50-277/278

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 7, 1995

Mr. Richard Ochs
Maryland Safe Energy Coalition
P.O. Box 33111
Baltimore, MD 21218

Dear Mr. Ochs:

In William T. Russell's letter of December 2, 1994, the NRC acknowledged receipt of your press release of October 6, 1994, in which you requested that the NRC (1) immediately shut down both reactors at Peach Bottom until the risk of fire near electrical control cables due to combustible insulation is corrected; (2) suspend the Peach Bottom license until an analysis of the synergistic effects of cracks in multiple parts is conducted; (3) immediately shut down both reactors at Peach Bottom until all safety class component parts in both reactor vessels, including the cooling system, the heat transfer system and the reactor core, are inspected; and (4) immediately shut down both reactors at Peach Bottom pending correction of numerous equipment problems identified in recent NRC inspection reports. In his letter, Mr. Russell stated that your press release was being treated as a petition in accordance with 10 CFR 2.206 of the NRC's regulations. In addition, Mr. Russell denied your requests for immediate action and indicated that the remaining issues raised in the petition would be addressed within a reasonable time.

I have been assigned petition manager for this petition. Part of my responsibility as petition manager is to provide you with periodic updates of our review of the petition. This letter is to advise you of the status of our review.

As a result of the indictments against Thermal Science Incorporated (TSI), which you referenced in item (1) of your press release, and the concerns regarding information provided by TSI and others, the staff is reevaluating all technical actions that were previously based on that information. The NRC staff issued a letter to PECO Energy Company (PECO, the licensee) on December 29, 1994 requesting additional information on the Thermo-Lag barriers installed at Peach Bottom in order to evaluate these concerns. A copy of the December 29, 1994 letter is provided as Enclosure 1. I will provide you with copies of all future correspondence between the NRC staff and PECO Energy Company regarding the use of Thermo-Lag at Peach Bottom Atomic Power Station.

In his December 2, 1994 letter, Mr. Russell discussed the recent core shroud inspections at Peach Bottom Units 2 and 3. Subsequently, the staff completed an in-depth review of the Unit 2 and Unit 3 shroud inspection results. In order to provide you with updated information on items (2) and (3) of your press release, I have included the staff's safety evaluations on the licensee's core shroud inspection results as Enclosure 2. I will provide you with copies of future correspondence between the NRC staff and PECO Energy Company regarding core shroud and reactor vessel internal components issues at Peach Bottom.

Mr. Russell also addressed your concerns regarding the operability of the emergency service water system on August 3, 1994. Subsequent to the August 3, 1994 event, the staff conducted an enforcement conference with the licensee on

DF01

DISTRIBUTION: Letter to Mr. Richard Ochs, Dated: March 7, 1995

Docket File 50-277/50-278
PUBLIC
PDI-2 Reading
S. Varga (w/o attachments)
J. Zwolinski (w/o attachments)
J. Stolz
J. Shea
M. O'Brien
C. Anderson, RGN I
J. Longo, OGC
M. Gamberoni
C. Norsworthy (w/o attachments)
R. Cooper, RGN I
M. Thadani

October 18, 1994. Following the conference, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalty by letter dated November 21, 1994. The November 21, 1994 letter was provided to you via the NRC Region I correspondence distribution list. By letter dated December 21, 1994, the licensee responded to the Notice of Violation. In a separate correspondence, also dated December 21, 1994, the licensee agreed to pay the civil penalty in the amount proposed. As discussed in Mr. Russell's December 2, 1994 letter, the staff considered the emergency service water system restored to an operable status on August 3, 1994. I have included copies of both of the December 21, 1994 letters from PECO Energy Company to the NRC as Enclosures 3 and 4.

Please feel free to contact me, as the petition manager, at (301) 415-1428, if you have any questions. I have enclosed a brochure on the NRC's 10 CFR 2.206 process (Enclosure 5). I will provide you with additional periodic updates while the staff prepares its final response to your petition. Finally, I have placed you on our distribution list for documents related to the issues in your petition.

Sincerely,
/S/
Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Letter from R. Zimmerman, NRC, to G. Hunger, PECO, dated December 29, 1994
2. Letter from J. Shea, NRC, to G. Hunger, PECO, dated February 6, 1995
3. Letter from D. Smith, PECO, to Director, Office of Enforcement, NRC, dated December 21, 1994
4. Letter from G. Hunger, PECO, to J. Lieberman, NRC, dated December 21, 1994
5. NUREG/BR-0200, "Public Petition Process"

cc w/enclosure 5:
Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

* Previous Concurrence

OFFICE	PDI-2/LA	PDI-2/PM	OGC	PDI-2/PD		
NAME	MO'Brien	JShea	GLongo *	JStolz		
DATE	2/ /95	2/21/95	02/21/95	2/22/95		

OFFICIAL RECORD COPY DOCUMENT NAME: C:\PEACH\OCH.UPD

October 18, 1994. Following the conference, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalty by letter dated November 21, 1994. The November 21, 1994 letter was provided to you via the NRC Region I correspondence distribution list. By letter dated December 21, 1994, the licensee responded to the Notice of Violation. In a separate correspondence, also dated December 21, 1994, the licensee agreed to pay the civil penalty in the amount proposed. As discussed in Mr. Russell's December 2, 1994 letter, the staff considered the emergency service water system restored to an operable status on August 3, 1994. I have included copies of both of the December 21, 1994 letters from PECO Energy Company to the NRC as Enclosures 3 and 4.

Please feel free to contact me, as the petition manager, at (301) 415-1428, if you have any questions. I have enclosed a brochure on the NRC's 10 CFR 2.206 process (Enclosure 5). I will provide you with additional periodic updates while the staff prepares its final response to your petition. Finally, I have placed you on our distribution list for documents related to the issues in your petition.

Sincerely,

Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Letter from R. Zimmerman, NRC, to G. Hunger, PECO, dated December 29, 1994
2. Letter from J. Shea, NRC, to G. Hunger, PECO, dated February 6, 1995
3. Letter from D. Smith, PECO, to Director, Office of Enforcement, NRC, dated December 21, 1994
4. Letter from G. Hunger, PECO, to J. Lieberman, NRC, dated December 21, 1994
5. NUREG/BR-0200, "Public Petition Process"

cc w/enclosure 5:
Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 29, 1994

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: FOLLOWUP TO THE REQUEST FOR ADDITIONAL INFORMATION REGARDING GENERIC LETTER 92-08, ISSUED PURSUANT TO 10 CFR 50.54(f), PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3, AND LIMERICK GENERATING STATION, UNIT NOS. 1 AND 2, (TAC NOS. M85586, M85587, M85565 and M85566)

Dear Mr. Hunger:

In response to U.S. Nuclear Regulatory Commission (NRC) staff requests for information regarding Generic Letter (GL) 92-08, "Thermo-Lag 330-1 Fire Barriers," you indicated that PECO Energy Company planned to continue to rely on Thermo-Lag fire barriers to comply with NRC fire protection regulations. On September 29, 1994, the U.S. Attorney for the District of Maryland and the NRC Inspector General (IG) announced the indictment of Thermal Science, Incorporated (TSI), the company that manufactures and supplies Thermo-Lag fire barrier materials, and its president, Mr. Rubin Feldman. The indictment alleges that TSI and Mr. Feldman conspired with Industrial Testing Laboratories, Incorporated (ITL), and others to make false statements and conceal material facts within the jurisdiction of the NRC and to defraud the United States by impeding, impairing, obstructing, and defeating the NRC's administration of the Atomic Energy Act. ITL had pleaded guilty in U.S. District Court in Maryland in April 1994.

In a letter of November 7, 1992, TSI informed the staff that preshaped Thermo-Lag conduit sections received by Texas Utilities Electric Company (TU Electric) for Comanche Peak Steam Electric Station, Unit 2 (CPSES 2) showed signs of delamination and voids. The NRC staff was concerned that the use of such materials could affect the results of TU Electric's fire tests and the performance of the Thermo-Lag barriers installed at CPSES 2. In a letter of December 15, 1992, TU Electric described the actions it had taken to ensure that the fire barrier materials used in its fire test program were representative of the materials installed at CPSES 2, and described how it had addressed the delamination and void concerns. On the basis of its evaluation of the TU Electric submittal, the staff concluded that the fire test specimens were representative of the materials installed at CPSES 2 and that TU Electric had adequately addressed the delamination and void concerns. The IG has informed the staff that TSI may not have implemented certain measures to correct the void and delamination problems even though it had informed TU Electric that it had done so. Specifically, we believe that TSI representatives informed TU Electric that it had trained its employees to

Enclosure 1

~~9501060071~~

repair the delaminations, cracks, and voids and that it had provided TU Electric with signed training certificates to document this training. In fact, we believe that TSI may not have trained its employees to perform these repairs. This situation calls into question the reliability of TSI's quality assurance program for Thermo-Lag materials, and the quality of Thermo-Lag materials.

The NRC staff has considered the effect of the indictment on the plans of NRC staff and industry to resolve the technical issues associated with Thermo-Lag fire barriers. In my letter of September 20, 1994, I informed you that the Commission was requiring that all plants with Thermo-Lag fire barriers return to compliance with existing NRC fire protection regulations. The indictment does not alter this decision. Licensees planned to use information and data supplied by TSI to demonstrate that Thermo-Lag fire barrier installations conform to NRC regulations. However, the concerns and issues underlying the indictment and the TU Electric experience sharpened concerns previously expressed by the NRC staff to the licensees about the reliability of information and data supplied by TSI that have been or could be used to make judgments regarding Thermo-Lag materials. Therefore, the staff will request licensees to take actions to fully address the technical issues discussed in GL 92-08, independent of information and data supplied by TSI, before the staff makes any determination regarding whether the use of Thermo-Lag fire barriers complies with NRC regulations.

The NRC staff and industry have relied on the results of tests and analyses conducted by NRC staff and industry to draw conclusions regarding the performance of Thermo-Lag fire barrier materials. However, such conclusions require that the materials tested be representative of the broad class of material actually installed in the plant. Judgments regarding representativeness, in turn, require reasonable assurance that appropriate quality assurance measures were taken in the manufacture of the Thermo-Lag materials or, alternatively, that the licensees determine that the properties and quality of the materials are appropriate for their applications and satisfy the staff that the determinations are correct. On the basis of the concerns underlying the indictment and the TU Electric experience, the staff has determined that reliance should not be placed on TSI's quality assurance program for the purpose of assessing the adequacy of Thermo-Lag materials that are currently installed or that are installed in the future. The staff has also concluded that it is not enough for licensees to rely on generic information on Thermo-Lag materials. The licensee must also have valid information on the specific Thermo-Lag materials installed at its plant if it intends to retain or expand its Thermo-Lag fire barrier installations.

The staff previously addressed the uniformity of Thermo-Lag materials in Section II, "Important Barrier Parameters," of the request for additional information (RAI) of December 1993 regarding Generic Letter 92-08. In Section II of the RAI, the staff stated:

> [B]ecause of questions about the uniformity of the Thermo-Lag fire barrier materials produced over time, NUMARC [now Nuclear Energy

Institute] stated in its letter of July 29, 1993, that "[c]hemical analysis of Thermo-Lag materials provided for the program, as well as samples from utility stock, will be performed, and a test report prepared comparing the chemical compositions of the respective samples." The results of the chemical analyses may indicate that variations in the chemical properties of Thermo-Lag are significant and may require additional plant-specific information in the future.

Where the licensees plan to rely on fire endurance test results to draw conclusions regarding the qualifications of specific Thermo-Lag fire barrier installations, such conclusions require that installed materials and configurations be representative of tested materials and configurations. This, in turn, requires that the installation parameters for the tested configuration bounded the installation parameters of the in-plant configuration and that appropriate quality assurance measures were taken in the manufacture of the Thermo-Lag materials, and the construction of the test specimen and the in-plant fire barrier. In Section II of the RAI of December 1993, the staff listed 24 important fire barrier installations parameters and eight important cable parameters. At least two of the parameters, panel thickness and conduit panels, are controlled by TSI at the point of manufacture. Other parameters, such as panel rib orientation, tie-wire spacing, and proximity of cables to the unexposed surfaces of the fire barrier, are determined during barrier design and construction. The remaining parameters, such as cable size and type, are established by plant design. After the RAI was issued, many licensees informed the staff that they had not verified some of the parameters and several licensees reported deviations and defects in fire barrier installations that were revealed only after destructive examination of in-plant Thermo-Lag fire barriers. The staff informed licensees of installation deficiencies found at Enrico Fermi Atomic Power Plant, Unit 2, in Information Notice 92-79, Supplement 1, "Deficiencies Found in Thermo-Lag Fire Barrier Installation," August 4, 1994. Later, Grand Gulf Nuclear Station reported installation deficiencies found during destructive fire barrier examinations (Licensee Event Report 94-008).

On the basis of its inspections of Thermo-Lag fire barriers and industry experience finding installation defects during destructive examinations, the staff has concluded that some of the installation parameters discussed in Section II of the RAI of December 1993 cannot be verified or determined by simple walkdowns of in-plant barriers, or by comparing as-built barriers with installation records or with the installation procedures used to construct the barriers. The staff has also concluded that some of the parameters can only be obtained and verified by detailed examination such as disassembling a representative sample of in-plant fire barrier configurations. The licensee must have valid and verifiable information on each of the parameters for its in-plant Thermo-Lag barriers if it intends to retain, modify, or expand its Thermo-Lag fire barrier installations.

The NRC staff and licensees have also relied on information, data, and calculations supplied by TSI to draw conclusions regarding the seismic capabilities of Thermo-Lag materials and barriers. These conclusions are also being reevaluated by the staff.

You are required, pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), to submit written reports, under oath or affirmation, that contain the information specified in the enclosure to this letter in Sections 1.a, 1.b, 1.c, 2.a, 2.b, and 2.c, within 90 days from the date of this letter. Retain on site all information and documentation used to prepare your response; these may be reviewed during future NRC audits or inspections. You are also reminded of the following GL 92-08 reporting requirement: "When corrective actions have been completed, confirm in writing their completion."

The information collection contained in this request is covered by the Office of Management and Budget clearance number 3150-0011, which expires on July 31, 1997. The public reporting burden for this collection of information is covered by the previous estimate of 420 person-hours plus an increase of 120 person-hours, for a total of 540 person-hours for each addressee's response. This includes time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

If you have any questions about this matter, please contact Joseph W. Shea at 301-504-1428, Frank Rinaldi at 301-504-1447 or Edward Connell at 301-504-2838.

Sincerely,

Steven A. Varga for

Roy P. Zimmerman
Associate Director for Projects
Office of Nuclear Reactor Regulation

Docket No. 50-277/50-278
50-352/50-353

Enclosure: As stated

cc w/encl: See next page

Mr. George A. Hunger, Jr.
PECO Energy Company

Peach Bottom Atomic Power Station,
Units 2 and 3
Limerick Generating Station,
Units 1 and 2

cc:

J. W. Durham, Sr., Esquire
Sr. V.P. & General Counsel
PECO Energy Company
2301 Market Street, S26-1
Philadelphia, Pennsylvania 19101

PECO Energy Company
ATTN: Mr. G. R. Rainey, Vice President
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

PECO Energy Company
ATTN: Regulatory Engineer, A4-5S
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

Resident Inspector
U.S. Nuclear Regulatory Commission
Peach Bottom Atomic Power Station
P.O. Box 399
Delta, Pennsylvania 17314

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Roland Fletcher
Department of Environment
201 West Preston Street
Baltimore, Maryland 21201

A. F. Kirby, III
External Operations - Nuclear
Delmarva Power & Light Company
P.O. Box 231
Wilmington, DE 19899

Mr. Rich R. Janati, Chief
Division of Nuclear Safety
Pennsylvania Department of
Environmental Resources
P.O. Box 8469
Harrisburg, Pennsylvania 17105-8469

Board of Supervisors
Peach Bottom Township
R. D. #1
Delta, Pennsylvania 17314

Public Service Commission of Maryland
Engineering Division
Chief Engineer
6 St. Paul Centre
Baltimore, MD 21202-6806

Mr. Richard McLean
Power Plant and Environmental
Review Division
Department of Natural Resources
B-3, Tawes State Office Building
Annapolis, Maryland 21401

Mr. David P. Helker, 62A-1
Manager - Limerick Licensing
PECO Energy Company
965 Chesterbrook Boulevard
Wayne, Pennsylvania 19087-5691

Mr. David R. Helwig, Vice President
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

Mr. Robert Boyce
Plant Manager
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

Mr. George A. Hunger, Jr.
PECO Energy Company

Peach Bottom Atomic Power Station,
Units 2 and 3
Limerick Generating Station,
Units 1 and 2

Mr. Craig L. Adams
Superintendent - Services
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. James L. Kantner
Manager - Experience Assessment
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. Larry Hopkins
Superintendent-Operations
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. James A. Muntz
Superintendent - Technical
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

Mr. Neil S. Perry
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 596
Pottstown, PA 19464

Library
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. John Doering, Chairman
Nuclear Review Board
PECO Energy Company
965 Chesterbrook Boulevard
Mail Code 63C-5
Wayne, Pennsylvania 19087

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

REQUEST FOR ADDITIONAL INFORMATION REGARDING
GENERIC LETTER 92-08
"THERMO-LAG 330-1 FIRE BARRIERS"
PURSUANT TO 10 CFR 50.54(f)

1. Thermo-Lag Materials

   a. Describe the specific tests and analyses that will be performed to verify that the Thermo-Lag fire barrier materials that are currently installed at Peach Bottom Atomic Power Station (PBAPS) and Limerick Generating Station (LGS), or that will be installed in the future, are representative of the materials that were used to address the technical issues associated with Thermo-Lag barriers and to construct the fire endurance and ampacity derating test specimens. The tests and analyses shall address the material properties and attributes that were determined or controlled by TSI during the manufacturing process and the quality assurance program. The tests and analyses shall also address the material properties and attributes that contribute to conclusions that the Thermo-Lag materials and barriers conform to NRC regulations. These include:

      (1) chemical composition
      (2) material thickness
      (3) material weight and density
      (4) the presence of voids, cracks, and delaminations
      (5) fire endurance capabilities
      (6) combustibility
      (7) flame spread rating
      (8) ampacity derating
      (9) mechanical properties such as tensile strength, compressive strength, shear strength, and flexural strength.

   b. Describe the methodology that will be used to determine the sample size and demonstrate that the sample size will be large enough to ensure that the information and data obtained will be sufficient to assess the total population of in-plant Thermo-Lag barriers and the materials that will be installed in the future. In determining the sample size, consider the time of installation and manufacture of the various in-plant materials and barrier installations. Give the number and types (e.g., panels, conduit preshapes, trowel-grade material, stress skin) of samples that will be tested or analyzed.

   c. Submit the schedule for verifying the Thermo-Lag materials.

   d. After the analyses and tests have been completed, submit a written supplemental report that confirms that this effort has been completed and provide the results of the tests and analyses. Describe any changes to previously submitted plans or schedules that result from the tests or analyses.

ENCLOSURE

2. Important Barrier Parameters

   a. Describe the examinations and inspections that will be performed to obtain the important barrier parameters given in Section II of the RAI of December 1993 for the Thermo-Lag fire barrier configurations installed at PBAPS and LGS.

   b. Describe the methodology that will be applied to determine the number and type of representative in-plant fire barrier configurations that will be examined in detail and demonstrate that the sample size is adequate to ensure that the information and data that will be obtained are adequate to assess the total population of in-plant Thermo-Lag barriers. A large enough sample of the total population of configurations should be examined to provide reasonable assurance that the materials and important barrier parameters used to construct the in-plant barriers and any future barrier installations or modifications, are representative of the parameters used to construct the fire endurance test specimens.

   c. Submit the schedule for obtaining and verifying all of the important barrier parameters.

   d. After the information has been obtained and verified, submit a written supplemental report that confirms that this effort has been completed and provides the results of the examinations and inspections. Verify that the parameters of the in-plant configurations are representative of the parameters of the fire endurance test specimens. Describe any changes to previously submitted plans or schedules that result from the examinations.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 6, 1995

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: GENERIC LETTER (GL) 94-03, "INTERGRANULAR STRESS CORROSION CRACKING OF CORE SHROUDS IN BWRs," PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3, (TAC NOS M90105 AND M90106)

Dear Mr. Hunger:

By letter dated August 24, 1994, the PECO Energy Company (PECo) provided its response to Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs," for the Peach Bottom Atomic Power Station, Units 2 and 3. The NRC staff requested in GL 94-03 that licensee's take the following actions with respect to their core shrouds: (1) inspect their core shrouds in their BWR plants no later than the next refueling outage; (2) perform materials-related and plant-specific consequence safety analyses with respect to their core shrouds; (3) develop core shroud inspection plans which address inspection of all core shroud welds and which takes into account the latest available inspection technology; (4) develop plans for evaluation and/or repair of their core shrouds; and (5) work closely with the BWR Owners Group with respect to addressing intergranular stress corrosion cracking of BWR internals.

The NRC staff requested that licensee's submit, under oath or affirmation, the following information in response to GL 94-03 within 30 days of the date of issuance: (1) a schedule for inspection of their core shrouds; (2) a safety analysis, including a plant-specific safety analysis as appropriate, which supports continued operation of the facility until inspections are conducted; (3) a drawing(s) of the core shroud configurations; and (4) a history of shroud inspections completed to date. The NRC staff also requested that licensee's submit, under oath or affirmation, no later than 3 months prior to performing their core shroud inspections, their scope for inspection of their core shrouds and their plans for evaluating and/or repairing their core shrouds based on their inspection results. The NRC staff further requested licensee's to submit, under oath or affirmation, their core shroud inspection results and flaw evaluation within 30 days of completing their shroud examinations.

Based on the staff's review of PECo's August 24, 1994, response to GL 94-03, and in regard to the information that was requested to be submitted within 30 days of the date of issuance of the GL, the staff concludes that PECo has provided the operational, fabrication and materials related information requested for both the Peach Bottom Units 2 and 3.

Enclosure 2

~~9502090005~~

The staff notes that PECo has previously examined the Peach Bottom Unit 3 (PBAPS 3) core shroud during refueling outage (RFO) 3R09. Based on the results of the materials-based structural analysis of the PBAPS 3 core shroud, the staff concludes that the structural margins for the PBAPS 3 core shroud will be maintained during the current PBAPS 3 operating cycle (Unit 3, Cycle 10). The staff therefore concludes that the results of the licensee's materials-based structural analysis are sufficient to justify continued safe operation for the remainder of the current PBAPS 3 operating cycle without necessitating a detailed consequence safety analysis or a modification of the PBAPS 3 core shroud. The staff's evaluation of PECo's GL 94-03 reponse for Unit 3 is provided as Enclosure 1.

Per the reporting requirements of GL 94-03, the licensee is reminded that, for inspection scope and shroud evaluation/repair information that has been requested but not yet been submitted (i.e., PBAPS Unit 3), the inspection scope and evaluation/repair scope information should be submitted within 3 months of performing their scheduled core shroud inspections. TAC M90106 will remain open for Peach Bottom Unit 3 pending submittal of the shroud inspection/repair plans.

The staff also notes that PECo has recently completed the Peach Bottom Unit 2 (PBAPS 2) core shroud examinations, which were performed during the recently completed refueling outage 2R010, and which were performed per the actions requested by GL 94-03. The staff has received the November 7, 1994 submittal containing the results and evaluation of the PBAPS 2 core shroud examinations which were performed during RFO 2R10. The results of the PBAPS 2 shroud inspections indicate that the cracks in the PBAPS 2 core shroud are bounded by those recorded for PBAPS 3, and are therefore acceptable for service during the next operating cycle (Operating Cycle 11). The staff's evaluation of PECo's GL 94-03 reponse for Unit 2 is provided as Enclosure 2. Staff action for PECO's response to GL 94-03 for Unit 2 is completed and TAC M90105 is closed.

Sincerely,

Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-277/278

Enclosures: As stated

cc w/encls: See next page

cc:

J. W. Durham, Sr., Esquire
Sr. V.P. & General Counsel
PECO Energy Company
2301 Market Street, S26-1
Philadelphia, Pennsylvania 19101

PECO Energy Company
ATTN: Mr. G. R. Rainey, Vice President
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

PECO Energy Company
ATTN: Regulatory Engineer, A4-5S
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

Resident Inspector
U.S. Nuclear Regulatory Commission
Peach Bottom Atomic Power Station
P.O. Box 399
Delta, Pennsylvania 17314

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Roland Fletcher
Department of Environment
201 West Preston Street
Baltimore, Maryland 21201

A. F. Kirby, III
External Operations - Nuclear
Delmarva Power & Light Company
P.O. Box 231
Wilmington, DE 19899

Mr. Richard Ochs
Maryland Safe Energy Coalition
P.O. Box 33111
Baltimore, MD 21218

Mr. Rich R. Janati, Chief
Division of Nuclear Safety
Pennsylvania Department of
Environmental Resources
P. O. Box 8469
Harrisburg, Pennsylvania 17105-8469

Board of Supervisors
Peach Bottom Township
R. D. #1
Delta, Pennsylvania 17314

Public Service Commission of Maryland
Engineering Division
Chief Engineer
6 St. Paul Centre
Baltimore, MD 21202-6806

Mr. Richard McLean
Power Plant and Environmental
Review Division
Department of Natural Resources
B-3, Tawes State Office Building
Annapolis, Maryland 21401

Mr. John Doering, Chairman
Nuclear Review Board
PECO Energy Company
965 Chesterbrook Boulevard
Mail Code 63C-5
Wayne, Pennsylvania 19087

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RESPONSE TO GENERIC LETTER 94-03

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

PECO ENERGY COMPANY

DOCKET NO. 50-277

1.0 BACKGROUND

The core shroud in a Boiling Water Reactor (BWR) is a stainless steel cylindrical component within the reactor pressure vessel (RPV) that surrounds the reactor core. The core shroud serves as a partition between feedwater in the reactor vessel's downcomer annulus region and the cooling water flowing up through the reactor core. In addition, the core shroud provides a refloodable volume for safe shutdown cooling and laterally supports the fuel assemblies to maintain control rod insertion geometry during operational transients and accidents.

In 1990, crack indications were observed at core shroud welds located in the beltline region of an overseas BWR. This reactor had completed approximately 190 months of power operation before discovery of the cracks. As a result of this discovery, General Electric Company (GE), the reactor vendor, issued Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud Crack Indications," on October 3, 1990, to all owners of GE BWRs. The RICSIL summarized the cracking found in the overseas reactor and recommended that at the next refueling outage plants with high-carbon-type 304 stainless steel shrouds perform a visual examination of the accessible areas of the seam welds and associated heat-affected zone (HAZ) on the inside and outside surfaces of the shroud.

Subsequently, a number of domestic BWR licensees performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic inspection experience. Of the inspections performed to date, significant cracking was reported at several plants. The combined industry experience from these plants indicates that both axial and circumferential cracking can occur in the core shrouds of GE designed BWRs.

On July 25, 1994, the NRC issued Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," to all BWR licensees (with the exception of Big Rock Point, which does not have a core shroud) to address the potential for cracking in their core shrouds. GL 94-03 requested BWR licensees to take the following actions with respect to their core shrouds:

- inspect their core shrouds no later than the next scheduled refueling outage;

Enclosure 1

9502090006

- perform a safety analysis supporting continued operation of the facility until the inspections are conducted;
- develop an inspection plan which addresses inspections of all shroud welds, and which delineates the examination methods to be used for the inspections of the shroud, taking into consideration the best industry technology and inspection experience to date on the subject;
- develop plans for evaluation and/or repair of the core shroud; and
- work closely with the BWROG on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to intergranular stress corrosion cracking.

The PECO Energy Company (PECo), the licensee for the Peach Bottom Atomic Power Station Unit 3 (PBAPS 3), responded to GL 94-03 on August 24, 1994 (Reference 1). Part of the licensee's response included PECo's inspection scope for the planned re-inspections of the PBAPS 3 core shroud, which have been scheduled for refueling outage (RFO) 3R10 in the fall of 1995. The licensee completed an inspection of the PBAPS 3 core shroud during the previous RFO in the fall of 1993. The General Electric Nuclear Energy Division formally submitted the examination results and assessment of core shroud structural integrity to the NRC by letter dated December 3, 1993 (Reference 2). PECo amended the results and assessment by letter dated March 14, 1994 (Reference 3).

2.0 STAFF'S EVALUATION OF THE LICENSEE'S RESPONSE TO GL 94-03

PECo completed a limited visual inspection of the PBAPS 3 core shroud during the 3R9 RFO in the fall of 1993. The licensee has planned a more comprehensive inspection of the PBAPS 3 core shroud for the next RFO, scheduled for the fall of 1995.

2.1 Susceptibility of the PBAPS 3 Core Shroud to IGSCC

The core shroud cracks which are the subject of GL 94-03, result from intergranular stress corrosion cracking (IGSCC) which is most often associated with sensitized material near the component welds. IGSCC is a time-dependent phenomena requiring a susceptible material, a corrosive environment, and a tensile stress within the material.

Industry experience has shown that austenitic stainless steels with low carbon content are less susceptible to IGSCC than stainless steels with higher carbon content. BWR core shrouds are constructed from either type 304 or 304L stainless steel. Type 304L stainless steel has a lower carbon content that type 304 stainless steel. During the shroud fabrication process when the sections of the core shroud are welded together, the heating of the material adjacent to the weld metal sensitizes the material. Sensitization involves carbon diffusion out of solution forming carbides at grain boundaries upon moderate heating. The formation of carbides at the grain boundaries depletes the chromium in the adjacent material. Since the corrosion resistance of stainless steel is provided by the presence of chromium in the material, the area adjacent to the grain boundary depleted of chromium is thereby susceptible to corrosion. Increased material resistance to IGSCC will result if the carbon content is kept below 0.035%, as specified for type 304L grade material.

Currently available inspection data indicate that shrouds fabricated with forged ring segments are more resistant to IGSCC than rings constructed from welded plate sections. The current understanding for this difference is related to the surface condition resulting from the two shroud fabrication processes. Welded shroud rings are constructed by welding together arcs machined from rolled plate. This process exposes the short transverse direction in the material to the reactor coolant. Elongated grains and stringers in the material exposed to the reactor coolant environment are believed to accelerate the initiation of IGSCC.

Water chemistry also plays an important role in regard to IGSCC susceptibility. Industry experience has shown that plants which have operated with a history of high reactor coolant conductivity have been more susceptible to IGSCC than plants which have operated with lower conductivities[1]. Furthermore, industry experience has shown that reactor coolant systems (RCSs) which have been operated at highly positive, electrochemical potentials (ECPs) have been more susceptible to IGSCC than RCSs that have been operated at more negative ECPs[2]. The industry has made a considerable effort to improve water chemistry at nuclear facilities over the past 10 years. Industry initiatives have included the introduction of hydrogen water chemistry as a means of lowering ECPs (i.e., making the ECPs more negative) in the RCS. The effectiveness of hydrogen water chemistry in reducing the susceptibility of core shrouds to IGSCC initiation has not been fully evaluated; however, its effectiveness in reducing IGSCC in recirculation system piping has been demonstrated.

Welding processes can introduce high residual stresses in the material at the

---

[1]Conductivity is a measure of the anionic and cationic content of liquids. As a reference, the conductivity of pure water is ~0.05 μs/cm. Reactor coolants with conductivities below 0.20 us/cm are considered to be relatively ion free; reactor coolants with conductivities above 0.30 μs/cm are considered to have a relatively high ion content.

[2]The electrochemical potential (ECP) is a measure of a material's susceptibility to corrosion. In the absence of an externally applied current, and therefore, for reactor internals in the RCS, the electrochemical potential is equal to the open circuit potential of the material. Industry experience has shown that crack growth rates in reactor internals are low when the ECP $\leq$ --0.230 volts.

weld joint. The high stresses result from thermal contraction of the weld metal during cooling. A higher residual tensile weld stress will increase the material's susceptibility to IGSCC. Although weld stresses are not easily quantified, previous investigation into weld stresses indicate that tensile stresses on the weld surface may be as high as the yield stress of the material. The stress decreases to compressive levels in the center of the welded section.

PECo has reviewed the materials, fabrication and operational histories of the PBAPS 3 core shroud and has submitted this information to the staff in their response to GL 94-03. The PBAPS 3 plant-specific susceptibility factors are summarized below:

- The shroud support, top guide support, and core support plate rings are fabricated from two welded 304 stainless steel, forged ring segments, with carbon contents of ~0.030%. The shroud shell region was fabricated by welding rolled 304 stainless steel plates together. The carbon content of the PBAPS 3 shroud plates are in the range of 0.050 - 0.065%.

- Welding of the shroud plates and rings for circumferential welds H1 - H6 was accomplished by submerged arc welding using ER308 filler metal. Welding of the bi-metallic weld, H7, was accomplished by gas metal arc welding using filler metal 82. Weld residual stress levels resulting from these fabrication processes are high.

- PBAPS 3 operated at high reactor coolant ionic content levels during the initial years of operation. The initial five year average coolant conductivity for PBAPS 3 was 0.695 μS/cm, which is considerably higher than the average for other U.S. BWRs (where the conductivities range from ~0.123 μS/cm to 0.717 μS/cm, and average ~ 0.340 μS/cm).

- PBAPS 3 has operated for 11 cumulative years at full power, which is slightly above the median for U.S BWRs (range is 3.7 years - 17.8 years, with a median of 10.8 years).

A review of the plant-specific factors which increase the potential for IGSCC in BWR core shrouds reveals that PBAPS 3 initially operated at high reactor coolant conductivity during the first five cycles of operation. In addition, the carbon content of the material which comprises the PBAPS 3 core shroud is relatively high. On these bases, the Boiling Water Reactor Vessels & Internals Project (BWRVIP) has classified the PBAPS 3 core shroud as a susceptible Category "C" shroud. The staff has also determined that the PBAPS 3 core shroud is susceptible to IGSCC, and therefore concludes that the BWRVIP's susceptibility assessment is acceptable. This conclusion is supported by the identification of moderate cracking during the previous core shroud inspection. This is discussed further in the following section.

2.2 Inspection of the Peach Bottom Unit 3 Core Shroud

PECo inspected the PBAPS 3 core shroud during RFO 3R9 in the fall of 1993. The staff previously reviewed the licensee's evaluation of the PBAPS 3 core shroud and determined that the licensee's assessment justified continued operation of PBAPS 3 for the current operating cycle (Operating Cycle 10). The staff's assessments of the licensee's inspection scope and flaw evaluation are provided in References 4 and 5 listed under Section 5.0 of this Safety Evaluation (SE). The following is a description and staff assessment of the licensee's core shroud inspection.

2.2.1 Inspection Scope and Results for Core Shroud Examinations

The inspections completed during RFO 3R9 were done in accordance with recommendations of SIL-572, Revision 1. The scope of the inspections included examination via enhanced VT-1 methods. The licensee initially completed a partial examination of the core shroud circumferential welds. Their original inspection scope required enhanced VT-1 examinations at eight (8) cell locations of the H1, H2, H3, H4, and H5 welds. The licensee expanded the inspection scope after discovering indications at the H3 and H4 welds.

The expanded scope included the following examinations:

- 100% enhanced VT-1 from the inside diameter (ID) of the H3 and H4 welds;
- 100% enhanced VT-1 of accessible areas of weld H4 on the outside diameter (OD);
- enhanced VT-1 examinations of the H3 weld from the OD in areas where cracking was not indicated on the ID;
- an enhanced VT-1 examination of the H3 weld from the OD in areas where cracking was indicated on the ID;
- enhanced VT-1 examinations at six (6) locations of the H6 weld;
- enhanced VT-1 examinations at two (2) locations of the respective H7 and H8 welds;
- enhanced VT-1 examination of one (1) vertical weld between the H3 and H4 welds; and
- enhanced VT-1 examination of the of the mid-shroud plates.

The licensee's VT-1 examinations identified a large (~105 inch) crack in the H3 weld (the weld joining the top guide support ring to the upper mid-shroud shell). Less extensive cracking was also found at the H4 weld (< 30 inches total). Minor cracking was determined to exist at weld H1 and at one of the vertical shroud welds.

2.2.2 Evaluation of the Peach Bottom 3 Core Shroud Inspection Results

PECo's evaluation and disposition of the inspection data was the basis for justifying operation of the PBAPS 3 Unit during the current operating cycle (Cycle 10). PECo issued a preliminary draft on the Peach Bottom Unit 3 core shroud flaw evaluation during the PECo/NRC meeting of November 3, 1993, at Rockville, Maryland. PECo formally submitted this flaw evaluation to the staff on December 3, 1993 (Reference 2), and amended it on March 14, 1994 (Reference 3). The licensee's flaw evaluation was performed in accordance with the methods found in General Electric (GE) Document GENE-523-141-1093, "Evaluation and Screening Criteria for the Peach Bottom Unit 3 Shroud Indications," Rev. 0 (Reference 2) and Rev. 1 (Reference 3). The licensee's submittal included the results of the PBAPS 3 core shroud inspections performed during the previous RFO.

Flaw evaluations of the PBAPS 3 shroud were performed in accordance with the structural margin criteria found in Section XI of the ASME Code. Evaluations of the indications of the PBAPS 3 core shroud, which included adjustments to account for crack proximities, crack growth and non-destructive examination uncertainties, indicated that the PBAPS 3 core shroud would maintain sufficient structural integrity for the current operating cycle (Operating Cycle 10).

2.2.3 Staff Assessment of the Peach Bottom Unit 3 Inspection and Evaluation

The staff concluded (References 4 and 5), after reviewing PECo's inspection scope for the VT-1 examinations, that the inspection scope was sufficient to ascertain the condition of the PBAPS 3 shroud. The staff also concluded (References 4 and 5) that the licensee's flaw evaluation method was acceptable and that PBAPS 3 core shroud would meet structural margin requirements during the current operating cycle. PECo is required by GL 94-03 to submit its inspection scope for re-inspection of the PBAPS 3 core shroud 90 days prior to entering the fall 1995 RFO.

3.0 CONCLUSIONS

Based on a review of the PBAPS 3 core shroud materials, fabrication processes and operating history, the staff concludes that the licensee's core shroud is susceptible to IGSCC. PECo completed an examination of the PBAPS 3 core shroud during RFO 3R9. The licensee's assessment of identified weld cracking indicates that the PBAPS 3 core shroud will maintain sufficient structural margins throughout the current operating cycle. The staff concluded that the licensee's flaw evaluation of the PBAPS 3 core shroud was acceptable and justified operation of the PBAPS 3 reactor for the current operating cycle (References 4 and 5).

4.0 OUTSTANDING ISSUES/FUTURE ACTIONS

In accordance with the reporting requirements of GL 94-03, the licensee shall submit to the NRC, no later than 3 months prior to performing the core shroud inspections, both the inspection plan and the licensee's plans for evaluating and/or repairing of the shroud based on the inspection results. In addition, results should be provided to the NRC within 30 days from the completion of the inspection. If the licensee identifies any core shroud cracking requiring an analysis per the ASME code, details of such evaluations must also be submitted to the NRC for review.

It should be noted that the industry is currently encountering difficulties performing comprehensive inspections of lower shroud welds and /or lower vessel regions due to NDE equipment accessibility problems. The staff urges licensees to work with the members of the EPRI NDE Center in order to develop improved tooling for inspections of shroud welds and lower vessel regions which are highly obstructed. Should improved inspections techniques become available, the staff recommendation is for licensee's to re-inspect the lower shroud welds at the earliest opportunity.

At present, the NRC has not approved the inspection guidelines proposed by the BWRVIP. Considerable differences remain with regard to the recommended scope of core shroud inspections. The staff cautions the licensee against modifying their plans according to BWRVIP recommendations which have not undergone review and approval by the NRC. The staff's current position with regard to the scope of inspections is a recommendation for the inspection of 100% of the accessible core shroud welds. Should the licensee opt to install a preemptive repair in lieu of performing a comprehensive core shroud inspection the only required inspection is that mandated in the staff approval of the repair option.

5.0 REFERENCES

1. Letter from G. A. Hunger. Jr., Director of Licensing, PECO Energy Company, to the U.S. Nuclear Regulatory Commission forwarding the "Peach Bottom Atomic Power Station, Units 2 and 3, Limerick Generating Station Units 1 and 2 Response to Generic Letter 94-03, 'Intergranular Stress Corrosion Cracking of Core Shroud in Boiling Water Reactors,'" dated August 24, 1994.

2. Letter from M. L. Herrera and H. Mehta, General Electric Nuclear Energy, to the U.S. Nuclear Regulatory Commission forwarding the General Electric "Evaluation and Screening Criteria for the Peach Bottom Unit-3 Shroud Indications," Rev. 0, (GENE-523-141-1093) dated December 3, 1993.

3. Letter from G. A. Hunger, Jr., Director of Licensing, PECO Energy Company, to the U.S. Nuclear Regulatory Commission forwarding the General Electric "Evaluation and Screening Criteria for the Peach Bottom Unit-3 Shroud Indications," Rev. 1, (GENE-523-141-1093) dated March 14, 1994.

4. NRC internal memorandum from Jack R. Strosnider, Chief, Materials and Chemical Engineering Branch, Division of Engineering, to Larry E. Nicholson, Acting Director, Project Directorate, I-2, Division of Reactor Projects I/II, forwarding staff's "Evaluation of Peach Bottom Shroud Cracks," dated November 9, 1993.

5. Stephen Dembek, Project Manager, Project Directorate I-2, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation, issuance of "Meeting Summary, Evaluation of Core Shroud Indications at Peach Bottom, Unit 3 (TAC No. M88099)," dated December 2, 1993.

Principal Contributor: J. Medoff

Date: February 6, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RESPONSE TO GENERIC LETTER 94-03

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

PECO ENERGY COMPANY

DOCKET NO. 50-277

1.0 BACKGROUND

The core shroud in a Boiling Water Reactor (BWR) is a stainless steel cylindrical component within the reactor pressure vessel (RPV) that surrounds the reactor core. The core shroud serves as a partition between feedwater in the reactor vessel's downcomer annulus region and the cooling water flowing up through the reactor core. In addition, the core shroud provides a refloodable volume for safe shutdown cooling and laterally supports the fuel assemblies to maintain control rod insertion geometry during operational transients and accidents.

In 1990, crack indications were observed at core shroud welds located in the beltline region of an overseas BWR. This reactor had completed approximately 190 months of power operation before discovery of the cracks. As a result of this discovery, General Electric Company (GE), the reactor vendor, issued Rapid Information Communication Services Information Letter (RICSIL) 054, "Core Support Shroud Crack Indications," on October 3, 1990, to all owners of GE BWRs. The RICSIL summarized the cracking found in the overseas reactor and recommended that at the next refueling outage plants with high-carbon-type 304 stainless steel shrouds perform a visual examination of the accessible areas of the seam welds and associated heat-affected zone (HAZ) on the inside and outside surfaces of the shroud.

Subsequently, a number of domestic BWR licensees performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic inspection experience. Of the inspections performed to date, significant cracking was reported at several plants. The combined industry experience from these plants indicates that both axial and circumferential cracking can occur in the core shrouds of GE designed BWRs.

On July 25, 1994 the NRC issued Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," to all BWR licensees (with the exception of Big Rock Point, which does not have a core shroud) to address the potential for cracking in their core shrouds. GL 94-03 requested BWR licensees to take the following actions with respect to their core shrouds:

- inspect their core shrouds no later than the next scheduled refueling outage;

- perform a safety analysis supporting continued operation of the facility until the inspections are conducted;
- develop an inspection plan which addresses inspections of all shroud welds, and which delineates the examination methods to be used for the inspections of the shroud, taking into consideration the best industry technology and inspection experience to date on the subject;
- develop plans for evaluation and/or repair of the core shroud; and
- work closely with the BWROG on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to intergranular stress corrosion cracking.

The PECO Energy Company (PECo), the licensee for the Peach Bottom Atomic Power Station Unit 2 (PBAPS 2), responded to GL 94-03 on August 24, 1994 (Reference 1). Part of the licensee's response included PECo's inspection scope for the planned inspection of the PBAPS 2 core shroud, scheduled for refueling outage (RFO) 2R10, which commenced on September 16, 1994. PECo also submitted an analysis of its proposed modification for the shroud circumferential welds. This modification was not implemented during the Unit 2 RFO 2R10. The staff's evaluation of PECo's proposed modification will be addressed in a separate Safety Evaluation Report (SER).

2.0 EVALUATION OF THE LICENSEE'S RESPONSE TO GL 94-03

PECo scheduled and performed comprehensive inspections of the PBAPS 2 core shroud during the unit's RFO 2R10, which commenced on September 16, 1994. The following gives the staff's assessment of the susceptibility of the PBAPS 2 core shroud, the scope of the inspection completed during RFO 2R10, and the licensee's assessment of identified cracking.

2.1 Susceptibility of the PBAPS 2 Core Shroud to IGSCC

The core shroud cracks which are the subject of GL 94-03, result from intergranular stress corrosion cracking (IGSCC) which is most often associated with sensitized material near the component welds. IGSCC is a time-dependent phenomena requiring a susceptible material, a corrosive environment, and a tensile stress within the material.

Industry experience has shown that austenitic stainless steels with low carbon content are less susceptible to IGSCC than stainless steels with higher carbon content. BWR core shrouds are constructed from either type 304 or 304L stainless steel. Type 304L stainless steel has a lower carbon content that type 304 stainless steel. During the shroud fabrication process when the sections of the core shroud are welded together, the heating of the material adjacent to the weld metal sensitizes the material. Sensitization involves carbon diffusion out of solution forming carbides at grain boundaries upon moderate heating. The formation of carbides at the grain boundaries depletes the chromium in the adjacent material. Since the corrosion resistance of stainless steel is provided by the presence of chromium in the material, the area adjacent to the grain boundary depleted of chromium is thereby susceptible to corrosion. Increased material resistance to IGSCC will result if the carbon content is kept below 0.035%, as specified for type 304L grade material.

Currently available inspection data indicate that shrouds fabricated with forged ring segments are more resistant to IGSCC than rings constructed from welded plate sections. The current understanding for this difference is related to the surface condition resulting from the two shroud fabrication processes. Welded shroud rings are constructed by welding together arcs machined from rolled plate. This process exposes the short transverse direction in the material to the reactor coolant. Elongated grains and stringers in the material exposed to the reactor coolant environment are believed to accelerate the initiation of IGSCC.

Water chemistry also plays an important role in regard to IGSCC susceptibility. Industry experience has shown that plants which have operated with a history of high reactor coolant conductivity have been more susceptible to IGSCC than plants which have operated with lower conductivities[1]. Furthermore, industry experience has shown that reactor coolant systems (RCSs) which have been operated at highly positive, electrochemical potentials (ECPs) have been more susceptible to IGSCC than RCSs that have been operated at more negative ECPs[2]. The industry has made a considerable effort to improve water chemistry at nuclear facilities over the past 10 years. Industry initiatives have included the introduction of hydrogen water chemistry as a means of lowering ECPs (i.e., making the ECPs more negative) in the RCS. The effectiveness of hydrogen water chemistry in reducing the susceptibility of core shrouds to IGSCC initiation has not been fully evaluated; however, its effectiveness in reducing IGSCC in recirculation system piping has been demonstrated.

Welding processes can introduce high residual stresses in the material at the

---

[1]Conductivity is a measure of the anionic and cationic content of liquids. As a reference, the conductivity of pure water is ~0.05 μs/cm. Reactor coolants with conductivities below 0.20 us/cm are considered to be relatively ion free; reactor coolants with conductivities above 0.30 μs/cm are considered to have a relatively high ion content.

[2]The electrochemical potential (ECP) is a measure of a material's susceptibility to corrosion. In the absence of an externally applied current, and therefore, for reactor internals in the RCS, the electrochemical potential is equal to the open circuit potential of the material. Industry experience has shown that crack growth rates in reactor internals are low when the ECP $\leq$ --0.230 volts.

weld joint. The high stresses result from thermal contraction of the weld metal during cooling. A higher residual tensile weld stress will increase the material's susceptibility to IGSCC. Although weld stresses are not easily quantified, previous investigation into weld stresses indicate that tensile stresses on the weld surface may be as high as the yield stress of the material. The stress decreases to compressive levels in the center of the welded section.

PECo has reviewed the materials, fabrication and operational histories of the PBAPS 2 core shroud and has submitted this information to the staff in their response to GL 94-03. The PBAPS 2 plant-specific susceptibility factors are summarized below:

- The shroud support, top guide support, and core support plate rings are fabricated from two welded 304 stainless steel, forged ring segments, with carbon contents of ~0.030%. The shroud shell region was fabricated by welding rolled 304 stainless steel plates together. The carbon content of the PBAPS 2 shroud plates are in the range of 0.050 - 0.065%.

- Welding of the shroud plates and rings for circumferential welds H1 - H6 was accomplished by submerged arc welding using ER308 filler metal. Welding of the bi-metallic weld, H7, was accomplished by gas metal arc welding using filler metal 82. Weld residual stress levels resulting from these fabrication processes are high.

- PBAPS 2 operated at high reactor coolant ionic content levels during the initial years of operation. The initial five year average coolant conductivity for PBAPS 2 was 0.593 μS/cm, which is considerably higher than the average for other U.S. BWRs (where the conductivities range from ~0.123 μS/cm to 0.717 μS/cm, and average ~ 0.340 μS/cm).

- PBAPS 2 has operated for 11.8 cumulative years at full power, which is slightly above the median for U.S. BWRs (range is 3.7 years - 17.8 years, with a median of 10.8 years).

The PBAPS 2 and Peach Bottom Unit 3 (PBAPS 3) reactors have operated for approximately the same amount of time at full power, and have in common a history of operation with high ionic content reactor coolants during the initial five years of power operation. As a basis for comparison, previous inspections of circumferential and vertical welds in the PBAPS 3 core shroud revealed the existence of a moderately sized crack (~105 inches in length) along the lower heat affected zone of the shroud's H3 circumferential weld, in addition to some less significant cracking at the H1 and H4 weld locations. From the perspective of materials and fabrication methods, the PBAPS 2 core shroud was fabricated in the same manner as was the PBAPS 3 core shroud. The Boiling Water Reactor Vessels & Internals Project (BWRVIP) has classified the PBAPS 2 core shroud as a susceptible Category "C" shroud. The staff finds that the BWRVIP's categorization of the PBAPS 2 core shroud is acceptable and considers the core shroud at PBAPS 2 to be as susceptible to IGSCC as the core shroud in the PBAPS 3 sister unit.

2.2 Inspection of the Peach Bottom Unit 2 Core Shroud

By letter dated November 7, 1994, PECo submitted the PBAPS 2 core shroud inspection scope, examination results and their flaw evaluation.

2.2.1 Scope of Core Shroud Inspection

The PBAPS 2 shroud examinations were performed using the ultrasonic testing (UT) methods developed by the General Electric Corporation (GE). The UT examinations utilized GE's Smart-2000 Data Acquisition System and the GE OD Tracker and suction cup scanners. The extent of the planned UT examinations included all accessible portions of circumferential shroud welds H1 - H7. The UT examinations were performed using three UT transducers, a 45° shear wave transducer, a 60° longitudinal wave transducer, and a creeping wave transducer which was used to pick up surface indications. The creeping wave transducer was not used on the H3 weld due to equipment failure. The licensee also performed some additional enhanced VT-1 examinations of shroud weld H6, which was highly obstructed by the proximity of the jet pumps and therefore highly inaccessible to the GE UT equipment. The licensee indicated that it had completed the following PBAPS 2 core shroud UT examinations:

- 33% of the length (230") of weld H1, distributed over 66% of the weld's circumference,
- 84% of the length (583") of weld H2,
- 88% of the length (574") of weld H3,
- 89% of the length (580") of weld H4,
- 83% of the length (540") of weld H5,
- 10% of the length (148") of weld H6, plus an additional 13% of weld H6 by enhanced VT-1 examination techniques, and
- 9% of the length (59") weld H7, in areas which were accessible by way of the access hole covers.

2.2.2 Core Shroud Examination Results

The following summarizes the cracking identified at each weld during the examination of the PBAPS 2 core shroud.

- H1 Weld - The examination detected 11 indications by UT using 45°S/60°RL transducers, totalling 33.93 inches, with a maximum length of 4.75 inches and a maximum depth of 0.74 inches at Indication #7;
- H2 Weld - Examinations were negative for indications;
- H3 Weld - 19 indications were detected by UT using 45°S/60°RL transducers, totalling 68.48 inches, with the maximum length being 8.75 inches at Indication #16 (indications were not depth sized).

- H4 Weld - 8 indications were identified, totalling 11.46 inches, with the maximum length of 5.76 inches at Indication #4 (indications were not depth sized) as detected by UT using 45°S/60°RL transducers, and remaining seven indications detected by UT creeping wave measurements;
- H5 Weld - 1 indication 2.28 inches in length was detected by UT creeping wave (indication was not depth sized);
- H6 Weld - 1 indication was detected by UT using 45°S/60°RL transducers, 4.73 inches in length and 0.45 inches in depth;
- H7 Weld - examinations were negative for indications.

The licensee's inspections of welds H6, the core support ring-to-lower shroud weld, H7, the lower shroud-to-shroud support cylinder weld, and H8, the shroud support cylinder-to-jet pump support ledge weld, were conducted through accessible areas of the access hole covers. Interference from jet pump assemblies, the reactor core, and other internals located at lower vessel elevations limited access to the lower shroud welds. The licensee's inspection plan is consistent with the staff's position recommending a 100% inspection of all accessible shroud weld areas.

2.2.3 Assessment of the PBAPS 2 Core Shroud Inspection Results

Flaws identified in welds receiving a comprehensive examination during the fall 1994 RFO were evaluated in accordance with the methodology outlined in the "BWR Core Shroud Inspection and Flaw Evaluation Guidelines" (Reference 2). These guidelines closely follow the flaw evaluation guidelines found in Section XI of the ASME Code. The staff has reviewed the BWRVIP evaluation guidelines and approves of the use of the quantitative assessment methods.

The licensee's evaluations were based on the following assumptions and conditions:

- For welds that were largely accessible to examinations and for which comprehensive examinations were performed, all as-found indications were assumed to be through-wall, which removed the necessity for depth characterization. Additionally, any inaccessible areas were assumed to contain through-wall indications over their entire inaccessible lengths.

- For welds that were predominantly inaccessible to examination, conditions found within the inspected regions were extrapolated over the entire weld areas that were inaccessible to examination equipment. The extrapolated conditions were then evaluated for structural integrity. Thus, evaluations of the H1 and H6 welds, in which indications were found and which were sized for depth, were based upon the assumption that the majority of the welds' circumferences contained indications.

- For the H7 weld, in which no indications were found, calculations were performed to calculate the depth which could be tolerated assuming a 360° crack existed in the weld.

- As-found crack lengths were adjusted for crack growth, non-destructive examination uncertainties, and crack proximity factors in accordance with the guidelines (Reference 2).

Inspection results for those welds receiving comprehensive inspections were compared to the initial screening criteria established in GENE 523-176-1293, "Evaluation and Screening Criteria for the Peach Bottom Unit 2 Shroud" (Reference 3), and if unacceptable, evaluated for safety margins using limit load methodology found in the "BWR Core Shroud Inspection and Flaw Evaluation Guidelines" (Reference 2). The inspection results of the H3 and H4 welds were also subject to evaluation using linear elastic fracture machanics methods to account for high neutron fluences which are common at these weld elevations.

Safety margins were calculated against the most limiting design basis loading conditions, derived in GENE 523-176-1293, "Evaluation and Screening Criteria for the Peach Bottom Unit 2 Shroud" (Reference 3). This equated to use of faulted condition loadings for evaluations of circumferential welds H1 - H5, and upset condition loadings for circumferential welds H6 and H7. For all postulated loadings the licensee showed that the loadings conditions for the as-found conditions in welds H1 - H7 were less than the ASME Code stress intensity allowables. The licensee's evaluations of the PBAPS 2 core shroud indicate that the shroud will maintain its structural integrity even under the most severe loading conditions for a given shroud weld location. The staff has reviewed the licensee's methodology, and has determined that the licensee's method of evaluating the PBAPS 2 core shroud is acceptable and that the licensee's evaluation results justify operation of the PBAPS 2 unit for the next operating cycle.

3.0 CONCLUSIONS

Based on a review of the PBAPS 2 core shroud materials, fabrication processes and operating history the staff concludes that the licensee's core shroud is susceptible to IGSCC. PECo completed an examination of the PBAPS 2 core shroud during RFO 2R10. The licensee's evaluation of the PBAPS 2 core shroud indicates that the PBAPS 2 core shroud will maintain sufficient structural margins to justify operation of the PBAPS 2 reactor for another operating cycle without necessitating a modification of the PBAPS 2 core shroud.

4.0 OUTSTANDING ISSUES/FUTURE ACTIONS

The licensee's difficulty inspecting the some of the circumferential core shroud welds is not unique to this plant. It should be noted that the industry is currently encountering difficulties performing comprehensive inspections of lower shroud welds due to NDE equipment accessibility problems. The staff urges licensees to work with the members of the EPRI NDE Center in order to develop improved tooling for inspections of lower shroud welds and/or lower vessel regions which are highly obstructed. Should improved inspections techniques become available, the staff recommendation is for licensee's to reinspect the lower shroud welds at the earliest opportunity.

5.0 REFERENCES

1. Letter from G. A. Hunger, Jr., PECO to the U.S. Nuclear Regulatory Commission forwarding the "Peach Bottom Atomic Power Station, Units 2 and 3, Limerick Generating Station Units 1 and 2 Response to Generic Letter 94-03, 'Intergranular Stress Corrosion Cracking of Core Shroud in Boiling Water Reactors," dated August 24, 1994.

2. Letter From C. D. Terry, Executive Chairman, Assessment Committee, BWR Vessel & Internals Project, to the U.S. Nuclear Regulatory Commission forwarding the "BWR Core Shroud Inspection and Evaluation Guidelines," dated September 2, 1994.

3. M. L. Herrera and S. Raganath, "Evaluation and Screening Criteria for the Peach Bottom Unit-2 Shroud," Rev. 0, (GENE-523-276-1093) dated December 13, 1993.

Principal Contributor: J. Medoff

Date: February 6, 1995

Dickinson M. Smith
Senior Vice President and
Chief Nuclear Officer

PECO ENERGY

PECO Energy Company
Nuclear Generation Group
965 Chesterbrook Blvd., 63C-3
Wayne, PA 19087-5691
610 640 6600
Fax 610 640 6611

10CFR 2.201
10CFR 2.205

December 21, 1994

Docket Nos. 50-277
50-278
License Nos. DPR-44
DPR-56

Director, Office of Enforcement
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Peach Bottom Atomic Power Station Units 2 & 3
Reply to Notice of Violation and Proposed Imposition of a Civil Penalty NRC Inspection Report Nos. 50-277/94-24; 50-278/94-24

Gentlemen:

In response to your letter dated November 21, 1994, which transmitted the Notice of Violation (NOV) and Proposed Civil Penalty, PECO Energy Company submits the attached reply. The NOV was identified in a special safety inspection (94-24/24) that evaluated activities performed August 3, 1994, that placed the Emergency Service Water (ESW) system in an unanalyzed configuration for approximately 50 minutes.

A check in payment of the civil penalty made payable to the Treasurer of the United States was transmitted separately by PECO Energy letter to the Director, Office of Enforcement dated December 21, 1994.

If you have any questions or desire further information, please do not hesitate to contact us.

Dm Smith

270077

Enclosure 3

JE14 /1

Attachment and Affidavit

cc: R. A. Burricelli, Public Service Electric & Gas
R. R. Janati, Commonwealth of Pennsylvania
T. T. Martin, USNRC, Administrator, Region I
W. L. Schmidt, USNRC, Senior Resident Inspector
H. C. Schwemm, VP - Atlantic Electric
R. I. McLean, State of Maryland
A. F. Kirby III, DelMarVa Power

COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF CHESTER :

D. M. Smith, being first duly sworn, deposes and says:

That he is Senior Vice President and Chief Nuclear Officer of PECO Energy Company; that he has read the attached reply to Notice of Violation and Proposed Imposition of a Civil Penalty NRC Inspection Report No. 94-24, for Peach Bottom Atomic Power Station Facility Operating Licenses DPR-44 and DPR-56 and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

Senior Vice President and
Chief Nuclear Officer

Subscribed and sworn to
before me this 21st day
of December 1994.

Notary Public

Notarial Seal
Erica A. Santori, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires July 10, 1995

RESPONSE TO NOTICE OF VIOLATION 94-24-01

Restatement of the Violation

A. 10 CFR 50, Appendix B, Criterion III, requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into procedures and instructions.

Contrary to the above, on August 3, 1994, the licensee conducted a testing activity on the emergency service water (ESW) system that placed the system in a configuration that was not within the design basis described in the Updated Safety Analysis Report. Specifically, ESW system valve MO-498, the system's normal return to the ultimate heat sink (UHS), was shut and left unattended. As a result, the ESW system flow to safety-related components was reduced to the extent that adequate cooling was not available in the event that the design basis accident occurred at the design basis UHS maximum temperature. (01013)

B. 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions and procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures and instructions.

Contrary to the above, on August 3, 1994, the licensee tested ESW System Valve MO-498, an activity affecting quality, in a manner that was not prescribed by documented instructions and procedures of a type appropriate to the circumstances. Valve MO-498, the ESW system normal return to the ultimate heat sink and important to maintaining adequate cooling water flow to safety-related components, was shut and procedures were not in place to require personnel to remain at the valve and immediately open the valve if needed in the event of an accident. As a result of the inadequate procedure, after shutting the valve, maintenance personnel left the valve unattended and in the shut position for approximately 50 minutes. (01023)

This is a Severity Level III problem (Supplement I).

Admission or Denial of Alleged Violation

The PECO Energy Company acknowledges the violation.

Background

On August 3, 1994, at approximately 12:21 PM a clearance was applied to Motor Operated Valve MO-0-33-498 to allow diagnostic testing of the valve. This valve controls ESW discharge flow to the Susquehanna River. Testing was performed in accordance with Maintenance Procedure M-511-130, "Procedure for Diagnostic Testing of Limitorque Motor Operated Valves using Liberty Technologies 'Votes' Method." This procedure dealt with the mechanics of performing the test and did not address system operability issues that could arise.

The MO-498 breaker was blocked and locked in the open condition. A Special Condition Tag was hung on the breaker to allow Maintenance technicians to operate the breaker and the valve during the VOTES test. Maintenance technicians received the key to unlock the breaker as part of the clearance. With the valve breaker in the open position, control room indication of valve position became unavailable.

At 6:27 PM two Maintenance technicians entered the Control Room to obtain permission to begin VOTES testing of another ESW valve, MO-0-33-841, the Emergency Cooling Water Pump Discharge Valve. Approximately 10 minutes later, two other Maintenance technicians entered the Control Room to obtain permission to VOTES test MO-498. While both groups were in the Control Room they each received permission to begin testing from the Work Control Supervisor. In addition, the MO-498 work crew received permission to begin work from the Unit 2 Reactor Operator.

The Unit 2 Reactor Operator had reservations about allowing work to be done on MO-498 and expressed his concerns to the Control Room Shift Supervisor. The Control Room Shift Supervisor addressed these concerns by questioning one of the Maintenance technicians who he thought was working on MO-498. Through this questioning he confirmed that the testing would not mechanically disable the valve, that the valve would be immediately available to the operator if needed, and that the technicians had a radio so that they could be immediately contacted by the Control Room. Satisfied that operators would be able to take control of the valve immediately if necessary, the Control Room Shift Supervisor informed the Unit 2 Reactor operator that valve testing was permissible. In reality, however, the Control Room Shift Supervisor had questioned the lead technician working on MO-841.

At approximately 7:07 PM testing began on MO-498. The testing required the Maintenance technicians to close the valve breaker and operate the valve locally from its breaker in the E-4 diesel bay. During this testing the Maintenance technicians did not notify the Control Room when the valve's position was changed. They believed that the operator signoff in their test procedure which granted permission to perform VOTES testing also constituted the operator's permission to change valve position as needed without prior control room notification.

At 10:22 PM the Maintenance technicians temporarily stopped work and left the work area. At that time, they left MO-498 in the closed position, reopened the valve breaker and locked it. The key for the valve breaker lock remained with the Maintenance technicians who did not notify the Control Room operators that they had left the valve area or that the valve was in the closed position. With MO-498 closed, service water which is normally supplied to ECCS cooling loads was discharged to the Emergency Cooling Tower instead of the river. The technicians believed that they were leaving the valve in a safe condition. The work package did not provide any information on a preferred valve position nor did it prohibit the valve from being left unattended.

Sometime after MO-498 was closed the emergency cooling tower high/low level alarm was received in the Control Room. Operators confirmed that tower level was high using a control room level indicator. They attributed the level increase to rain. Per the alarm response card, the appropriate action was to reduce tower level using the Emergency Cooling Water pump and MO-841. Typically this condition does not require an immediate response and with MO-841 under test, an immediate pump down of the tower was not undertaken.

At 11:09 PM the Maintenance technicians returned to MO-498. At about the same time, the afternoon and night shift Unit 2 Reactor Operators had completed their turnover and the oncoming Reactor Operator began to think about possible reasons for the emergency cooling tower high level alarm. He was skeptical that the alarm was caused by rain. At 11:15 PM just before the Reactor Operator recognized the connection between the emergency cooling tower high level alarm and the work on MO-498, a security guard notified the Control Room that water was overflowing the Emergency Cooling Tower basin. The Reactor Operator immediately informed the Control Room Shift Supervisor that the overflow was probably caused by the work on MO-498.

The Control Room Shift Supervisor contacted the Maintenance technicians informing them of the Emergency Cooling Tower overflow and the need to open MO-498 and MO-841 to allow the cooling tower to drain down. The two valves were opened and the restoration of the cooling tower level to normal was completed. Once the MO-498 was stroked to the open position, the ESW system was returned to an analyzed condition.

Following identification of this problem by the NRC, calculations by PECO Engineering determined that ESW flow would have been reduced by approximately 40%. Additional calculations were performed using this reduced flow rate to determine the operability of emergency diesel generators and ECCS equipment assuming the worst case plant licensing event, a loss of coolant accident with a loss of offsite power. These calculations showed that with the river and air temperatures that existed on the day of the event, all ECCS room coolers and equipment coolers would have performed their design function throughout the event. In addition, the required number of emergency diesel generators would have remained operable during the first ten minutes without operator action. The diesels would have remained operable following the first ten minutes if diesel loads were balanced to below their continuous rating of 2600 kw. Analysis also showed, however, that the reduced ESW flow would have prevented the diesels from performing their safety function had the design basis accident occurred with river water at its design maximum temperature of 90 degrees F. Actual river temperature on the day of the event was 81 degrees F.

Reason for the Violation

Administrative controls to ensure that MO-498 would remain operable during VOTES testing were not clearly established as part of the planning for this activity. Likewise the impact of closing MO-498 on emergency cooling tower level and the ESW system were not addressed in the work package. Continued operability of MO-498 during VOTES testing came to depend solely on the controls the Operators put in place at the start of the job. The challenges encountered during this event could have been avoided had adequate planning taken place before the work request reached the Control Room.

During the planning process it was decided that MO-498 could remain operable during VOTES testing, however, the operability impact associated with this decision was not carefully evaluated or managed. Enhanced work controls to limit the chance of an undetected inoperable condition should have been written into the work package to supplement any verbal controls imposed by Operations. Although written instructions had been successfully used in the past to control work activities, an expectation that such instructions be consistently included in work packages involving operable safety related equipment had not been established. As a result, no one was responsible to verify that it was included in the work package and the absence of enhanced guidance and control was not questioned.

Diagnostic MOV testing had been conducted for several years with no adverse consequences. As a result, VOTES testing was perceived to be a low risk operation with little cause for concern. This perception caused some personnel to be less sensitive to the potential for a problem during the testing of MO-498. Personnel interviewed had a very general understanding of the VOTES testing process and thought the process simply involved the momentary stroking of a valve to obtain test data from installed sensors. There were no previous problems that would have caused this concept of VOTES testing to be questioned or compelling reasons to research the actual details of the testing procedure. This lack of knowledge about the details of the VOTES test reduced the likelihood that personnel who understood the design and operation of the ESW system would foresee the impact on Emergency Cooling Tower level and restrict the time that the valve could be left closed. Such a restriction may have prevented the maintenance technicians from leaving the valve in the closed position.

Extensive reviews had been previously conducted to determine if equipment operability could be maintained during testing. Tests where equipment could remain operable were reviewed to ensure that appropriate controls were established and written into procedures. This review was restricted to surveillance and routine testing. VOTES testing is a preventative maintenance task which does not fall into either category, therefore it was never thoroughly evaluated.

The request to conduct VOTES testing on MO-498 should have initiated the imposition of enhanced test controls and increased monitoring of the condition of the valve by Operations. Several opportunities to establish these controls existed, but were not effectively achieved. The first opportunity came when Maintenance requested permission from the work control supervisor to initiate work. The work control supervisor recalled being concerned about simultaneous work on MO-498 and MO-841, but did not establish any special controls. Secondly, concern was expressed by the Unit 2 Reactor Operator when he was asked to grant permission to allow testing on MO-498. His concerns were directed to the Control Room Shift Supervisor who resolved the concerns to their mutual satisfaction. Although these individuals recognized that MO-498 was a safety significant valve, the degree of monitoring and control established over the testing of MO-498 was inadequate in view of its safety significance.

The Control Room Shift Supervisor tried to affirm the acceptability of working on the MO-498 valve by questioning one of the maintenance technicians who was in the Control Room seeking permission to perform VOTES testing. The technician questioned, however, was actually working on MO-841. The questions asked by the Shift Supervisor were appropriate, but were general in nature so that neither party realized that they were talking about different valves. As a result, the Maintenance Technicians working on MO-498 never heard the Shift Supervisor's questions and had no awareness of the RO's concerns.

interaction between the Maintenance Crew and Control Room Operators during MO-498 testing was less than adequate. The Clearance and Tagging Manual requires that Shift Management permission be obtained immediately prior to each Special Condition Tag (SCT) component manipulation. However, the manual also provides an exception to this requirement stating that at the discretion of Shift Management, permission may extend through a series of manipulations not to exceed the shift of the individual granting the permission. During the event the Maintenance technicians did not notify Shift Management immediately prior to each manipulation of the valve. The technicians interpreted the Work Control Supervisor signoff in their test procedure granting permission to perform the test as also granting the exemption from making the notifications. In the mind of the Maintenance Technician, the permission to conduct VOTES testing automatically included permission to stroke the valve and apply the exception for SCT component manipulation notification. Previous experience and the absence of any contrary direction from Operations validated these assumptions.

When the Maintenance technicians left the work area, they left MO-498 in the closed, deenergized position thinking that this was a safe configuration that did not adversely impact plant safety. They did not understand the function of the valve in relation to the ESW system and therefore made an incorrect decision. Had the technicians been clearly informed of the function of the valve and its safety significance by a pre-job briefing, this event may have been averted. This information, however, was not provided to the technicians before they went to the Control Room to get permission to initiate testing. It also was not provided by any of the Operations personnel who had contact with the technicians.

<u>Corrective Steps That Have Been Taken and The Results Achieved</u>

A Performance Enhancement Program (PEP) investigation (PEP-I0002628) was initiated September 7, 1994, to determine the causal factors of placing the ESW system in an unanalyzed configuration and to develop appropriate corrective actions to prevent recurrence.

Appropriate counselling and disciplinary actions were administered commensurate with individual's level of responsibility.

This event was reviewed with Maintenance and Operations and Planning personnel.

Required reading packages were developed and communicated to Operations personnel on September 12 & 13, 1994. Operations personnel were instructed to consider Motor Operated Valves inoperable during VOTES testing and were given specific instruction to consider systems inoperable with components being worked under action requests, minor maintenance, SCT, or "Fix it Now" (FIN) team work unless otherwise determined by a licensed operator.

MO-498 was information tagged indicating that it shall only be operated using PORC approved procedures that specifically address MO-498. The VOTES test procedure is an example of a procedure that does not meet this criteria.

Expectations for manipulation of components covered under an SCT were issued to Maintenance and Operations personnel stating that effective communication must occur between Shift Management and Maintenance prior to component manipulation. Additionally, the terms "Shift Management" and "immediately prior to" were clearly defined.

The work planning and Operations Service Group have been reorganized to facilitate improved planning and work coordination. Expectations for improved planning and coordination of work activities, especially those performed on operable equipment, have been communicated to personnel in the planning organization. This includes the expectation that appropriate information and controls related to equipment operability be documented in the work packages.

Corrective Steps that Will be Taken To Avoid Further Violations

An Operations Improvement Plant was developed December 13, 1994, to reinforce proper standards and expectations to improve overall performance. This plan will be implemented through 1995 to ensure clear understanding of roles and responsibilities of Operations personnel, involvement of upper level management when operating limits could be unnecessarily challenged, and the need to continually maintain a healthy skepticism and questioning attitude during work evolutions. This plan also includes reinforcement of management expectations regarding the conduct of pre-job briefings, verbal communication standards, and the need for heightened operator awareness and control during the conduct of work activities involving operable equipment.

Enhanced controls are being added to the VOTES test procedure. Maintenance Procedure M-511-130, "Procedure for Diagnostic Testing of Limitorque Motor Operated Valves using Liberty Technologies 'VOTES' Method" will be revised to clearly delineate a section where Operations can document restrictions or controls on the performance of VOTES testing. This revision will be completed by January 31, 1995.

Procedures governing releases of equipment for maintenance are being enhanced to provide clear guidance regarding equipment operability and control requirements. This item will be completed by March 31, 1995.

The Date When Full Compliance Was Achieved

Full compliance was achieved August 3, 1994, when MO-498 was re-opened and the ESW system was returned to an analyzed condition.

Stati n Support Department

DCS

PECO ENERGY

PECO Energy Company
Nuclear Group Headquarters
965 Chesterbrook Boulevard
Wayne, PA 19087-5691

December 21, 1994

Docket Nos. 50-277
50-278
License Nos. DPR-44
DPR-56

Mr. James Lieberman
Director, Office of Enforcement
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Peach Bottom Atomic Power Station, Units 2 and 3
Remittance of Civil Monetary Penalty

Dear Mr. Lieberman:

This letter is being submitted in response to an NRC letter dated November 21, 1994, issuing a Notice of Violation and proposed imposition of civil penalty in the amount of $87,500 for violations of NRC regulations as set forth in 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." We are remitting the enclosed check in the amount of $87,500 for payment of the cited civil penalty.

If you have any questions or require additional information, please do not hesitate to contact us.

Very truly yours,

G. A. Hunger, Jr.

G. A. Hunger, Jr.
Director - Licensing

Enclosure

cc: T. T. Martin, Administrator, Region I, USNRC (w/o enclosure)
W. L. Schmidt, USNRC Senior Resident Inspector, PBAPS (w/o enclosure)
R. R. Janati, Commonwealth of Pennsylvania (w/o enclosure)

Rec'd w/check
12/23/94

280015

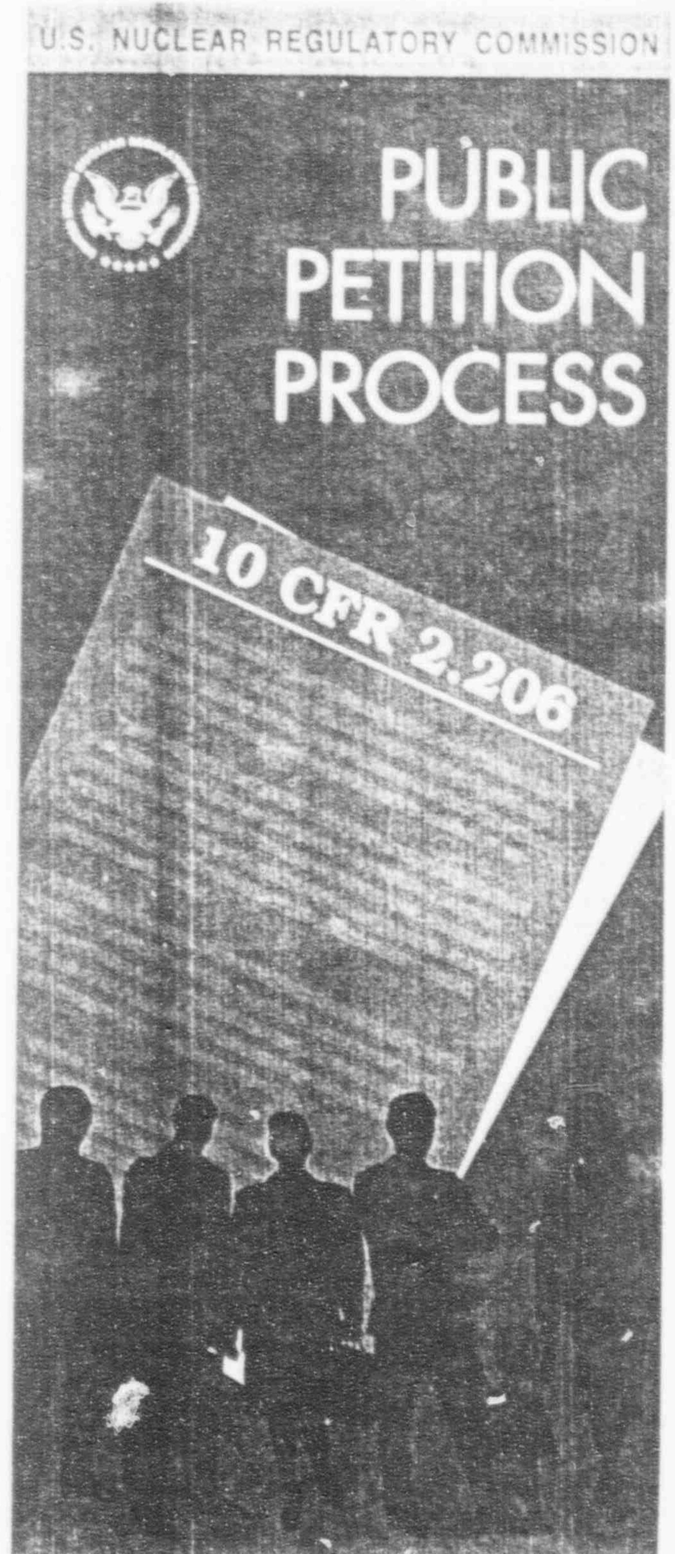
Enclosure 4

IE14
10

Director's Decision

Office of Public Affairs
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Telephone 301-415-8200

NUREG/BR-0200



not offer any preliminary decisions during the informal public hearing. Although not adjudicatory in nature, the informal public hearing is transcribed, and the text is made public shortly afterwards.

Director's Decision

The official NRC response to a 2.206 petition is a written Director's Decision that addresses the concerns raised in the petition. The agency's goal is to issue a decision, by the appropriate office director, within 120 days from the date of the acknowledgment letter unless an investigation is involved or additional time is needed for a hearing. The Director's Decision includes the professional staff's evaluation of all pertinent information from the petition, correspondence with the petitioner and the licensee, information from any informal public hearing, results of any investigation, and any other documents related to petition issues. The Director's Decision is provided to the petitioner and the licensee and is published in the *Federal Register*.

Director's Decisions may be issued as follows:

- A decision granting a petition, in full, explains the basis for the decision and grants the action requested in the petition (e.g., NRC issuing an order to modify, suspend, or revoke a license).
- A decision denying a petition, in full, provides the reason for the denial and discusses all matters raised in the petition.
- A partial Director's Decision may be issued when:
  - NRC decides not to grant the action requested in the petition but takes other appropriate action (e.g., issuing a Notice of Violation or a civil penalty) to resolve the identified safety concerns, thus partially denying the petition; or
  - Some of the issues associated with the petition can be completed and significant schedule delays are anticipated before resolution of the entire petition.

The Commission will not entertain requests for review of a Director's Decision. However, on its own, it may review a decision within 25 calendar days. Afterwards, NRC issues a notice indicating whether the Commission has reviewed the decision and sends copies to the petitioner, the licensee, and the *Federal Register* for publication.

NRC Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions," contains more detailed information on citizen petitions. For a copy of the directive, write to the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082, or call 202-512-2249.

The electronic bulletin board on 2.206 petitions may be accessed, using a personal computer (PC) and a modem, by calling 1-800-303-9672 (communication parameters 8-N-1-F). There are PCs located at the main PDR and several LPDRs which are available to the public. Call 202-634-3273 for information about PC access at the PDR and call 1-800-638-8081 for the LPDRs.

Office of Public Affairs
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Telephone 301-415-8200

NUREG/BR-0200

U.S. NUCLEAR REGULATORY COMMISSION

PUBLIC PETITION PROCESS

10 CFR 2.206

Introduction

The U.S. Nuclear Regulatory Commission (NRC) was established in 1975 to protect public health and safety in the civilian use of nuclear power and materials in the United States. As part of its responsibilities, the NRC is interested in assessing all potential health and safety issues related to licensed activities and encourages members of the public to bring serious issues to its attention.

Section 2.206 of Title 10 of the *Code of Federal Regulations* (10 CFR 2.206) describes the petition process—the primary mechanism for the public to raise potential health and safety issues in a public process.* This process permits anyone to petition the NRC to take action against one or more licensees. Depending on the results of its evaluation, NRC could modify, suspend, or revoke an NRC-issued license or take any other appropriate action to resolve a problem.

In 1993, the NRC reassessed the 2.206 petition process to determine whether it was effective, understandable, and credible. As part of its reassessment, the agency held a public workshop and obtained extensive comments from citizens' groups, the nuclear industry, former petitioners, and State and local governments. As a result, NRC made improvements to the 2.206 process to increase opportunities for meaningful public participation and to improve communications between the petitioner and NRC.

These improvements include—

- Offering, under certain circumstances, an informal public hearing to a petitioner.
- Providing copies of all pertinent correspondence to all participants involved in a petition issue.
- Identifying a single agency contact for each petition.
- Establishing an electronic bulletin board to provide the status of all pending petitions to the public.

*The NRC also has an allegation process in which individuals who raise potential safety concerns for NRC review are afforded a degree of protection of their identity. Specific guidance on the allegation process is contained in NRC Management Directive 8.8, "Management of Allegations," and described in a separate pamphlet available from the Office of Public Affairs.

The Petition Process

The 2.206 process provides a simple, effective mechanism to identify potential health and safety issues for prompt, thorough, and objective evaluation by NRC. It is separate and distinct from the processes for rulemaking and licensing, although they too allow the public to raise safety concerns to the NRC.

Under the 2.206 process, the petitioner submits a request in writing to NRC's Executive Director for Operations, identifying the affected licensee, the requested action to be taken, and the facts the petitioner believes provide sufficient grounds for NRC to take action. Unsupported assertions of "safety problems" or general opposition to nuclear power are not considered sufficient grounds.

After receiving a request, NRC determines whether (1) the request qualifies as a 2.206 petition, (2) an investigation of potential wrongdoing is appropriate, and (3) an informal public hearing is warranted. Note that the informal public hearing can be offered at any time during NRC's review of a petition. The NRC sends an acknowledgment letter to the petitioner, with a copy to the licensee. If the request is accepted for review as a 2.206 petition, NRC publishes a notice in the *Federal Register*.

Based on evaluation of the petition, the appropriate office director issues a decision and, if warranted, NRC takes appropriate action. Throughout the evaluation process, NRC sends copies of all pertinent correspondence to the petitioner and the affected licensee. In most cases, NRC places correspondence in the Public Document Room (PDR) in Washington, D.C., as well as the appropriate Local Public Document Room (LPDR) near the affected facility. However, the agency withholds information that would compromise an investigation or ongoing enforcement action relating to issues in the petition. The NRC also sends the petitioner other information such as pertinent generic letters and bulletins.

The NRC notifies the petitioner of the status of the petition every 60 days, or more frequently if a significant action occurs.

Monthly updates on all pending 2.206 petitions are available on the electronic bulletin board, which is available to the public, and a status report is available in the PDR.

Informal Public Hearing

An informal public hearing serves not only as a source of potentially valuable information for NRC to evaluate a 2.206 petition, but also affords the petitioner substantive involvement in the review and decisionmaking process through direct discussions with NRC and the licensee. An informal public hearing is offered only if the petition meets certain requirements; it is not offered automatically or solely at the petitioner's request. To qualify for an informal public hearing, the petition must present new information that raises a significant safety issue or alleges violation of NRC requirements involving a significant safety issue for which new information or a new approach has been provided. Information is considered "new" if it presents a significant safety issue not previously evaluated or provides a new approach or new information on a significant safety issue previously evaluated by NRC. No informal public hearing is offered if the petition involves sensitive information such as safeguards, facility security, proprietary, or confidential commercial information. The NRC publishes a notice in the *Federal Register* 30 days in advance of each hearing.

The informal public hearing is usually held near the affected facility or, if the petition raises generic issues covering facilities nationwide, in the Washington, D.C., area. To the extent practicable, the informal public hearing is scheduled during the evening hours and should last no longer than three hours. The NRC does