71707)

The inspectors observed the planned reactor plant shutdown for a refueling outage to determine whether the activities were conducted in accordance with NRC requirements and SSES procedures.

b. Observations and Findings

On March 17, Unit 1 was shutdown to begin a refueling outage. The inspectors noted that shift operations conducted a thorough reactivity briefing that included a discussion of the potential effects from a degraded control rod drive water pump. The inspectors noted that the shift operators had performed just-in-time simulator training for the plant shutdown evolutions the day prior to the plant shutdown.

The inspectors observed the reactor power reduction from 95% to 47%. The inspector observed good command and control, and consistent 3-way communications. Self checking and peer checking was observed. The inspectors concluded that the activities were conducted in accordance with NRC requirements and SSES procedures.

c. Conclusions

On March 17, 2000, Unit 1 was safely shutdown to begin a refueling outage.

O4.2 Unplanned Loss of the Supplemental Decay Heat Removal System

a. Inspection Scope (71707)

On March 25, the Unit 1 supplemental decay heat removal (SDHR) system unexpectedly tripped. The inspectors reviewed PPL's initial response and corrective actions to determine if activities were conducted in accordance with NRC requirements and SSES procedures.

b. Observations and Findings

On March 25, at 4:23 a.m., a nuclear plant operator identified that the Unit 1 SDHR system had tripped and was no longer removing spent fuel decay heat. The spent fuel decay heat load (Unit 1 reactor core, Unit 1 spent fuel pool, and Unit 2 spent fuel pool) required both the Unit 1 fuel pool cooling system (cooled by SDHR) and the Unit 2 fuel pool cooling system (cooled by Unit 2 service water) to be in operation to maintain fuel pool temperature within administrative limits. Because the Unit 1SDHR was not in operation, the Unit 2 fuel pool cooling system capacity was not sufficient and fuel pool temperature began to increase. Several attempts to restart the SDHR system using operating procedure OP-011-001, “SDHR System Operation,” were unsuccessful. At 5:05 a.m., with additional help from system engineering, the SDHR system was restarted and fuel pool temperature returned to normal.

PPL determined that the cause of the SDHR system trip was due to system low pressure. Operating procedure OP-011-001 did not contain the necessary instructions to restart the SDHR system following a low pressure condition. Technical Specification section 5.4.1(a) requires written procedures be established, implemented, and maintained for items recommended in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33 section 6.i, requires procedures for "Loss of Component Cooling Systems and Cooling to Individual Components." SSES procedure OP-011-001, "SDHR System Operation," did not contain adequate instructions to restart the SDHR system following a system low pressure trip. Failure to establish an adequate procedure is a Severity Level IV violation, which is being treated as a Non-Cited Violation, consistent with section VII.B.1.a of the NRC Enforcement Policy. This violation was documented in PPL's corrective action program as Condition Report 243926. **(NCV 50-387/00-02-01)**

c. Conclusion

On March 25, 2000, when the supplemental decay heat removal system was needed to remove spent fuel pool decay heat, the system operating procedure did not provide adequate guidance to restart the supplemental decay heat removal system when it tripped on system low pressure. The failure to establish adequate procedures for this condition is a Severity Level IV violation, which is being treated as a Non-Cited Violation, consistent with section VII.B.1.a of the NRC Enforcement Policy. This violation was documented in PPL's corrective action program as Condition Report 243926.

O8 Miscellaneous Operations Issues

O8.1 Design Deficiency of the Primary Containment Isolation Valves for the Traversing Incore Probe System

a. Inspection Scope (37551,40500,92700)

The inspectors reviewed PPL's actions for a postulated failure of the traversing incore probe (TIP) system primary containment isolation valves (PCIVs) which had not been previously evaluated. The inspectors also reviewed Licensee Event Report (LER) 50-387/99-008-00.

b. Observations and Findings

PPL identified that a postulated failure of a non-safety related control circuit could result in an unexpected opening of the TIP system PCIVs during a loss of coolant accident, enabling the drywell atmosphere to vent into secondary containment through the TIP indexers and instrument tubes. The TIP PCIVs are safety related ball valves which are spring loaded to close when de-energized and have non-safety related control circuits to energize and open the valves. PPL determined that this postulated failure had not been previously reviewed by either PPL or the NRC. PPL further stated that the design did not meet the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 56, "Primary Containment Isolation."

Although PPL's initial evaluation predicted a primary containment leakage rate of about 8 times higher than the Technical Specification limit of 1% volume per day, the likelihood of such an occurrence is low. PPL estimated the radiological consequences could exceed the design basis accident dose values documented in the SSES Final Safety

Analysis Report (FSAR). PPL's initial corrective actions included compensatory actions to remove electrical power to the non-safety control circuits when the TIP ball valves are closed.

The inspectors performed an in-field review of associated corrective actions, operations activities including TS requirements, FSAR design descriptions, and engineering activities. The inspectors determined this issue was a violation of 10 CFR 50 Appendix B, Criterion III, Design Control (GDC design requirements not translated into design specifications). The inspectors concluded that PPL took effective interim corrective actions. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with section VII.B.1.a of the NRC Enforcement Policy. This violation was documented in PPL's corrective action program as Condition Report 97-4106. LER 50-387/99-008-00 is closed. **(NCV 50-387,388/00-02-02)**

c. Conclusions

PPL identified that a postulated failure of a non-safety related circuit could result in opening the traversing incore probe system primary containment isolation valves during a loss of coolant accident. PPL determined that this design deficiency did not meet NRC requirements. PPL's interim corrective actions were adequate. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with section VII.B.1.a of the NRC Enforcement Policy. This violation was documented in PPL's corrective action program as Condition Report 97-4106.

O8.2 Licensee Event Report (LER) Review (37551,40500,92700)

(Closed) LER 50-387/99-007-00
Incorrect Frequency for Surveillance Test

In November 1999, PPL determined that the Technical Specification (TS) reactor coolant system (RCS) total leakage calculation was being performed every 24 hours instead of every 12 hours. The TS surveillance requirement was changed to from 24 hours to12 hours in October 1998, when improved TS's were issued. The improved TS's changed all three RCS leakage (unidentified leakage, unidentified leakage increase within limits, and total leakage) surveillance requirements to 12 hours. The RCS "unidentified" and "unidentified leakage increase within limits" surveillances were performed satisfactorily every 12 hours as required.

The inspectors concluded that the consequence of calculating the RCS total leakage every 24 hours instead of every 12 hours was of minor safety significance. We verified that procedure SO-100/200-006, "Shiftily Surveillance Operating Log," was changed to require operators to calculate RCS total leakage every 12 hours. PPL's corrective actions appropriately addressed the failure to perform the surveillance at the correct interval. This oversight constitutes a violation of minor significance that is not subject to formal enforcement action. This LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Surveillance and Pre-Planned Maintenance Activity Review

a. Inspection Scope (61726,62707,40500)

The inspectors observed and reviewed selected portions of pre-planned maintenance and surveillance activities, to determine whether the activities were conducted in accordance with NRC requirements and SSES procedures.

b. Observations and Findings

The inspectors observed portions of the following work activities and surveillances:

SM-175-204	24VDC Battery Performance Test
SO-054-003	"C" ESW Quarterly Flow Verification Test
SI-158-303	Reactor Vessel Steam Dome Pressure Switch Calibration
SO-150-004	RCIC HV-150-F045 Stroke Time Verification
SE-259-070	Suppression Pool Purge Exhaust Valves Local Leak Rate Test
SI-199-220	XV-143-F012A, Excess Flow Check Valve Functional Test
SE-159-108	"A" Outboard MSIV Local Leak Rate Test (LLRT)
SM-188-104	1D650 250VDC Battery Modified Performance Discharge Test
SI-178-203	Refueling Outage - Weekly Functional Test of Intermediate Range Monitors Channels A, B, C, D, E, F, G, and H
SO-153-002	Standby Liquid Control System 24-month Injection Flow Test
SE-159-108	LLRT of MSIV Penetration Number X-7A, B, C, D using Main Steam Line Plugs
TP-202-010	2D630 125VDC Battery Post-replacement Testing
TP-134-030	Zone I Recirculation Isolation Damper Inspection
TP-104-013	Bus 1C (1A203) Outage Coordination Procedure
PCWO 103532	"C" ESW Pump Motor Protective Relay Calibration
PCWO 233346	Fire Protection System DSP-021 Rework
PCWO 204085	LIS-B21-2N025D 3-way Valve Replacement
PCWO 102876	HPCI HV-155-F012 Votes Test
PCWO 191014	Installation of Intercell Straps on 2D630
PCWO 208910	2D630 125VDC Battery Replacement
PCWO 224059	"A" RHR Pump Room Unit Cooler Clean and Inspect

In addition, selected portions of procedures and drawings associated with the maintenance and surveillance activities were also reviewed and determined to be acceptable. In general, maintenance personnel were knowledgeable of their assigned activities.

M4 Maintenance Staff Knowledge and Performance

M4.1 Reactor Protection System Instrument Rack

a. Inspection Scope (62707,71707)

On March 21, Unit 2 received an unexpected reactor protection system (RPS) actuation (half-scram). The inspectors reviewed PPL's response to determine whether the work activities were conducted in accordance with NRC requirements and SSES procedures.

b. Observations and Findings

On March 21, following an Instrument and Control (I&C) maintenance activity on the 2C005 instrument rack, an unexpected RPS half-scram was received. PPL determined that the cause of the half-scram was a pressure spike induced during the opening of an instrument isolation valve. The valve manipulation was performed by a nuclear plant operator (NPO) during restoration from a system isolation. This issue was documented in PPL's corrective action program as Condition Report 240822.

The inspectors noted that instrument rack valve manipulations are typically performed by I&C technicians who receive special training intended to reduce the possibility of a pressure spike during valving. The inspectors also noted that NPOs did not receive similar training. PPL's interim corrective actions included a management decision to have only trained I&C technicians operate valves on scram sensitive instruments. The inspectors determined that PPL’s initial response to the half-scram was appropriate. No violation of NRC requirements was identified.

c. Conclusion

On March 21, 2000, PPL experienced an unexpected reactor protection system actuation (half-scram). PPL determined that the cause of the half-scram was a pressure spike induced during the opening of an instrument isolation valve by a nuclear plant operator (NPO). The inspectors determined that PPL’s initial response to the half-scram was appropriate.

IV. Plant Support

F8 Miscellaneous Fire Protection Issues

F8.1 Automatic Depressurization System/Core Spray Post-Fire Safe Shutdown Methodology

a. Inspection Scope (64150,92904)

The inspectors reviewed the portion of the alternate shutdown methodology associated with the automatic depressurization system/core spray (ADS/CS) issue identified at SSES during a fire protection functional inspection. The purpose of this review was to determine if PPL had performed appropriate calculations and modifications to procedures necessary to adhere to the performance goal of maintaining downcomer water level above the top of active fuel (TAF) as stated in 10 CFR 50 Appendix R, Section III.L.2.b.

b. Observations and Findings

During a 1997 fire protection functional inspection, the NRC opened unresolved item 50-387, 388/97-201-03, "Failure of the ADS/CS post-fire safe shut down methodology to meet the performance goal of maintaining the reactor water level above TAF." During additional review of the unresolved item, the inspectors determined that PPL would not be able to maintain downcomer water level above TAF and issued notice of violation (NOV) 98-09-04A for failure to meet their license condition. The NOV was issued on October 19, 1998. In a December 30, 1998, response letter, PPL did not contest the violation and committed to perform a study to determine the efficacy of revising the minimum water level for operator initiation of ADS such that downcomer water level remains above TAF. PPL also committed that upon confirmation that the revised minimum water level meets all pertinent requirements, PPL would initiate appropriate revisions to plant procedures, engineering documentation, licensing documentation, and would implement the requisite operator training. In a March 5, 1999, letter, the NRC acknowledged these corrective actions and stated that when these corrective actions were completed, an inspection of this issue would be scheduled.

In a June 23, 1999, letter, PPL stated that they had completed the required study and determined that the downcomer water level would not drop below TAF in a boildown transient with no high-pressure injection as long as ADS was initiated before wide range indicated level dropped to -91 inches. PPL further stated that subsequent to PPL’s response to the NOV, the issue of using safety relief valves and low pressure systems as an Appendix R, Section III.G.1 and 2 shutdown path had become an industry generic issue for boiling water reactors and that the Boiling Water Reactor Owners Group (BWROG) had submitted a position paper to Nuclear Reactor Regulation (NRR) for review of this issue. Additionally, PPL stated that they did not plan to initiate any actions to revise plant procedures and other documentation until resolution of this industry generic issue was achieved.

The inspectors reviewed calculation EC-THYD-1035, “In-Shroud Level Response for a Boildown Transient,” Revision 2, and found that PPL had performed a detailed evaluation to determine what actions should be taken by the operators to maintain downcomer water level above TAF. The inspectors reviewed the inputs and assumptions of the calculation and found them to be reasonable. The calculation appropriately considered level setpoint for ADS initiation by the operator, calculated fuel clad surface temperature, and void fraction within the shroud when ADS is initiated. The calculation concluded that if the operator initiated ADS before wide range indication reaches -91 inches instead of -129 inches (automatic initiation) or -161.2 inches (manual initiation), downcomer water level would not drop below TAF in a boildown transient with no high-pressure injection. The inspector noted no discrepancies in the calculation and found the conclusion to be reasonable.

Additionally, the inspectors found that PPL had not initiated any actions, as stated in their June 23, 1999, letter, to revise plant procedures and other documentation pending resolution of the industry generic issue regarding ADS/CS. In an April 1, 1999, meeting with NRR, the BWROG requested that the staff stop pursuing outstanding plant specific issues involving the use of safety relief valves and low pressure systems as a post fire safe-shutdown method until this industry generic issue was resolved. NRC senior management agreed with this approach. Based on review of this issue, the NRC postponed action on the industry generic issue pending a final position by NRR.

c. Conclusions

The inspectors concluded that PPL had performed a detailed evaluation to determine what actions should be taken by the operators to maintain downcomer water level above the top of active fuel for a postulated Appendix "R" fire. PPL had not initiated any changes to procedures or other plant documentation pending NRC resolution of the automatic depressurization system/core spray (ADS/CS) industry generic issue for boiling water reactors.

F8.2 Adequacy of Fire Protection Operability Determination

a. Inspection Scope (64150,92904)

The inspectors performed a review of an operability determination for fire detection and suppression systems at SSES to determine the adequacy of PPL's review.

b. Observations and Findings

PPL performed an operability determination of their fire detection and suppression systems as outlined in Condition Report 224274. The condition report was generated in response to findings that occurred as a result of the fire protection functional inspection. PPL had agreed to perform a review of fire detection and suppression systems design and installation at SSES.

Through discussions with PPL's fire protection personnel, review of fire area combustible loading calculations, and review of installed fire detection and suppression systems in selected fire areas, the inspector determined that PPL had performed a detailed operability determination of their fire detection and suppression systems for SSES. The inspector found that in determining the operability of the fire detection and suppression systems, PPL appropriately considered installed combustibles, transient combustibles, fire hazards, fire brigade response, fire protection systems design, and the ability of the plant to achieve and maintain safe shutdown in the event of a fire.

Also, PPL has maintained appropriate compensatory measures in areas of the plant where Thermo-Lag was installed. The inspector checked that the compensatory measures were still in place through a review of recent roving and continuous fire watch logs. The inspector noted that PPL had continued to further evaluate the fire detection and suppression systems at the site and modifications could occur as a result of this effort. The inspector found that this was acceptable.

c. Conclusions

The inspectors concluded that PPL had performed a detailed operability determination for their fire detection and suppression systems. PPL presented evidence that they had appropriate compensatory measures in place for fire areas where Thermo-lag was installed.

S3 Security Program Plans

S3.1 Security Program Plan Revisions

a. Inspection Scope (81700)

The inspectors reviewed changes to the PPL Security Program Plans.

b. Observations and Findings

An in-office review was conducted of changes to the Susquehanna Physical Security and Training and Qualification Plans, identified as Revisions PP and J, respectively, submitted to the NRC in October 1999 and January 2000, in accordance with the provisions of 10 CFR 50.54(p).

c. Conclusion

Based on a limited review of the changes, as described in the plan revisions, no NRC approval of these changes is required, in accordance with 10 CFR 50.54(p). These changes will be subject to future inspection to confirm that the changes, as implemented, have not decreased the overall effectiveness of the security plan.

V. Management Meetings

X1 Exit Meeting Summary

A Region I specialist presented the results of a fire protection program follow-up inspection to members of PPL management at the conclusion of the inspection on March 2, 2000. PPL acknowledged the findings presented.

The inspectors presented the inspection results to members of PPL management at the conclusion of the inspection period, on April 10, 2000. PPL acknowledged the findings presented.

The inspectors asked PPL whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

INSPECTION PROCEDURES USED

IP 37551	Onsite Engineering Observations
IP 40500	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726	Surveillance Observations
IP 62707	Maintenance Observations
IP 64150	Plant Support - Fire Protection Program - Triennial Post-fire Safe Shutdown Capability
IP 71707	Plant Operations
IP 81700	Physical Security Program for Power Reactors
IP 92700	On Site Followup of Reports
IP 92904	Follow-up - Plant Support
IP 93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened NONE

Opened/Closed

50-387/00-02-01	NCV	Unplanned Loss of the Supplemental Decay Heat Removal System (section O4.2)
50-387,388/00-02-02	NCV	Design Deficiency of Traversing Incore Probe Equipment Could Result in Potential Release Path during Design Basis Accident (section O8.1)

Updated NONE

Closed

50-387/99-008-00	LER	Design Deficiency of Traversing Incore Probe Equipment Could Result in Potential Release Path during Design Basis Accident (section O8.1)
50-387/99-007-00	LER	Incorrect Frequency for Surveillance Test (section O8.2)

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
CS	Core Spray
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
GDC	[10 CFR 50, Appendix A] General Design Criteria
HPCI	High Pressure Coolant Injection System
I&C	Instrument and Control
LER	Licensee Event Report
LLRT	Local Leak Rate Test
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NPO	Nuclear Plant Operator
NOV	[NRC] Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PCIV	Primary Containment Isolation Valve
PCWO	Plant Component Work Order
PPL	Pennsylvania Power and Light Company
RCIC	Reactor Core Isolation Cooling System
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
SDHR	Supplemental Decay Heat Removal
SSES	Susquehanna Steam Electric Station
TAF	Top of Active Fuel
TIP	Traversing Incore Probe
TS	Technical Specification