JUL 2 1974

Docket No. 50-278

Philadelphia Electric Company
ATTN: J. L. Hankins
Vice President
2301 Market Street
Philadelphia, Pennsylvania 19101

Gentlemen:

The Atomic Energy Commission has issued Facility Operating License No. DPR-56. The licensees of DPR-56 are the Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company. DPR-56 authorizes operation of the Peach Bottom Atomic Power Station, Unit 3 in accordance with the Technical Specifications, Appendices A & B currently in effect for Unit 2 as modified by the page changes attached hereto. The steady state reactor core power levels as authorized by DPR-56 shall not exceed 3293 megawatts thermal. A copy of the license and the page changes for the Technical Specifications are provided as Enclosure 1.

Note that the Technical Specification, Appendix A, Section 6.10 provides that the licensees shall not undertake initial operation of Unit 3 with a primary coolant temperature in excess of 212°F until specifically approved in writing by the Commission. Representatives of the Directorate of Regulatory Operations will be at the site during fuel loading and subsequent operations and will verify the completion and assessment of the items listed in Enclosure 2. We will inform you promptly of the results of their review. On satisfactory verification of the items identified in Enclosure 2, we will prepare the necessary approval in accordance with Section 6.10 of the Technical Specifications.

A related notice, which is being forwarded to the Office of Federal Register for filing and publication, is enclosed for your information (Enclosure 3).

Five signed originals of Amendment 16 to the Indemnity Agreement No. B-28, which covers the activities authorized under License No. DPR-56, are provided as Enclosure 4 for review and acceptance by the licensees. One copy signed by all the licensees should be returned to this office.

RETYPED PER J.SCINTO, OGC. SEE PREVIOUS YELLOW

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Docket No. 50-278

Philadelphia Electric Company
ATTN: J. L. Hankins
Vice President
2301 Market Street
Philadelphia, Pennsylvania 19101

Gentlemen:

The Atomic Energy Commission has issued Facility Operating License No. DPR-56. The licensees of DPR-56 are the Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company. DPR-56 authorizes operation of the Peach Bottom Atomic Power Station, Unit 3 in accordance with the Technical Specifications, Appendices A & B currently in effect for Unit 2 as modified by the page changes attached hereto. The steady state reactor core power levels as authorized by DPR-56 shall not exceed 3293 megawatts thermal. A copy of the license and the page changes for the Technical Specifications are provided as Enclosure 1.

Add Sentence per my note discussion with Butler JB

Note that the Technical Specification, Appendix A, Section 6.10 provide that the licensees shall not undertake initial operation of Unit 3 with a primary coolant temperature in excess of 212°F until specifically approved in writing by the Commission. Representatives of the Directorate of Regulatory Operations will be at the site during fuel loading and subsequent operations and will verify the completion and assessment of the items listed in Enclosure 2. We will inform you promptly of the results of their review. Insert

A related notice, which is being forwarded to the Office of Federal Register for filing and publication, is enclosed for your information (Enclosure 3).

Five signed originals of Amendment 16 to the Indemnity Agreement No. B-28, which covers the activities authorized under License No. DPR-56, are provided as Enclosure 4 for review and acceptance by the licensees. One copy signed by all the licensees should be returned to this office.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Also enclosed (Enclosure 5) for your information is a copy of the Atomic Safety and Licensing Board's Initial Decision dated June 14, 1974.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Directorate of Licensing

Enclosures:
1. Facility Operating License, DPR-56
2. List of Outstanding Items
3. Federal Register Notice
4. Indemnity Agreement - B-28, Amend. 16
5. Initial Decision, dated June 14, 1974

cc w/o encl: Mr. R. A. Heiss, Coordinator
Pennsylvania State Clearinghouse
Governor's Office of State Planning
and Development
P. O. Box 1323
Harrisburg, Pennsylvania 17120

cc w enclosures: See attached list
bcc: A. Rosenthal, ASLAB
N. Goodrich, ASLBP

DISTRIBUTION:
Docket No. 50-278
AEC/PDR
L/PDR
LWR 1-2 File
JScinto, OGC
RO (3)
NDube (w/o tech. specs)
MJinks (w/2 enc)
RCDeYoung
RVollmer (w/tech. specs)
CHebron, F&M
DFoster, F&M
EBrown, F&M
ABraitman, OAI (w/o tech specs)
MJMaigret
RPowell
SKari (w/o tech. specs)
WMiller, DR:OA (w/o tech specs)
GDicker, EP-2
DMuller, A/D, EP
LWR 1-2 Brahch Chiefs (w/o tech specs)
ACRS (16)
BSchraf

OFFICE	MMaigret/ RPowell:ldm	L:LWR 1-2	OGC	Telecopied OAI	A/D/EP GKD	AD/LWR-1
SURNAME	L:LWR 1-2	NButler		A. BRAITMAN		
DATE	6/27/74	6/29/74	6/ /74	6/28/74	7/1/74	6/ /74

Troy B. Conner, Jr., Esq.
Conner, Hadlock & Knotts
1747 Pennsylvania Ave., N.W.
Washington, D.C. 20006

Eugene J. Bradley, Esq.
2301 Market Street
Philadelphia, Pa. 19101

Raymond L. Hovis, Esq.
35 South Duke Street
York, Pa. 17401

W. W. Anderson, Esq.
Deputy Attorney General
Department of Justice
Second Floor - Capitol Annex
Harrisburg, Pa. 17120

John B. Griffith, Esq.
Special Assistant Attorney
General, Maryland
Annapolis, Md. 21401

Warren Rich, Esq.
Special Assistant Attorney
General, Maryland
Annapolis, Md. 21401

Frank R. Clokey, Esq.
Special Assistant Attorney General
Room 219, Towne House Apartments
Harrisburg, Pa. 17105

Mr. Thomas M. Gerusky, Director (2)
Office of Radiological Health
Dept. of Environmental Resources
P. O. Box 2063
Harrisburg, Pa. 17105

Mr. Albert R. Steele, Chairman
Board of Supervisors
Peach Bottom Township
R. D. No. 1
Delta, Pa. 17314

Mr. Robert Blanco
Environmental Protection Agency
Curtis Building
6th and Walnut Street
Philadelphia, Pa. 19106

Mr. Wilmer P. Bolton
Chairman, Board of Supervisors
Drumore Township
R. D. No. 1
Holtwood, Pa. 17532

Mr. Warren D. Hodges, Director
Dept. of State Planning
301 West Preston Street
Baltimore, Md. 21201

Mr. N. C. Vasuki, Director
Div. of Environmental Control
Dept. of Natural Resources and
Environmental Control
Tatnall Building
Dover, Delaware 19901

Mr. Bruce Blanchard
Environmental Projects Review
Dept. of the Interior, Room 5321
18th and C Streets, N.W.
Washington, D.C. 20240

Mr. J. R. Buchanan, Assistant Director
NSIC
Oak Ridge National Lab.
P. O. Box Y
Oak Ridge, Tenn. 37830

Mr. T. B. Abernathy
U. S. AEC-DTIE
Document Management Branch
P. O. Box 62
Oak Ridge, Tenn. 37830

Mr. J. Cusak
Brookhaven National Laboratory
Building 197C
Associated Universities, Inc.
Upton, New York 11973

Chief, TIRB
Technology Assessment Division
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 647A East Tower, Waterside Mall
401 M Street, S.W.
Washington, D.C. 20460

OFFICE ➤					
SURNAME ➤					
DATE ➤					

CHECKLIST FOR ISSUANCE OF FACILITY LICENSE

APPLICANT Philadelphia Electric Company, et al DOCKET NO. 50-278

FACILITY Peach Bottom Unit 3

PROJECT MANAGER Raymond Powell

LICENSING ASSISTANT Madelyn J. Maigret

	DATE
Notice of Consideration of Issuance of License:	
Published in FEDERAL REGISTER	
Action Date	
OR	
Initial Decision or Order	June 14, 1974
Safety Review:	
L Safety Evaluation	August 11, 1972
ACRS Letter	September 21, 1972
Environmental Review:	
Final Environmental Statement	April 1973
Published in FEDERAL REGISTER	April 10, 1973
Antitrust Review:	
OAI Concurrences	June 19, 1974
Notifications Required by Act & Commission Rules*:	
State Official	October 9, 1970
Local Official	August 31, 1970
Water Quality Certification:	
Submittal by Applicant	May 8, 1974
Transmitted to EPA	June 26, 1974
License Fee:	
Amount: $545,160 Paid	June 27, 1974
Indemnity Agreement:	
OAI Concurrence	June 19, 1974
Status of Outstanding Construction Items Checked w/RO	June 18, 1974
Regulatory Operations Final Report: (If Available)	July 2, 1974
Technical Specifications: Issued with P. B. Unit 2's license	
RP Concurrence	October 25, 1973
EP Concurrence Revised Pages - July 1, 1974	October 25, 1973
Public Announcement (to be released): ATTACHED	July 3, 1974
Issuance Package: OGC Concurrence	
1. License	July 2, 1974
2. FEDERAL REGISTER Notice	July 2, 1974
3. Letter to Applicant	July 2, 1974
4. Information Report	July 2, 1974

* Date Initial Application Forwarded

Revised: MAY 7 1973

AEC ISSUES FULL POWER LICENSE FOR OPERATION OF UNIT 3
OF PEACH BOTTOM ATOMIC POWER STATION IN PENNSYLVANIA

A license authorizing full power operation of Unit 3 of the Peach Bottom Atomic Power Station in York County, Pennsylvania, was issued on _____ to Philadelphia Electric Company and three other utilities by the Atomic Energy Commission's Directorate of Licensing. At full power, Unit 3 will have an electrical output of about 1,065 megawatts.

Full power operating licenses for Units 2 and 3 were the subject of a public hearing conducted by an Atomic Safety and Licensing Board. On September 14, 1973, the Board authorized issuance of a full power license for Unit 2, but deferred decision on Unit 3 pending further consideration of water quality matters related to the combined discharge from both units.

Further evidentiary hearing on Unit 3 was held in January 1974, and on June 14, 1974 the Licensing Board issued its decision authorizing issuance of a license for Unit 3. As a condition of licensing, the Board is requiring that a closed cycle cooling system be installed and put into operation by January 1, 1977. Operation of the facility with the once-through cooling system, using forced draft (open cycle) helper cooling towers to partially cool the water before discharge will be permitted until then.

From among several alternative closed cycle cooling systems, the licensee will select a preferred ~~one~~ system and submit to the AEC Regulatory Staff ~~a detailed~~ evaluation of the economic and environmental impact of the preferred system. After approval by the

Staff, the system must then be designed, built and placed in operation by the 1977 deadline. ~~If it cannot be accomplished by that time, the licensee may seek approval of the Staff for operation at such reduced power levels as will insure that there will be no violations of Federally-approved water quality standards of both Maryland and Pennsylvania.~~

Before the license for Unit 3 was issued, it was inspected by AEC representatives to assure that it has been satisfactorily constructed and was ready for fuel loading. The full term of the license is forty years from January 31, 1968, the date the AEC construction permit was issued.

Units 2 and 3 of the Peach Bottom Station each use a boiling water reactor and each has an electrical generating capacity of about 1,065 megawatts. The Station is located in Peach Bottom Township on the Susquehanna River. Sharing ownership in Units 2 and 3 with Philadelphia Electric are Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company.

#

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY
DOCKET NO. 50-278
PEACH BOTTOM ATOMIC POWER STATION, UNIT 3
FACILITY OPERATING LICENSE

License No. DPR-56

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for license filed by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company (the licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Peach Bottom Power Station, Unit 3 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-38 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (1) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (2) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. Philadelphia Electric Company is technically qualified and the licensees are financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

G. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Facility Operating License No. DPR-56 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied; and

I. The receipt, possession, and use of by-product and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including 10 CFR Section 30.33, 40.32, and 70.23 and 70.31.

2. Facility Operating License No. DPR-56 is hereby issued to the Philadelphia Electric Company (PECO), Public Service Electric and Gas Company (PSE&G), Delmarva Power and Light Company (DP&LC), and Atlantic City Electric Company (ACEC), to read as follows:

A. This license applies to the Peach Bottom Atomic Power Station, Unit 3, a direct cycle, forced circulation, boiling water nuclear reactor and associated equipment (the facility), owned by the licensees and operated by PECO. The facility is located in Peach Bottom, York County, Pennsylvania and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 1 through 31) and the Environmental Report as supplemented and amended (Supplements 1 through 6).

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Philadelphia Electric Company, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility and PSE&G, DP&LC, and ACEC to possess the facility at the designated location in Peach Bottom, York County, Pennsylvania in accordance with the procedures and limitations set forth in this license;

(2) PECO, pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material," to receive, possess, and use at any time in connection with operation of the facility:

a. Uranium enriched in the U-235 isotope, containing up to 6000 kilograms of U-235;

b. U-235 contained in sealed neutron detectors, not to exceed 0.99 grams;

(3) PECO, pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Licensing of By-Product Material," to receive, possess, and use in connection with operation of the facility:

a. Any by-product material with Atomic Numbers between 1 and 83, inclusive, in any form not to exceed 10 millicuries;

b. Antimony 124, in sealed sources, not to exceed 14 sources of 700 curies each;

c. Krypton 85, in gaseous form, not to exceed 1 curie;

d. Xenon 133, in gaseous form, not to exceed 1 curie;

e. Americium 241, in sealed or solid neutron sources, not to exceed 1 source of 5 curies, 1 source of 1 curie, and 1 source of 6 curies;

f. Cesium 137, in sealed or solid sources, not to exceed 2 sources of 10 curies each, 2 sources of 300 millicuries each, and 3 sources of 35 millicuries each;

g. Hydrogen 3, in liquid or gaseous form, not to exceed 100 millicuries.

(4) PECO, pursuant to the Act and 10 CFR Part 40, "Licensing of Source Material," to receive, possess, and use in connection with operation of the facility, natural or depleted uranium, not to exceed 3 millicuries; and

(5) PECO, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such by-product and special nuclear material as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PECO is authorized to operate the Peach Bottom Atomic Power Station, Unit 3, at steady state reactor core power levels not in excess of 3293 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B as issued with DPR-44 on August 8, 1973, and revised October 25, 1973 are hereby incorporated in this license as amended herein. Amended pages to Technical Specifications Appendix A issued as Change No. 1 to License No. DPR-44, dated February 12, 1974 and Change No. 2 to License No. DPR-44, dated April 19, 1974 are attached hereto. Also attached are amended pages 5, 5a and the cover sheet for Technical Specifications B, which modify the Technical Specifications B issued with DPR-44. These Technical Specifications are also hereby incorporated in this license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license is subject to the following conditions for the protection of the environment:

A. If the actual milk sample measurements taken at the nearby farms in accordance with the Technical Specifications, predict a dose to a child's thyroid, based on actual combined operation of Unit 2 and Unit 3, that exceeds 15 mrem/year, licensees shall install for operation no later than the next refueling cycle for each unit the necessary equipment to reduce the projected dose to such levels.

B. Operation of the facility with once-through, helper cooling towers will be permitted until January 1, 1977 and thereafter a closed-cycle cooling system will be required.

C. From among the alternative closed-cycle cooling systems considered in the licensees' Environmental Report and in the Regulatory staff's FES, the licensees shall select a preferred system for installation. A detailed evaluation of the economic and environmental impacts of the preferred system shall be submitted to the Atomic Energy Commission for review by December 14, 1974.

D. After approval by the Regulatory staff, the alternative cooling system shall be designed, built, and placed in operation by the licensees no later than January 1, 1977.

E. If installation of the preferred closed-cycle cooling system cannot be accomplished by the licensees by January 1, 1977, the licensees may apply in writing for the approval of the Regulatory staff to operate the facility at such reduced power levels that would insure

that there would not be violations of the federally-approved water quality standards of both Maryland and Pennsylvania, or violations of other provisions of the FWPCA. If satisfied that the reduced operation would not result in the aforementioned violations, the Regulatory staff may approve the reduced operation for the period during which installation of the closed-cycle system is being completed.

F. In the interim period during installation of the closed-cycle cooling system, all available cooling tower units shall be operated whenever either Unit 2 or Unit 3 is operating at a substantial power level.

G. The monitoring of the temperatures in the discharge canal may consist only of accurate measurement of the temperature of the coolant flow at the points of discharge and entrance to the discharge canal.

H. Philadelphia Electric Company shall submit to the Pennsylvania Department of Environmental Resources, by September 12, 1974, an application for modification of its Industrial Waste Permits #5681011 and #6769204 to reflect the changed treatment facilities (the closed-cycle cooling system) and a schedule for completing such facilities.

4. This license is effective as of the date of issuance and shall expire at midnight on January 31, 2008.

FOR THE ATOMIC ENERGY COMMISSION

A. Giambusso, Deputy ~~Director~~
for Reactor Projects
Directorate of Licensing

Attachments:
Amended pages to Appendices A & B
DPR-44 & DPR-56 Technical
Specifications

Date of Issuance: JUL 2 1974

ENCLOSURE 2

Docket No. 50-278

Philadelphia Electric Company

Peach Bottom Unit 3

Outstanding Items List

The following list of outstanding items must be completed prior to initial operation of the Peach Bottom Unit 3 reactor with the primary coolant temperature above 212°F:

1. Verify completion of review of preoperational tests in the radiological area. A review of results of the preoperational tests has been completed, except for interface testing between Unit 2/3.

2. Verify that the licensees document disposition of test exceptions for the ILRT.

3. Verify that the licensees replaced the drywell head flange gaskets prior to installation of the drywell head.

4. Verify that the licensees replaced drywell head bolts as necessary prior to installation of the drywell head.

5. Verify the accuracy of the instruments for measurement of Iodine-131 in effluents.

6. Verify completion, after interfaces of Units 2/3 have been lifted, of the uncompleted portions of S. U. Test No. 15/23, Demineralizer/ Condensate Tank and S. D. Test No. 64, Primary Containment Atmospheric Control.

7. Verify completion of preoperational test program with data within acceptable limits as stated below:

   a. The following preoperational tests will be completed following fuel loading:

      (1) Off Gas/Recombiner
      (2) CAD
      (3) Generator Seal Oil/CO_2 and H_2
      (4) Stator Cooling
      (5) Main Generator Exciter

   b. Complete the preoperational test completion register items, exceptions to preoperational tests and outstanding items from the preoperational tests which were conducted prior to issuance of the O.L.

8. Verify completion of rework, retest and reinstallation of Dresser Safety Valves.

9. Verify installation of orifice blocks in hydraulic snubbers.

10. Complete all Non-Conformance Reports items.

11. Confirm that all residual indications of cable tray overload have been corrected and complete the installation of cable tray barriers.

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-278

NOTICE OF AVAILABILITY OF INITIAL DECISION OF THE ATOMIC SAFETY AND LICENSING BOARD FOR THE PEACH BOTTOM ATOMIC POWER STATION, UNIT 3 AND ISSUANCE OF OPERATING LICENSE

Pursuant to the National Environmental Policy Act of 1969 and the United States Atomic Energy Commission's regulation in Appendix D, Section A.9 and A.11, to 10 CFR Part 50, notice is hereby given that an Initial Decision dated June 14, 1974, by the Atomic Safety and Licensing Board in the above captioned proceeding authorizing issuance of an operating license to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company (licensees) for authorization to operate the Peach Bottom Unit 3 facility located in York County, Pennsylvania, is available for inspection by the public in the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. and in the Martin Memorial Library, 159 E. Market Street, York, Pennsylvania 17401.

The Initial Decision is also being made available at the Office of State Planning and Development, 510c Finance Building, Harrisburg, Pennsylvania 17120, and at the York County Planning Commission, 1320 West Market Street, York, Pennsylvania 17404.

The Decision of the Atomic Safety and Licensing Board modified in certain respects the contents of the Final Environmental Statement

prepared by the Commission's Directorate of Licensing relating to the operation of the Peach Bottom Atomic Power Station, Units 2 and 3. A copy of this Final Environmental Statement is also available for public inspection at the above designated locations.

Pursuant to the provisions of 10 CFR, Part 50, Appendix D, Section A.11, the Final Environmental Statement is deemed modified to the extent that the findings and conclusions relating to environmental matters contained in the Initial Decision are different from those contained in the Final Environmental Statement, dated April 1973. As required by Section A.11 of Appendix D, copies of the Initial Decision by the Atomic Safety and Licensing Board and the Final Environmental Statement have been transmitted to the Council on Environmental Quality and made available to the public as noted herein.

Pursuant to the above mentioned Initial Decision, the Atomic Energy Commission (the Commission) has issued DPR-56 Facility Operating License to Philadelphia Electric Company, et al for operation of the Peach Bottom Atomic Power Station, Unit 3, a boiling water reactor, at steady state reactor core levels not to exceed 3293 megawatts thermal.

In addition to the Initial Decision, copies of (1) Facility Operating License No. DPR-56, (2) the report of the Advisory Committee on Reactor Safeguards, dated September 21, 1972, (3) the Directorate of Licensing's Safety Evaluation, dated August 11, 1972,

(4) Supplement No. 1 to the Safety Evaluation, dated December 11, 1972, (5) Supplement No. 2 to the Safety Evaluation, dated May 23, 1973, (6) Supplement No. 3 to the Safety Evaluation, dated October, 1973, (7) the Final Safety Analysis Report and amendments thereto, (8) the applicant's Environmental Report, dated June 4, 1971 and supplements thereto, (9) the Draft Environmental Statement dated October, 1972, and (10) the Final Environmental Statement, dated April, 1973, are also available for public inspection at the above-designated locations in Washington, D. C. and York, Pennsylvania. Single copies of the Initial Decision by the Atomic Safety and Licensing Board, Facility Operating License DPR-56, the Final Environmental Statement, and the Safety Evaluation and amendments may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing, Regulation.

Dated at Bethesda, Maryland this 2nd day of July 1974.

For the Atomic Energy Commission.

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Directorate of Licensing

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-171

AMENDMENT TO INDEMNITY AGREEMENT NO. B-28

AMENDMENT NO. 16

Effective July 2, 1974, Indemnity Agreement No. B-28 between Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company and the Atomic Energy Commission, dated December 31, 1964, as amended, is hereby further amended as follows:

Item 3 of the Attachment to the indemnity agreement is deleted in its entirety and the following substituted therefor:

Item 3 - License number or numbers

SNM-851	(From 12:01 a.m., December 31, 1964, to 12 midnight, January 23, 1966, inclusive)
SNM-1274	(From 12:01 a.m., November 17, 1971, to 12 midnight, August 7, 1973, inclusive)
SNM-1421	(From 12:01 a.m., October 31, 1973, to 12 midnight, July 1, 1974, inclusive)
DPR-12	(From 12:01 a.m., January 24, 1966)
DPR-44	(From 12:01 a.m., August 8, 1973)
DPR-56	(From 12:01 a.m., July 2, 1974)

FOR THE UNITED STATES ATOMIC ENERGY COMMISSION

Abraham Braitman, Chief
Office of Antitrust & Indemnity
Directorate of Licensing

Accepted _____, 1974

By _____
PHILADELPHIA ELECTRIC COMPANY

Accepted _____, 1974

By _____
PUBLIC SERVICE ELECTRIC & GAS CO.

Accepted _____, 1974

By _____
DELMARVA POWER & LIGHT COMPANY

Accepted _____, 1974

By _____
ATLANTIC CITY ELECTRIC COMPANY

Instructions for updating the Appendix A to Operating License DPR-44, Technical Specifications and Bases for the Full Power, Full Term Peach Bottom Atomic Power Station, Units No. 2 and 3 and making Appendix A applicable to the Unit 3 Operating License DPR-56.

1. Replace cover sheet with Appendix A. Revised June 1974.
2. Replace page 38 with April 1974 page 38.
3. Replace page 48 with April 1974 page 48.
4. Replace page 61 with April 1974 page 61.
5. Replace page 63 with April 1974 page 63.
6. Replace page 67 with April 1974 page 67.
7. Replace page 69 with April 1974 page 69.
8. Replace page 90 with April 1974 page 90.
9. Replace page 102 with February 1974 page 102.
10. Replace page 104 with February 1974 page 104.
11. Replace page 117 with April 1974 page 117.
12. Replace page 133a with April 1974 page 133a.
13. Replace page 166 with April 1974 page 166.
14. Insert page 234a dated April 1974.
15. Insert page 236a dated April 1974.
16. Replace page 257 with April 1974 page 257.
17. Replace pages 259 & 260 with April 1974 pages 259 & 259a.
18. Insert pages 260 & 261 dated June 1974 and April 1974.
19. Insert pages 261a & 262 dated April 1974 and April 1973.
20. Replace page 266 with April 1974 page 266.
21. Replace page 267 with June 1974 page 267.

Docket Nos. 50-277 O.L. DPR-44
50-278 DPR-56

Peach Bottom Atomic Power Station, Units 2 & 3

Issuance of the Peach Bottom Atomic Power Station, Unit 2 operating license DPR-44 included Appendix A. Technical Specifications and Bases for the Full Power Full Term Peach Bottom Atomic Power Station, Units No. 2 & 3. Since the issuance of the Unit 2 operating license there have been two changes to the Appendix A. The attached pages should replace the original Appendix A pages to incorporate the two approved changes and make the Appendix A applicable to the Unit 3 license. These replacement pages are being sent to all parties which received the original Appendix A.

APPENDIX A

TO

OPERATING LICENSES DPR-44 & DPR-56

TECHNICAL SPECIFICATIONS AND BASES

FOR

THE FULL POWER FULL TERM

PEACH BOTTOM ATOMIC POWER STATION

UNITS NO. 2 & 3

PHILADELPHIA ELECTRIC COMPANY

DOCKET NOS. 50-277 & 50-278

APPENDIX A
REVISED
JUNE 1974

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (7)	Startup	Run		
1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
2	APRM High Flux	(.66W+54)(2.63/P.F) (12)(13)			X	6 Instrument Channels	A or B
2	APRM Inoperative	(11)	X	X	X	6 Instrument Channels	A or B
2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	≤15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	≤1055 psig	X(9)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥0 in. Indicated Level	X	X	X	4 Instrument Channels	A

Table 3.1.1 (Cont'd.)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (7)	Startup	Run		
2	High Water Level in Scram Discharge Volume	≤50 Gallons	X(2)	X	X	4 Instrument Channels	A
2	Turbine Condenser Low Vacuum	≥23 in. Hg. Vacuum	X(3)	X(3)	X	4 Instrument Channels	A or C
2	Main Steam Line High Radiation	≤3 X Normal Full Power Background	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	8 Instrument Channels	A
2	Turbine Control Valve Fast Closure	500<P<850 psig Control Oil Pressure Between Fast Closure Solenoid and Disc Dump Valve			X(4)	4 Instrument Channels	A or D
4	Turbine Stop Valve Closure	≤10% Valve Closure			X(4)	8 Instrument Channels	A or D

T.S. Change #7

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertant criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference subsection 7.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e, an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

The APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in

3.1 BASES (Cont'd.)

the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

T.S. Change #2

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to limit fission product release so that 10 CFM Part 100 guidelines are not exceeded. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
2 (6)	Reactor Low Water Level	≥ 0" Indicated Level (3)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low Water Level	at or above -49" indicated level (4)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	≤ 2 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	≤ 3 X Normal Rated (8) Full Power Background	4 Inst. Channels	B
2	Low Pressure Main Steam Line	≥ 850 psig (7)	4 Inst. Channels	B
2 (5)	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤ 200 deg. F	4 Inst. Channels	B

T.S. Change #2

TABLE 3.2.A (Cont'd.)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
2	Main Steam Line Leak Detection High Temperature	≤ 200 deg. F	4 Inst. Channels	B
1	Reactor Cleanup System High Flow	≤ 300 % of Rated Flow	2 Inst. Channels	C
1	Reactor Cleanup System High Temperature	≤ 140 deg. F	1 Inst. Channel	C

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

3. Instrument set point corresponds to 177.7" above top of active fuel.

4. Instrument set point corresponds to 129.7" above top of active fuel.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

8. At a radiation level of 1.5 times the normal rated power background an alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level.

T.S. Change #2

TABLE 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Reactor Low-Low Water Level	≥ -49 in. indicated level	4 HPCI & RCIC Inst. Channels	Initiates HPCI & RCIC
2	Reactor Low-Low-Low Water Level	≥ -160 in. indicated level (4)	4 Core Spray & RHR Instrument Channels 4 ADS Instrument Channels	1. In conjunction with Low Reactor Pressure initiates Core Spray and LPCI 2. In conjunction with confirmatory low level High Drywell Pressure, 120 second time delay and LPCI or Core Spray pump interlock initiates Auto Blowdown (ADS) 3. Initiates starting of Diesel Generators

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	Core Spray Pump Start Timer	6 ± 1 sec 10 ± 1 sec	4 timers 4 timers	In conjunction with loss of power initiates the starting of CSCS pumps.
1	LPCI pump Start Timer	0 < t < 1 sec 5 ± 1 sec	2 timers 2 timers	
1	Auto Blowdown Timer	120 sec ± 5 sec	2 timers	In conjunction with Low Reactor Water Level, High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown.
2	RHR (LPCI) Pump Discharge Pressure Interlock	50 ± 10 psig	4 channels	Defers ADS actuation pending confirmation of Low Pressure core cooling system operation. (LPCI or Core Spray Pump running interlock.)
2	Core Spray Pump Discharge Pressure Interlock	185 ± 10 psig	4 channels	

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	RHR (LPCI) Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	NA	3 Inst. Channels	Monitors availability of power to logic systems.
1	HPCI Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Recirculation Pump A d/p	2 psid ± 1	4 Inst. Channels	Operates RHR (LPCI) break detection logic which directs cooling water into unbroken recirculation loop.
2	Recirculation Pump B d/p	2 psid ± 1	4 Inst. Channels	
2	Recirculation Riser d/p A>B	0.5 < p < 1.5 psid	4 Inst. Channels	
1	Core Spray Sparger to Reactor Pressure Vessel d/p	5 (±1.5) psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break.
2	Condensate Storage Tank Low Level	≥5' above tank bottom	2 Inst. Channels	Provides interlock to HPCI pump suction valves.
2	Supression Chamber High Level	≤ 5" above normal water level	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.

T.S. Change #2

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	RCIC Turbine High Flow	≤ 450" H_2O (2)	2 Inst. Channels	
2	RCIC Turbine Compartment Wall	≤ 200 deg. F (2)	4 Inst. } 16 Inst.	
6	RCIC Steam Line Area Temp.	≤ 200 deg. F (2)	12 Inst. }	
2	RCIC Steam Line Low Pressure	100 >P >50 psig (2)	4 Inst.	
1	HPCI Turbine Steam Line High Flow	≤ 225" H_2O (3)	2 Inst. Channels	

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (0") above the top of the active fuel closes all isolation valves except those in Groups 1, 4 and 5. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level this trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is 129.7" (-48") above the top of the active fuel (+9" on the instrument). This trip closes Main Steam Line Isolation Valves, Main Steam Drain Valves, Recirc Sample Valves (Group 1), initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 17.7"(-160") above the top of the active fuel. This trip activates the remainder of the CSCS subsystems, and starts

3.2 **BASES** (Cont'd)

the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low - low - low water level instrumentation; thus the results given above are applicable here also. See Spec. 3.7 for Isolation Valve Closure Group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures peak at approximately 1000°F and release of radioactivity to the environs is below CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel exhaust duct and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200°F for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR.

T.S. Change #2

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)
B. Control Rods	
1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.	b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position. c. During each refueling outage and after control rod maintenance, observe that the drive does not go to the overtravel position.
2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3. a. Whenever the reactor is in the startup or run modes below 30% rated power the Rod Sequence Control System shall be operable, that is no position switches shall be bypassed except as permitted in 3.3.A.2d, except during shut down margin testing. b. Whenever the reactor is in the startup or run modes below 25% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.	3. Prior to the start of control rod withdrawal towards criticality, and prior to attaining 25% rated power during rod insertion at shutdown, the capability of the Rod Worth Minimizer and Rod Sequence Control System to properly fulfill their functions shall be verified by the following checks: a. The capability of the Rod Sequence Control System to properly fulfill its function shall be verified by attempting to select and move a rod in each of the out-of-sequence groups.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)
c. Whenever the reactor is above 30% rated power, control rod patterns and the withdrawal or insertion sequences shall be established such that the maximum worth of any operable control rod, including an allowance for a selection error, is less than 1.25%Δk.	b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks: 1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified. 2. The RWM computer on line diagnostic test shall be successfully performed. 3. Prior to the start of control rod withdrawal only, proper annunication of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified. Change #1 4. The rod block function of the RWM shall be verified by withdrawing the first rod during start-up only as an out-of-sequence control rod no more than to the block point. T.S.
d. If Specifications 3.3.B.3.a through c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 25% rated power, it shall be brought to a shutdown condition immediately.	c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified and recorded.
e. Prior to the end of the first fuel cycle, analyses must be provided to the Directorate of Licensing to show that the Rod Sequence Control System will limit the control rod drop accident to a peak fuel enthalpy less than 280 calories per gram and doses to less than 10 CFR Part 100 guideline values for fuel loadings planned after the first fuel cycle.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either: a. Both RBM channels shall be operable: or b. Control rod withdrawal shall be blocked: or c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.	5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).
C. Scram Insertion Times	C. Scram Insertion Times
1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:	1. After each refueling outage all operable fully withdrawn insequence rods shall be scram time tested during operational hydrostatic testing or during startup from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to synchronizing the main turbine generator initially following restart of the plant.

Above 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	5.0

LIMITING CONDITION FOR OPERATION

3.3.C (cont'd.)

Below 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.475
20	1.10
50	2.0
90	5.0

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Above 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.398
20	0.954
50	2.120
90	5.300

Below 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.504
20	1.166
50	2.12
90	5.300

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

SURVEILLANCE REQUIREMENT

4.3.C (cont'd.)

After exceeding 30 percent power all previously untested operable control rods shall be tested as described above prior to exceeding 40 per cent power.

2. Whenever such scram time measurements are made (such as when a scram occurs and the scram insertion time recorders is operable) an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

T.S. Change #1

APPENDIX B

TO

OPERATING LICENSE DPR-44 AND DPR-56

ENVIRONMENTAL

TECHNICAL SPECIFICATIONS AND BASES

FOR

THE FULL POWER FULL TERM

PEACH BOTTOM ATOMIC POWER STATION

UNITS NO. 2 & 3

PHILADELPHIA ELECTRIC COMPANY

DOCKET NOS. 50-277 & 50-278

APPENDIX B
REVISED
JULY 1974

2.0 PROTECTION LIMITS

2.1 Thermal

Objective

To minimize adverse effects on the aquatic biota in the Conowingo Pond which could result from the increase in water temperature.

Specification

All three cooling towers shall be operated whenever Unit 2 or Unit 3 is operating at or above 20% of rated thermal power. For purposes of maintenance or repair, one cooling tower may be inoperable for a period not to exceed 48 hours. Such a shutdown shall be considered routine if the average flow of the Susquehanna River through Conowingo Pond as calculated daily exceeds 10,000 cfs and shall be reported as per Section 7.4.1. Cooling tower inoperability shall be considered an Environmental Deviation and shall be reported as per Section 7.4.2 if a) more than one tower is affected, b) the average flow of the Susquehanna River through Conowingo Pond as calculated daily is less than 10,000 cfs or c) full operation of all towers is not regained within 48 hours.

3.0 MONITORING REQUIREMENTS

3.1 Thermal

Objective

To assure that the water temperatures in the Conowingo Pond are not increased to a level that adverse effects on the aquatic biota would result.

Specification

The bulk temperature of the condenser inlet water and of the water discharged to Conowingo Pond from the discharge canal shall be monitored continuously. Temperatures shall be recorded and shall be transmitted to the PBAPS control room. For purposes of maintenance, the temperature monitoring system may be inoperable for a period not to exceed 7 days.

Thermal impact upon Conowingo Pond shall be periodically monitored and reported as specified in Section 6.3.

PROTECTION LIMITS (Continued)

Bases

A cooling tower shall be considered operable if its lift pump and 10 of the 11 cooling fans are operable.

The average flow of the Susquehanna River through Conowingo Pond will be calculated daily using predicted Holtwood Dam, Muddy Run, and Conowingo Dam operation and flows.

MONITORING REQUIREMENTS (Continued)

Bases

As indicated in "Predicted Prototype Temperature Distribution in Conowingo Pond," prepared for the Philadelphia Electric Company by the Bechtel Power Corporation in July, 1973, adherence to this specification will assure, in essentially all instances, compliance to the thermal criteria of the Maryland Water Quality Standard.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
C. Sodium Pentaborate Solution	C. Sodium Pentaborate Solution
At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:	The following tests shall be performed to verify the availability of the Liquid Control Solution:
1. The net volume versus concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4.1.	1. Volume: Check and record at least once per day.
2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2. This includes the piping between the standby liquid control tank and the suction inlet to the pumps.	2. Temperature: Check and record at least once per day.
3. If the concentration, volume, or temperature is outside the limits established by 3.4.C.1 or 3.4.C.2, the reactor shall be shutdown in accordance with 3.4.D unless the concentration, volume, and temperature is restored within allowable limits within the same time period.	3. Concentration: Check and record at least once per month. Also check concentration anytime water or boron is added to the solution or solution temperature is below the temperature required in Figure 3.4.2.
D. If specification 3.4.A through C cannot be met, the reactor shall be placed in a Cold Shutdown Condition with all operable control rods fully inserted within 24 hours.	

T. S. Change #2

INTENTIONALLY LEFT BLANK

3.5I Average Planar LHGR

During steady state power operation, the the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1.

3.5.J Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \left[1-\left(\frac{\Delta P}{P}\right)_{max}\left(\frac{L}{L_T}\right)\right]$$

$LHGR_d$ = Design LHGR = 18.5 kw/ft.

$\frac{\Delta P}{P}_{max}$ = Maximum power spiking penalty

= 0.037 unit 2
= 0.032 unit 3

L_T = Total core length

= 12 ft. unit 2
= 12.167 ft unit 3

L = Axial position above bottom of core

T.S. Change #2

4.5.I Average Planar LHGR

Daily during reactor power operation, the average planar LHGR shall be checked.

4.5.J. Local LHGR

Daily during reactor power operation, the local LHGR shall be checked.

3.5.A BASES

Core Spray and LPCI Subsystems

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant analysis included in General Electric Topical Report NEDO-10329 and the sensitivity studies given in Supplement 1 thereto and subsection 6.7 of the FSAR and the Philadelphia Electric Company letter dated August 26, 1971, and in accordance with the AEC's "Interim Acceptance Criteria for Emergency Core Cooling Systems" published on June 19, 1971, any of the following cooling systems provides sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel clad temperature to less than 2300°F to assure that core geometry remains intact, and to limit clad metal-water reaction to less than 1%; the two core spray subsystems; or either of the two core spray subsystems and two RHR pumps operating in the LPCI mode with operable LPCI injection valves.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Peach Bottom 2 and 3, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 105 psig.

The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem and the core spray subsystem provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling subsystems.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water volume - 122,000 ft^3
   b. Maximum water volume - 136,000 ft^3
   c. Maximum suppression pool temperature during normal power operation - 90°F
   d. Maximum suppression pool temperature during RCIC operation - 130°F
   e. In order to continue reactor power operation after being on RCIC operation, the suppression chamber temperature must be reduced to 90°F within 24 hours following the return to reactor power operation.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per day.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A Primary Containment	4.7.A Primary Containment
2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 Mw(t).	2. Integrated Leak Rate Testing
3. If the primary containment integrity is breached when it is required by 3.7.A.2, that integrity shall be reestablished within 24 hours or the reactor placed in a cold shutdown condition within 24 hours.	a. Integrated leak rate tests (ILRT's) shall be performed to verify primary containment integrity. Primary containment integrity is confirmed if the leakage rate does not exceed the equivalent of 0.5 percent of the primary containment volume per 24 hours at 49.1 psig.
	b. Integrated leak rate tests may be performed at either 49.1 psig or 25 psig, the leakage rate test period, extending to 24 hours of retained internal pressure. If it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period, the agreed-upon shorter period may be used.
	Prior to initial operation, integrated leak rate tests must be performed at 49.1 and 25 psig (with the 25 psig test being performed prior to the 49.1 psig test) to establish the allowable leak rate (in percent of containment volume per 24 hours) at 25 psig as the lesser of the following values:
	(L_a is 0.5 percent)
	$L_t = 0.5 \frac{L_{tm}}{L_{pm}}$

T.S. Change #2

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.11.D. Seismic Hydraulic Snubbers On Safety Related Systems	4.11.D. Seismic Hydraulic Snubbers On Safety Related Systems
1. During reactor power operation, the hydraulic snubbers installed on safety related piping systems must be operable except as specified in Spec. 3.11.D.2.	1. All hydraulic snubbers installed on safety related systems shall be visually inspected at the following frequency: a. Whenever the reactor is shut down for 72 hours or longer and the snubbers have not been inspected for 30 days. b. The maximum inspection interval of accessible units shall be 6 months.
2. If accessible hydraulic snubbers on safety related systems are observed to be inoperable at times other than the prescribed surveillance, continued operation is permissible for a period of seven days. If the defective snubbers have not been repaired within that period the reactor shall be shut down.	2. If the results of the inspection performed at the frequency required by Specification 4.11.D indicate reliable snubber operation for one operating cycle, the frequency of inspection shall be decreased to one per operating cycle.
3. The reactor shall not be started up with any safety related snubbers known to be inoperable.	3. Once per operating cycle a 2-1/2" snubber installed in a location representative of the worst environmental conditions shall be removed and disassembled for the purpose of inspection for seal degradation.

T.S. Change No. 2

4.11 Bases

D. Seismic Hydraulic Snubbers On Safety Related Systems

Hydraulic snubbers are necessary on safety related system piping to restrain seismic motion and yet allow unrestrained thermal growth.

Should a defective hydraulic snubber be discovered during reactor operation seven days is a reasonable time for repair without affecting operation. This allowance is justifiable since the probability of a seismic event during this period is considered low.

4.11 Bases

D. Seismic Hydraulic Snubbers on Safety Related Systems

A periodic inspection is required to establish their reliability and the compatibility of the seal materials with the hydraulic fluid and the operating environment.

It is considered that continued proper operation through one operating cycle and satisfactory findings of periodic inspections is sufficient justification to extend the inspection interval. The interval of once per operating cycle will minimize the effects on power operation.

Disassembly of a snubber installed in a location representative of the worst environmental conditions will provide an indication of the degree of hydraulic cylinder seal degradation.

T.S. Change No. 2

TABLE 6.5.1

PROTECTION FACTORS FOR RESPIRATORS

Description	Modes[1]	Protection Factors[2] Particulates and vapors and gases except tritium oxide[3]	Protection Factors[2] Tritium Oxide
I. Air-purifying respirator:			
Facepiece, half-mask	NP	10	1
Facepiece, full	NP	100	1
Facepiece, half-mask or full, or hood	PP	100	1
II. Atmosphere-supplying respirator:			
1. Air-line respirator:			
Facepiece, half-mask	CF	100	2
Facepiece, half-mask	D	100	2
Facepiece, full	CF	1000	2
Facepiece, full	D	100	2
Facepiece, full	PD	1000	2
Hood	CF	1000	2
Suit	CF	(4)	(4)
2. Self-contained breathing apparatus (SCBA):			
Facepiece, full	D	100	2
Facepiece, full	PD	1000	2
Facepiece, full	R	100	2
III. Combination respirator: Any combination of air-purifying and atmosphere-supplying respirator.		Protection factor for type and mode of operation as listed above.	

[1]See the following symbols:
CF: continuous flow.
D: demand
NP: negative pressure (i.e. negative phase during inhalation).

TABLE 6.5.1 (Continued)

PD: pressure demand (i.e. always positive pressure).
PP: positive pressure.
R: recirculating (closed circuit)

[2](a) for purposes of this specification, the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to determine the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:
(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.
(ii) for air-purifying respirators only when high efficiency particulate filters and/or sorbent appropriate to the hazard are used in atmospheres not deficient in oxygen.
(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.
[3]Excluding radioactive contaminants that present an absorption or submersion hazard.
[4]Appropriate protection factors must be determined taking account of the design of the suit and its permeability to the contaminant under conditions of use. No protection factor greater than 1000 shall be used except as authorized by the Commission.

Note 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. These protection factors may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines in accordance with their applicable schedules.
Note 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

6.6 Plant Operating Records

6.6.1 Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for five years unless a longer period is required by applicable regulations:

1. Records of normal plant operation, including power levels and periods of operation at each power level.

2. Records of principal maintenance activities, including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.

3. Records of abnormal occurrences.

4. Records of periodic checks, inspection and/or calibrations performed to verify that surveillance requirements are being met.

5. Records of any special reactor tests or experiments.

6. Records of changes made in the Operating Procedures.

7. Records of radioactive shipments.

6.6.2 The following records and/or logs will be permanently retained:

1. Updated and corrected drawings of the plant.

2. Records of new and spent fuel inventory and assembly histories.

3. Records of monthly plant radiation and contamination surveys.

4. Records of off-site environmental monitoring surveys.

5. Records of radiation exposure of all plant personnel, including all contractors and visitors to the plant who enter radiation control areas.

6. Records of radioactivity in liquid and gaseous wastes released to the environment.

7. Design Fatigue Usage Evaluation

1. Monitoring, recording, evaluating, and reporting requirements contained in Section 6.7.1.C will be met for various portions of the reactor coolant pressure boundary (RCPB) for which detailed fatigue usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed[1] for the conditions defined in the design specification. The location to be monitored shall be:

T. S. Change #2

a. The feedwater nozzles
b. The shell at or near the waterline
c. The flange studs

2. Monitoring, Recording, Evaluating and Reporting

a. Operational transients that occur during plant operation will, at least semi-annually, be reviewed and compared to the transient conditions defined in the component stress report for the locations listed in 1 above, and used as a basis for the existing fatigue analysis.

b. The number of transients which are comparable to or more severe than the transients evaluated in the stress report code fatigue usage calculations will be recorded in an operating log book. For those transients which are more severe, available data, such as the metal and fluid temperatures, pressures, flow rates, and other conditions will be recorded in the log book.

c. The number of transient events that exceed the design specifications quantity and the number of transients events with a severity greater than that included in the existing code fatigue usage calculations shall be added. When this sum exceeds the predicted number of design condition events by twenty five[2], a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

T. S. Change #2

6.7 Plant Reporting Requirements

In addition to applicable regulations the following reports will be submitted.

6.7.1 Routine Reports-Operations Reports

A. Startup Report

A summary report of plant startup and power escalation testing and the evaluation of the results from these test programs shall be submitted. The test results shall be compared with design predictions and specifications. The startup reports will be submitted within 90 days following commencement of commercial power operation.

Footnote:

1) See paragraph N-415.2, ASME Section III, 1965 Edition

2) The code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraph N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition)

T.S. Change #2

B. First Year Operation Report

A report should be submitted within 90 days after completion of the first year of operation. This report, however, may be incorporated into the semiannual operating report and should cover the following:

1. an evaluation of plant performance to date in comparison with design predictions and specifications;

2. a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analyses;

3. an assessment of the performance of structures, systems, an components important to safety;

4. a progress and status report on any items identified as requiring additional information during the operating license review or during the startup of the facility, including items discussed in the AEC's safety evaluation, conditions of the license, and items identified in the licensee's startup report.

C. Semiannual Operating Reports

Routine operating reports will be submitted within 60 days following the end of each 6-month period ending June 30 and December 31. The first such period shall begin with the date of initial criticality. These reports will include the following:

1. Operations Summary

   A summary of operating experience occurring during the reporting period that relates to the safe operation of the facility, including a summary of:

   a. changes in facility design

   b. performance characteristics (e.g., equipment and fuel performance)

   c. changes in procedures which were necessitated by (a) and (b) or which otherwise were required to improve the safety of operations.

   d. results of surveillance tests and inspections required by the licensee's technical specifications,

e. the results of any periodic containment leak rate tests performed during the reporting period,

f. a brief summary of those changes, tests, and experiments requiring authorization from the Commission pursuant to 10 CFR 50.59 (a),

g. any changes in the plant operating organization which involve positions for which minimum qualifications are specified in the technical specifications, and

h. Design Fatigue Usage Evaluation Report

T. S. Change #2

In the semi-annual Operating Report, a listing of the number of events identified in 6.6.2.7.2.b above will be tabulated and compared to the design or allowed quantity of comparable or more severe events. In those cases where recalculation of fatigue usage is required per 6.6.2.7.2.c and the calculated usage exceeds two times the design usage limit of the code, the report will define the inservice inspection that will be performed on that portion of the RCPB to monitor for crack initiation.

2. Power Generation

A summary of power generated during the reporting period including:

a. gross thermal power generated,

b. gross electrical power generated,

c. net electrical power generated,

d. number of hours the reactor was critical,

e. number of hours the generator was online,

f. histogram of thermal power versus time.

3. Shutdowns

Descriptive material covering outages occurring during the reporting period. For each outage, information shall be provided on:

a. the cause of the outage,

b. the method of shutting down the reactor; e.g. scram, automatic rundown, or manually controlled deliberate shutdown,

c. duration of the outage,

d. plant status during the outage; e.g. cold shutdown or hot standby,

e. corrective action taken to prevent repetition, if appropriate.

6.7.1.C.4. Maintenance

A discussion of safety-related maintenance (excluding preventative maintenance) performed during the reporting period on safety related systems and components. The discussion shall include the following information:

a. the system or component involved,

b. the cause of the malfunction,

c. the results and effect on safe operation,

d. corrective action taken to prevent repetition,

e. precautions taken to provide for reactor safety during repair,

f. time required for completion

5. Changes, Tests, and Experiments

A summary of all changes in the facility design and procedures that relate to the safe operation of the facility which do not require authorization from the Commission pursuant to 10 CFR 50.59(a). A brief description and summary of the safety evaluation for those changes, tests, and experiments, carried out without prior Commission approval pursuant to the requirements of 10 CFR 50.59 (b) will be included.

6. Effluent Releases

Effluent data should be summarized monthly, except in instances when more detailed data is needed, and the items listed below reported semiannually on the standard form "Report of Radioactive Effluents".

a. <u>Gaseous Releases</u>

1. total radioactivity released (in curies) of noble and activation gases.

2. maximum noble gas release rate during any one-hour period.

3. total radioactivity released (in curies) by nuclide, based on representative isotopic analyses performed.

4. percent of technical specification limit.

b. <u>Iodine Releases</u>

1. total (I-131, I-133, I-135) radioactivity released (in curies).

Directorate of Licensing (copy to the AEC Region I Director, Directorate of Regulatory Operations), in the event of an abnormal occurrence as defined in these technical specifications.

The written report on abnormal occurrences, and to the extent possible, the preliminary telephone and telegraph notification, should: (a) describe, analyze, and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is ascertained and (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems.

In addition, the written report should relate any failures or degraded performance of system and components for the incident to similar equipment failures that may have previously occurred at the facility. The evaluation of the safety implications of the incident should consider the cumulative experience obtained from the record of previous failures and malfunctions of the affected systems and components or of similar equipment.

B. Reporting of Events

1. A written report shall be forwarded within 30 days to the Director, Directorate of Licensing, and to the AEC Region I Director, Directorate of Regulatory Operations, in the event of:

   a. Discovery of any substantial changes in the transient or accident analyses, or in the methods used for such analysis as described in the safety analysis report or in the bases for the technical specifications.

   b. Any substantial variance from performance specifications contained in the technical specifications or in the safety analysis report.

   c. Any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

   d. When the release rate of radioactive liquid effluents, excluding tritium and noble gases, exceeds 2.5 curies per unit during any calendar quarter. The report shall identify the causes of the high release rate and shall describe the proposed program of action to reduce such release rates.

B. Reporting of Unusual Events (Cont'd.)

2. A written report shall be forwarded within 10 days to the Director, Directorate of Licensing, and to the director of the Regional Compliance Office when the release rate of gaseous radioactive effluents exceeds 4% of the limits of specifications 3.8.C.1 or 3.8.C.2 for a period of greater than 48 hours. The contents of the report shall be in accordance with specification 3.8.C.5.

3. Special Reports which shall be submitted in writing to the Directorate of Licensing, USAEC, Washington D.C. 20545. Reports on the following areas shall be submitted as noted:

Area	Reference	Submittal Date
a. Primary Containment Leak Rate Testing	4.7A	Upon completion of test.
b. Secondary Containment Leak Rate Testing	4.7C	Upon completion of test.

C. Reporting Results of Inspections of Seismic Hydraulic Snubbers

1. Notification of snubber failure shall be reported as an abnormal occurrence in accordance with 6.7.2.A.

2. If no defective snubbers are identified, a report shall be submitted in accordance with paragraph 6.7.1.C.1.d.

T.S. Change No. 2

6.8 INDUSTRIAL SECURITY PROGRAM

6.8.1 An industrial security program shall be maintained throughout the life of the plant in accordance with the provisions of the Plant Security Plan. Annual review of the Plant Security Plan shall be performed.

6.8.2 Any actual or attempted introduction into the Generating Station Area of any dangerous weapon, explosive or material capable of producing injury or damage to persons or property, or that in any way could seriously affect the safe operation of the generating units, shall be reported immediately upon detection to the shift supervisor and plant security forces.

6.8.3 Investigations shall be conducted by the Station Superintendent of all attempted or actual security infractions in cooperation with any Federal, State, or local agencies involved, and a report filed with the Departmental Management and Chairman of the O and SR committee.

6.8.4 The General or Assistant General Superintendent, Generation Div. shall report all instances of security violations described in Section 6.8.2 as specificed in Section 6.7.2A.

6.9 EMERGENCY PLAN

6.9.1 A Site Emergency Plan shall be maintained throughout the life of the plant in accordance with the provisions of 10 CFR 50, Appendix "E".

6.9.2 Site evacuation exercises shall be conducted in accordance with the Emergency Plan, FSAR Appendix O Section 11. This exercise shall involve coordination with offsite support groups and include communication checks.

6.9.3 The Emergency Plan and implementing procedures shall be reviewed and updated on an annual basis.

6.10 INITIAL OPERATION OF UNIT 3 WITH PRIMARY COOLANT TEMPERATURE ABOVE 212°F

6.10.1 The licensee shall not attempt initial operation of Unit 3 such that the primary coolant temperature exceeds 212°F until specifically approved in writing by the Commission.

INTENTIONALLY LEFT BLANK