July 21, 2000

EA-00-138
EA-00-142

Mr. G. Rainey, President
PECO Energy
Nuclear Group Headquarters
Correspondence Control Desk
P. O. Box 195
Wayne, Pennsylvania 19087-0195

SUBJECT: PEACH BOTTOM - NRC FIRE PROTECTION INSPECTION REPORT NOS. 05000277/2000-003 AND 05000278/2000-003

Dear Mr. Rainey:

This letter forwards the results of a triennial fire protection team inspection conducted on April 14-18, 2000, at Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3. The preliminary results of the inspection were discussed with Mr. C. Kerr and other members of your staff at a formal exit meeting on April 28, 2000. Changes to the preliminary results were discussed in several subsequent telephone conferences, and ultimately in a formal “re-exit” meeting on June 8, 2000 at the site with Mr. J. Doering and other members of your staff. Additionally, because of the complexity associated with some of the inspection findings, numerous internal discussions were held with NRC technical review staff prior to issuing this report.

During the inspection, the NRC team identified two findings associated with the adequacy of post-fire safe shutdown equipment circuit analyses at the station. Both of these issues were determined to be apparent violations of Sections III.G. and III.L. of 10 CFR 50 Appendix R. In the first case, the team found that you had not evaluated the potential for mechanical damage to safe shutdown equipment caused by fire-induced circuit failures, stating your position that such failures are not within the design and licensing basis of the facility. This position is contrary to the NRC staff’s established position regarding this matter, as articulated in Information Notice 92-18 and other generic industry correspondence. Regarding the second finding, your analyses for fire-induced circuit faults assumed that only one spurious actuation would occur for each system affected by any one fire. Again, this methodology conflicts with the NRC’s established position (documented in Generic Letter 86-10 and other public records) that multiple spurious actuations must be assumed and evaluated. For the fire areas evaluated during this inspection, the team assessed the significance of this issue as Green.

It is our understanding that you do not consider either of these two issues to be violations of 10 CFR 50 or your operating license. Additionally, we recognize that other commercial nuclear power plant operators, as represented by the Nuclear Energy Institute (NEI), have adopted a similar position regarding these issues. As such, in accordance with our current enforcement policy for these matters as described in Enforcement Guidance Memorandum (EGM) 98-002 Revision 2, “Disposition of Violations of Appendix R, Sections III.G. and III.L. Regarding Circuit Failures,” the NRC will defer any further enforcement action relative to these issues until the staff evaluates NEI’s proposed resolution methodology.

The deferral of enforcement action provided in EGM 98-002 is in part based on your implementation of “reasonable compensatory measures” until such time that the resolution methodology is developed and approved. We reviewed the compensatory measures that you established subsequent to the team’s discovery of the noted issues and found them to be acceptable for the interim period.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC’s document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket Nos. 05000277, 05000278
License Nos. DPR-44, DPR-56

Enclosure: NRC Inspection Report 05000277/2000003 and 05000278/2000003

cc w/encl:
J. Hagan, Senior Vice President, Nuclear Operations
J. Doering, Vice President, Peach Bottom Atomic Power Station
G. Johnston, Plant Manager, Peach Bottom Atomic Power Station
J. A. Hutton, Director, Licensing, PECO Nuclear
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J. H. Walter, Chief Engineer, Public Service Commission of Maryland
Mr. & Mrs. Dennis Hiebert, Peach Bottom Alliance
Mr. & Mrs. Kip Adams
Commonwealth of Pennsylvania
State of Maryland
TMI - Alert (TMIA)

Distribution w/encl:
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NRC Resident Inspector
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	05000277, 05000278
License Nos:	DPR-44, DPR-56
Report No:	50-277/00-03, 50-278/00-03
Licensee:	PECO Energy Company
Facility:	Peach Bottom Atomic Power Station, Units 2 & 3
Location:	1848 Lay Road Delta, PA 17314
Dates:	April 24-28, 2000
Inspectors:	R. Fuhrmeister, Sr. Reactor Inspector, Division of Reactor Safety (DRS) K. Young, Reactor Inspector, DRS C. Cahill, Reactor Inspector, DRS G. Morris, Reactor Inspector, DRS K. Sullivan, Contract Engineering Support
Approved By:	S. Morris, Acting Chief, Electrical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Peach Bottom Atomic Power Station, Units 2 & 3
NRC Inspection Report 05000277/2000-003, 05000278/2000-003

This report covers a one week baseline team inspection of post-fire safe shutdown capability at PBAPS. The inspection team conducted a detailed review of a sample of fire barriers (including penetrations seals and electrical raceway cables), emergency lighting, programmatic controls for transient combustible materials and potential ignition sources, and post-fire safe shutdown equipment circuit analyses. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process described in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- GREEN. PECO's specification for performing circuit analyses of post-fire safe shutdown equipment stipulates that only one spurious actuation for each system affected by any one fire be analyzed. For the areas inspected, the team determined that PECO adequately protected against fire-induced spurious actuations. The team did not identify any additional spurious actuations which would have prevented achieving safe shutdown conditions in the post-fire operating environment.

  The assumption that only a single spurious actuation need be considered for any one system for any one fire is an apparent violation of the requirements of Section III.G. and III.L. of Appendix R to 10 CFR 50. PECO entered this issue into their corrective action program and have implemented reasonable compensatory measures. However, the issue of multiple spurious actuations of equipment in a post-fire environment is in contention between the NRC and the nuclear industry. As such, any further enforcement action will be deferred pending final resolution of this issue by the Nuclear Energy Institute and the NRC staff, in accordance with Enforcement Guidance Memorandum 98-02, Revision 2, issued February 2, 2000. (Section 1RO5.5.b(2))

- NO COLOR. PECO adopted a licensing position that mechanical damage to alternative shutdown equipment resulting from fire-induced cable faults, as described in Information Notice 92-18, was outside the scope of the licensing and design bases of the facility. As a result, PECO did not evaluate the control circuits of the alternative shutdown equipment to determine if it was susceptible to this problem. Since a detailed review of the alternative shutdown capability at PBAPS was not performed as part of the scope of this inspection, the risk associated with this issue was not established.

  This issue is being treated as an apparent violation of Condition 2.C.4 of the operating licenses for both Unit 2 and Unit 3, which requires PECO to implement and maintain the fire protection program described in the NRC Safety Evaluation Reports. PECO has entered this issue into their corrective action program and has implemented reasonable compensatory measures pending final resolution of the issue. However, the issue of mechanical damage to safe shutdown equipment due to fire-induced cable faults is in contention between the NRC and the nuclear industry. As such, any further enforcement action will be deferred pending final resolution of this issue by the Nuclear Energy Institute and the NRC staff, in accordance with Enforcement Guidance Memorandum 98-02, Revision 2, issued February 2, 2000. (Section 1RO5.5b.(1))

Report Details

Summary of Plant Status:

Both units operated at or near 100% power for the duration of the inspection.

REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems

1R05 Fire Protection

1. Fire Barrier Penetration Seals

a. Inspection Scope

During plant tours, the team randomly selected four fire barrier penetration seals for detailed inspection to verify proper installation and qualification. The team reviewed associated design drawings, Brand Industrial Supply Corporation qualification test reports, Component Record List entries, and Penetration Seal Deviation No. 10 and its associated Factory Mutual Research Corporation test report. The team compared the observed in-situ seal configurations to the design drawings and tested configurations. The team also compared the penetration seal ratings with the ratings of the barriers in which they were installed.

b. Issues and Findings

There were no findings identified.

2. Raceway Fire Barrier Systems

a. Inspection Scope

The team reviewed PBAPS Engineering Change Request PB-99-02551-000 to determine which raceways required encapsulation for protection of post-fire safe shutdown capability, reviewed a listing of the codes of record for the sprinkler systems installed to protect encapsulated raceways, and verified the field condition of encapsulated raceways in fire areas 6N and 50. The team also observed the material condition of sprinkler systems in fire areas 6N, 13N, and 50. This review was performed to verify the adequacy of protection for post-fire safe shutdown equipment.

b. Issues and Findings

There were no findings identified.

3. Post-Fire Safe Shutdown Emergency Lighting

a. Inspection Scope

The team observed the placement and aim of emergency lighting units (ELUs) throughout the selected fire areas to evaluate their adequacy for illuminating access and egress pathways and any equipment requiring local operation for post-fire shutdown. In addition, the team witnessed a “blackout” test performed by PECO personnel in a switchgear room to demonstrate the adequacy of the installed ELUs.

b. Issues and Findings

There were no findings identified.

4. Programmatic Controls

a. Inspection Scope

During tours of the facility, the team observed the material condition of fire protection systems and equipment, the storage of permanent and transient combustible materials, and the control of ignition sources. This review was performed to evaluate the implementation of fire protection program administrative controls.

b. Issues and Findings

There were no findings identified.

5. Post-Fire Safe Shutdown Circuit Analyses

a. Inspection Scope

The team reviewed several piping and instrumentation drawings, system schematics, cable raceway drawings and wiring diagrams associated with systems required for post-fire safe shutdown to determine what equipment and components were needed for post-fire operation of the credited safe shutdown systems. These systems included the reactor core isolation cooling system, high pressure coolant injection system, residual heat removal/low pressure coolant injection system, automatic depressurization system, reactor vessel pressure and level instrumentation, and torus temperature instrumentation.

The team also reviewed PECO’s post-fire safe shutdown analyses for fire areas 6N, 13N, and 50. The team examined cable routing through these fire areas to confirm that circuits running through these areas were not required to achieve safe shutdown for a fire in that area. These reviews were conducted to verify that PECO had identified safety-related and non-safety related cables in the selected fire areas and had analyzed their configuration to show that they would not prevent post-fire safe shutdown because of hot shorts, open circuits, or shorts to ground.

The team conducted reviews of selected electrical power buses to verify that protective device coordination existed for equipment needed to conduct post-fire safe shutdown activities. The team also examined PBAPS's multiple high impedance fault (MHIF) analyses to determine if upstream electrical bus protective devices would be challenged if MHIF's occurred on cables supplying power to post-fire safe shutdown equipment. The team reviewed the electrical isolation capability of equipment needed for post-fire safe shutdown to ensure that such equipment could be operated locally if needed. Lastly, the team evaluated the adequacy of fire barriers for cables associated with post-fire safe shutdown equipment.

b. Issues and Findings

(1) Protection of Alternative Shutdown Equipment from Mechanical Damage Caused by Fire-Induced Cable Faults

Summary

The team discovered that PBAPS adopted a licensing position that the mechanical damage scenario described in NRC Information Notice (IN) 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire," was outside the design and licensing basis of the facility. As a result, PBAPS personnel did not evaluate the impact that this concern may have on their facility with regard to post-fire safe shutdown.

Background

IN 92-18 alerted licensees to conditions found at several reactor facilities that could result in the loss of capability to maintain the reactor in a safe shutdown condition in the event of a fire which forced operators to evacuate the control room. Specifically, the IN described the potential for control room fires causing "hot shorts" (i.e., short circuits between control wiring and power sources) for certain motor-operated valves (MOVs) needed for safe shutdown. The IN noted that hot shorts could cause valve/component damage during the period of time between when operators evacuated the control room and reached local controls at a remote/alternate shutdown panel. This uncontrolled MOV operation could ultimately result in mechanical damage to the valves or other components, rendering them incapable of performing their required post-fire safe shutdown function.

The team reviewed PECO specification NE-00296, "Specification for Post-Fire Safe Shutdown Program Requirements at Peach Bottom Atomic Power Station," Revision 0. Paragraph 6.3.2 states that the postulated mechanical damage phenomenon described in IN 92-18 was beyond the licensing and design bases of the PBAPS facility. The basis for this determination was a qualitative risk analysis in which PECO concluded that the probability of the phenomenon was low, and that its contribution to overall plant risk was negligible. This assessment is discussed in detail in supporting document NE-296-2, "PECO Final Position on NRC IN 92-18 Documentation & Correspondence." This evaluation states, "The time in which it has to occur is considered to be ten (10) minutes, i.e., the time it takes to abandon the control room and transfer control at the alternate control station." NE-296-2 further states, "It is highly improbable that damage that severe would occur in the first ten minutes of a fire." The inspectors noted that this position conflicts with the results of a 1987 fire test conducted by PECO and reported in test report NE-296-1, "Appendix R Multiple High Impedance Cable Fault Flame Test

Report," Revision 0." Specifically, Figure No. 13, "Test No. 3, Load Supply Fuse Current vs. Time During Flame Test," shows that within the first ten minutes of fire exposure, three separate fire-induced faults occurred, with durations of up to 54 seconds. Fault currents in excess of five amps were also documented. These current levels, for these durations, could be sufficient to cause MOV control circuits to actuate unexpectedly and inappropriately.

Enforcement Implications

PECO's failure to evaluate and ensure that alternative shutdown equipment is protected from the adverse effects of a control room fire appears to be a violation of Condition 2.C.4 of the operating licenses for both Unit 2 and Unit 3. This condition requires PECO to implement and maintain the fire protection program approved in the NRC Safety Evaluation Reports. Additionally, Section III.G.3 of Appendix R to 10 CFR 50 requires that alternative shutdown capability, independent of the fire area of concern, shall be provided when it cannot be demonstrated that at least one train of the redundant shutdown capability will remain free of fire damage. Section III.L of Appendix R provides the performance criteria to be met for the minimum shutdown capability provided to comply with Section III.G.3. Section III.L.7 states, in part, that "The safe shutdown equipment and systems for each fire area shall be known to be isolated from associated non-safety circuits in the fire area so that hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment."

PECO entered this issue into their corrective action program under Performance Enhancement Program Item No. I0011294. As compensatory measures, pending final resolution of the issue, PECO issued notifications to both plant operators and fire brigade members regarding this potential susceptibility. In addition, this issue was added to the continuing training program for the operators and fire brigade members. Further, PECO also evaluated the combustible material and ignition source control program guidance, as well as the fire detection and suppression systems and fire watch guidance to ensure their adequacy. The inspectors verified that these measures were in place. As such, this issue falls within the purview of the NRC's Enforcement Guidance Memorandum (EGM) 98-02, Revision 2, issued February 2, 2000. This EGM, titled "Disposition of Violations of Appendix R, Sections III.G and III.L Regarding Circuit Failures," permits deferral of further enforcement action pending final resolution of the issue by the Nuclear Energy Institute and the NRC Office of Nuclear Reactor Regulation. (**AV 05000277; 05000278/2000-003-01**)

(2) Number of Spurious Actuations Assumed to Occur for any One Postulated Fire

Summary

During a review of PECO specification NE-00296, "Post-Fire Safe Shutdown Program Requirements at Peach Bottom Atomic Power Station, " the team determined that Section 5.1.5, "Spurious Operations," subsection 3, "Non-high/low pressure interface devices," required that only one spurious actuation per system per fire be analyzed. It further stated, "Any number of hot shorts, open circuits, or shorts to ground may occur, but they will result in only one single spurious actuation." This position is contrary to the NRC staff interpretation of the fire protection requirements of Appendix R as explained in Generic Letter (GL) 86-10, and reiterated in a March 11, 1997 letter from the NRC to the Nuclear Energy Institute.

Risk Assessment

For the fire areas reviewed in this inspection, the team determined that PECO had implemented adequate measures to protect post-fire shutdown capability from the effects of spurious actuations. The team did not identify any instances in which spurious actuations would have prevented achieving safe shutdown conditions in a post-fire environment. As a result, this finding was evaluated as being of very low risk significance (Green). The team did not evaluate this issue for the other fire areas in the plant since they were not within the scope of this inspection. On that basis, the potential risk significance of this issue as it may apply to other fire areas was not evaluated.

Enforcement Implications

The NRC staff interpretation of the requirements of Appendix R to 10 CFR 50, as explained in the guidance in GL 86-10, is that all possible spurious actuations which could be caused by a fire must be considered. The staff further interpreted that guidance to require that each spurious actuation be considered, one at a time, for its potential to adversely affect the ability to achieve safe shutdown conditions.

However, this matter is also one of the generic issues being addressed by the nuclear industry; and, as such, it also falls under the purview of EGM 98-02, Revision 2. Accordingly, further enforcement action will be deferred pending final resolution of this issue. (**AV 05000277; 05000278/2000-003-02**)

(3) Multiple High Impedance Faults (MHIFs)

Summary

The team determined that the MHIF resolution methodology for electrical power distribution panels, approved in an NRC Safety Evaluation Report (SER), dated April 11, 1989, and subsequently incorporated into the PBAPS operating license, was not the method proposed and subsequently utilized at the station.

Background

At a March 15, 1989 meeting with NRC staff, PECO representatives proposed a three-phase methodology for evaluating the susceptibility of post-fire safe shutdown equipment power supplies to MHIF-induced trips. Phase 1 was comprised of two steps. The first step consisted of a comparison of the high impedance fault (HIF) current (defined as the summation of the 1000-second trip currents of the load circuit protection devices) to the 1000-second trip setting of the source protective device. The second step compared the HIF current summation to the 60-second trip setting of the source protective device. Phase 2, for those power supplies which did not screen out in Phase 1, considered only the post-fire safe shutdown equipment circuits on the power supplies. Phase 2 used the same two step evaluation process and criteria as Phase 1. This phase confirmed the ability to re-energize power supplies using manual bus restoration procedures developed for the plants. Phase 3 was intended to eliminate as many of the restoration procedures as practical, and consisted of reviewing the cable routing for all loads on each of the power supply busses. Specifically, only those cables traversing a fire area were considered to be faulted and thus contributed to HIF currents. The screening criteria used in Phase 3 were the same as the other two phases.

The team verified that the above described methodology was actually employed at PBAPS in part by reviewing calculation PE-0006, "Address Multiple High Impedance Faults Due to An Appendix R Fire," Revision 19. The team noted an assumption made in the calculation that a power supply panel would be considered to remain available if the sum of the HIF current values was less than either the 1000- or 60-second trip current of the source protective device. However, the team discovered that the NRC safety evaluation, dated April 11, 1989, which approved PECO's MHIF analytical methodology conflicts with what was actually conducted. Specifically, the SER states that the second step of each phase is a comparison of the 60-second trip settings (vice 1000-second) of the load circuit protective devices to the 60-second trip settings of the source protective device. As a result, the basis for the operating license requirement as established by the noted SER is not clear. This issue will remain unresolved pending a technical evaluation of the adequacy of the actual methodology PECO used by the NRC's Office of Nuclear Reactor Regulation staff. (**URI 05000277; 05000278/2000-003-03**)

OTHER ACTIVITIES

4OA4 Other

1. Corrective Actions for Fire Protection Deficiencies

a. Inspection Scope

The team reviewed the Fire Impairments Log, a list of corrective maintenance action requests against post-fire safe shutdown equipment, and selected action requests for post-fire safe shutdown equipment to evaluate the effectiveness of corrective actions and the prioritization for resolving fire protection related deficiencies.

b. Issues and Findings

No findings were identified.

4OA5 Management Meetings

Exit Meeting Summary

The inspectors presented their preliminary inspection results to Mr. C. Kerr and other members of the PECO Energy staff at an exit meeting on April 28, 2000.

Changes to the preliminary results were discussed by telephone on June 2, 2000, and at a "re-exit" meeting conducted at the site on June 8, 2000, with Mr. J. Doering and other members of the PECO Energy staff.

The inspectors asked whether any materials examined during the inspection should be considered proprietary. All information marked as proprietary was returned to PECO Energy at the end of the inspection.

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

PARTIAL LIST OF PERSONS CONTACTED

PECO Energy

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M. Warner, Plant Manager
G. Johnston, Engineering Director
I. Seddon, Nuclear Quality Assurance Manager
A. Winter, Experience Assessment manager
D. Henry, Operations Support Manager
A. Piha, Design Engineering Manager
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M. Taylor, Licensing Engineer
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S. Beck, Experience Assessment Engineer
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D. Pendleton, Nuclear Quality Assurance Assessor

Nuclear Regulatory Commission

B. Holian, Deputy Director, Division of Reactor Safety (DRS)
S. Morris, Acting Chief, DRS Electrical Engineering Branch
T. McMurtray, Senior Resident Inspector
M. Buckley, Resident Inspector
L. Mitchell, Intern

Brookhaven National Laboratory

K. Sullivan

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-277&278/00-03-01	AV	Apparent Violation of 10 CFR 50 Appendix R Sections III.G and III.L requirement to protect alternative shutdown equipment from the adverse effects of a fire. (Section 1R05.5.b(1))
50-277&278/00-03-02	AV	Apparent Violation of 10 CFR 50 Appendix R Sections III.G. and III.L. that multiple spurious actuations must be considered in the fire analyses. (Section 1R05.5.b(2))
50-277&278/00-03-03	URI	Multiple high impedance fault analysis methodology different from that approved by NRC. (Section 1R05.5.b(3))

Closed

None

Discussed

None

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
EGM	Enforcement Guidance Memorandum
ELU	Emergency Lighting Unit
HIF	High Impedance Fault
IN	Information Notice
IPEEE	Individual Plant Evaluation for External Events
LER	Licensee Event Report
MHIF	Multiple High Impedance Fault
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PBAPS	Peach Bottom Atomic Power Station
PECO	PECO Energy Company
RCIC	Reactor Core Isolation Cooling
SER	Safety Evaluation Report
TS	Technical Specification

LIST OF DOCUMENTS REVIEWED

Piping and Instrumentation Drawings

M-351, Sheet 1, Rev. 66	Nuclear Boiler
M-351, Sheet 2, Rev. 63	Nuclear Boiler - Unit 2 & Common
M-359, Sheet 1, Rev. 45	Reactor Core Isolation Cooling System
M-359, Sheet 2, Rev. 45	Reactor Core Isolation Cooling System
M-360, Sheet 1, Rev. 51	RCIC Pump Turbine Details
M-360, Sheet 2, Rev. 50	RCIC Pump Turbine Details
M-361, Sheet 1, Rev. 73	Residual Heat Removal System
M-361, Sheet 2, Rev. 65	Residual Heat Removal System
M-362, Sheet 1, Rev. 60	Core Spray Cooling System
M-362, Sheet 2, Rev. 60	Core Spray Cooling System
M-365, Sheet 1, Rev. 56	High Pressure Coolant Injection System
M-365, Sheet 2, Rev. 60	High Pressure Coolant Injection System
M-366, Sheet 1, Rev. 53	HPCI Pump Turbine Details

Control Circuit Schematics

M-1-S-23,	Sheet 3, Rev. 104	Primary Containment Isolation System
	Sheet 16, Rev. 97	
	Sheet 17, Rev. 98	
	Sheet 29, Rev. 99	
M-1-S-36,	Sheet 5, Rev. 74	High Pressure Coolant Injection System (System No. 23)
	Sheet 6, Rev. 74	
	Sheet 7, Rev. 74	
	Sheet 8, Rev. 73	
	Sheet 9, Rev. 73	
	Sheet 15, Rev. 77	
M-1-S-42,	Sheet 1, Rev. 77	Reactor Core Isolation Cooling System (System 13)
	Sheet 5, Rev. 80	
	Sheet 6, Rev. 75	
	Sheet 7, Rev. 75	
	Sheet 9, Rev. 75	
	Sheet 10, Rev. 74	
	Sheet 11, Rev. 74	
	Sheet 12, Rev. 74	
	Sheet 13, Rev. 74	
	Sheet 17, Rev. 74	
	Sheet 18, Rev. 75	

M-1-S-65,	Sheet 1, Rev. 27 Sheet 9, Rev. 96 Sheet 10, Rev. 96 Sheet 11, Rev. 96 Sheet 12, Rev. 97 Sheet 13, Rev. 96 Sheet 14, Rev. 96 Sheet 27, Rev. 101 Sheet 31, Rev. 100 Sheet 36A, Rev. 97 Sheet 36B, Rev. 99 Sheet 36C, Rev. 82	Residual Heat Removal System (System Number 10)
E-1,	Sheet 1, Rev. 38	Electrical Single Line Diagram
E-26,	Sheet 1, Rev. 75 Sheet 2, Rev. 56 Sheet 8, Rev. 16	Single Line Diagram 125/250 VDC System Unit 2
E-365,	Sheet 1, Rev. 50 Sheet 2, Rev. 48	Primary Containment Isolation System
E-611,	Sheet 1, Rev. 28	4160 V Switchgear
E-613,	Sheet 1, Rev. 32	4160 V Switchgear
E-1615,	Sheet 1, Rev. 60	Single Line Meter & Relay Diagram E124 & E224 Emerg.L.C; E124-R-C
E-1617,	Sheet 1, Rev. 49	Single Line Meter & Relay Diagram E324 & E424 Emerg.L.C; E324-R-B
E-2903,	Sheet 1, Rev. 4 Sheet 2, Rev. 1	Alternative Control Instrumentation

<u>Calculations</u>

EE-0007, Rev. 8	10CFR50 Appendix R Electrical Coordination Study Units 2 & 3
PE-0006, Rev. 19	Multiple High Impedance Faults due to an Appendix R Fire
PE-0191, Rev. 2	125/250 VDC System Fault Current Calculation
PE-0193, Rev. 3	480 V Load Center/MCC, Coordination Study for
PE-0194, Rev. 2	Coordination for 4.16kV 1E Switchgear
PE-0195, Rev. 3	120Vac Coordination Study
PE-0196, Rev. 2	125/250VDC System Coordination
PE-0205, Rev. 1	Load Study for the Station Aux. Power System

PEAF-0001, Rev.0	Flow Path Analysis for Development of INDMS Fire Safe Shutdown Database
PEAF-0002, Rev. 0	S/U Source Loading for Fire Safe Shutdown

<u>Safe Shutdown Analyses</u>

FPP, Rev. 11	Fire Protection Program
NE-00296, Rev. 0	Post-Fire Safe Shutdown Program Specification
PF-0016-06N, Rev. 0	Fire Area 6N Fire Safe Shutdown Analysis
PF-0016-13N, Rev. 0	Fire Area 13N Fire Safe Shutdown Analysis
PF-0016-50N, Rev. 0	Fire Area 50N Fire Safe Shutdown Analysis

<u>Procedures</u>

NE-C-250, Rev. 2	Fire Protection & Fire Safe Shutdown Review
NE-C-320, Rev.3	Cable and Raceway Management System
NE-CG-931, Rev. 4	Fuse & Molded Case Circuit Breaker Application Design Guide
M-C-700-230, Rev. 5	480 Volt ABB/ITE Load Center Breaker Maintenance
M-C-700-232, Rev.6	Testing and Control of 600 Volt Class Molded Case Circuit Breakers and Setpoints
T-306N-2 , Rev. 0	Unit 2 RB General Area North, El. 135'

<u>Modification Documents</u>

PESP-00-001D	Fire Protection and Fire Safe Shutdown Issues
ECR-PB-99-02551-000, MOD P00680	Update E-2471 for Encapsulation

Corrective Action Program Documents

A1200549 Potential for Sump Room Flooding due to Appendix R Hi/Lo Pressure Interface

A1215668 Appendix R Fire Impact on Unit 2 Vacuum Breaker MO-2-23B-4245

A1200833 Potential for Sump Room Flooding due to Appendix R Hi/Lo Pressure Interface

A1085813 Evaluate Deletion of T-300 Series Multiple HIF Procedures, Address Issues Contained in Associated Evaluation

A1263254 Two Cables w/o Fire Resistive Coating in CFZ, RB3 135' West

A1207645 Door #283 - Entrance to U/2 RFPT Lube Oil Rm. 150 el.

A1207321 Door #233

A1246669 CSR Fire Damper did Not Actuate Upon Simulated CO2 Injection

A1137159 Lack of Fire Detection in Areas w/Safety related Equipment

I0011294 Triennial Fire Protection Inspection Issues