OCTOBER 24 1979

Distribution
Docket
ORB #3
NRR Reading
Local PDR
NRC PDR
DEisenhut
WGammill
JMiller
LShao
BGrimes
RVollmer
TIppolito
DVerrelli
PKreutzer
Atty, OELD
OI&E (5)
BJones (5)
BScharf (10)
ACRS (16)
CMiles, OPA
RDiggs
HDenton
TERA
JRBuchanan

Docket No. 50-277
and 50-278

Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Bauer:

The Commission has issued Amendment Nos. 63 and 62 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Unit Nos. 2 and 3. These amendments revise the Technical Specifications and are in response to your applications dated May 8 and August 2, 1979 as supplemented by information contained in your letter dated October 2, 1979.

The amendment for Peach Bottom Unit 3 (DPR-56) involves:

(1) use of pre-pressurized fuel for Cycle 4 operation;

(2) modification of the APRM and RBM setpoint equations;

(3) deletion of the fuel densification power spiking penalty for 8x8 fuel;

(4) deletion of the reactor vessel pressure operating limit;

(5) continued use of the fast scram control rod drive during Cycle 4;

(6) increase of the standby liquid control system capacity;

(7) addition of a license condition which governs operation during any coastdown after end-of-cycle;

(8) revision of the withdrawal schedule for the reactor vessel material surveillance program, and

(9) administrative changes relating to reporting of primary and secondary leak rate test results, members of the Operation and Safety Review Committee, the table specifying fire detectors and correction of typographical errors.

7911020309

CP

OFFICE➤						
SURNAME➤						
DATE➤						

The amendment for Peach Bottom Unit 2 (DPR-44) involves the reactor vessel material surveillance program and administrative changes (items 8 and 9 above).

Copies of our Safety Evaluation and a related Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 63 to DPR-44
2. Amendment No. 62 to DPR-56
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

No legal objection to form of notice or license amend; SER not reviewed.

OFFICE ➤	ORB #3	ORB #3	AD:ORP	OELD	ORB #3	
SURNAME ➤	PKreutzer	DVerrelli:mjf	WGammill	Cutchin	TIppolito	
DATE ➤	10/23/79	10/19/79	10/19/79	10/22/79	10/19/79	

cc:

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Government Publications Section
State Library of Pennsylvania
Education Building
Commonwealth and Walnut Streets
Harrisburg, Pennsylvania 17126

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

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. DPR-44

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Philadelphia Electric Company, et al., (the licensee) dated May 8, 1979, 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;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 24, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 63

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
144	144
240e	240e
240f	240f
240h	240h
240i	240i
240k	240k
240m	240m
249	249
250	250
257	257
-	257a

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.A Thermal and Pressurization Limitations (Cont'd) Figures 3.6.1, 3.6.2, and 3.6.3 will be updated to account for radiation damage prior to 9 effective full power years of operation.	4.6.A Thermal and Pressurization Limitations (Cont'd) Selected neutron flux specimens shall be removed* and tested to experimentally verify or adjust the calculated values of integrated neutron flux that are used to determine the RT_{NDT} for Figure 3.6.4.
3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100°F.	3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.
4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.	4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.
5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.	5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.
6. Reactor vessel pressure shall not exceed 1020 psig at any time during normal steady state reactor power operation. In the event that this LCO is exceeded, steps shall be immediately initiated to reduce the pressure below 1020 psig. If this cannot be done, shutdown to cold conditions shall be accomplished within 24 hours.	6. The reactor pressure shall be logged once per day. *Specimen 1 7-9 EFPY 2 15-18 EFPY 3 Standby

Amendment No. ~~41~~, ~~45~~, 63

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. With one fire pump or logic inoperable, restore the equipment to an operable status within 7 days, or in lieu of any other report required by Specification 6.9.2, submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the equipment to an operable status. Reactor startup and/or continued reactor operation is permissible.	2. With one fire pump inoperable, the remaining fire pump shall be demonstrated to be operable immediately and at least every 72 hours thereafter until the inoperable pump is restored to an operable status.
3. With two fire pumps inoperable, a. establish a back-up water supply within 24 hours, b. notify the Commission pursuant to Specification 6.9.2.a within 24 hours, by telephone and in writing no later than the first working day following the event. Submit a report within 14 days outlining the actions taken and the plans and schedule for restoring the equipment to an operable status and, c. restore the equipment to an operable status within 14 days. d. If a. above cannot be fulfilled, place the reactor in Hot Standby within the next six hours and in Cold Shutdown within the following thirty hours.	3. None
4. Except as specified in 3.14.A.6 below, the fire hose stations serving the following structures shall be operable: a. Reactor Buildings b. Radwaste Building	4. The fire hose station inspections shall be performed as follows: a. Visual inspection of hose station equipment availability - once every 31 days. b. Hose and gasket inspection - once every 18 months.

LIMITING CONDITIONS FOR OPERATION

3.14.A (Cont'd)

c. Turbine Building
d. Circulating Water Pump Structure

6. When a hose station serving an area which contains equipment which is required to be operable becomes inoperable, establish a continuous fire watch equipped with portable fire suppression equipment within 1 hour and provide equivalent protection to the area served by the inoperable station from an operable hose station within 6 hours.

7. Except as specified in 3.14.A.8 below, the fire suppression spray system serving a Standby Gas Treatment System charcoal filter train shall be operable when a train is required to be operable.

8. If the requirements of 3.14.A.7 cannot be met,

   a. establish a fire watch patrol to inspect the area with inoperable fire suppression equipment at least once per shift,

   b. restore the system to an operable status within 14 days, or in lieu of any other report required by Specification 6.9.2 submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the system to an operable status. The SGTS may be considered operable for the purposes of Specification 3.7.B.

SURVEILLANCE REQUIREMENTS

c. Hose station valve operability and blockage check - once every 3 years.

d. Hose hydrostatic test at a pressure of 250 psig or replace with an appropriately tested hose every 3 years.

6. None

7. The SGTS fire suppression spray system testing shall be performed as follows:

   a. Simulated automatic actuation test- Once every 18 months.

   b. Inspection of nozzles and spray header- Once every 18 months.

   c. Header and nozzle air flow test- Once every 3 years.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.14.B.3 (Cont'd) b. an operable flow path to each room and c. four heat detectors, except that one detector may be inoperable for a period not to exceed 7 days. 4. If the requirements of 3.14.B.1, 2, or 3 cannot be met, a. establish a continuous fire watch with back-up fire suppression equipment for the unprotected area (HPCI, Cable Spreading, Computer, Diesel Generator) within 1 hour b. restore the system to an operable status within 14 days, or in lieu of any other report required by Specification 6.9.2, submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the system to an operable status. Reactor startup and/or continued reactor operation is permissible.	

LIMITING CONDITIONS FOR OPERATION

3.14.C Fire Detection

1. The fire detection instrumentation for each plant area listed in Table 3.14.C.1 shall be operable when the equipment in that area is required to be operable.

2. If the requirements of 3.14.C.1 cannot be met,

   a. establish a fire watch patrol to inspect each accessible area at intervals of at least:

      (1) Once per shift for areas with less than the minimum number of operable instruments required by Table 3.14.C.1 but with at least one instrument operable

      (2) Once every hour for areas without an operable instrument,

   b. restore accessible system components to an operable status within 14 days, or in lieu of any other report required by Specification

   c 6.9.2, submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the instruments to an operable status. Reactor startup and/or continued reactor operation is permissible.

SURVEILLANCE REQUIREMENTS

4.14.C Fire Detection

1. a. The smoke detectors listed in Table 3.14.C.1 shall be functionally tested semi-annually in accordance with the manufacturer's instructions.

   b. The heat detectors listed in Table 3.14.C.1 shall be functionally tested semi-annually with a heat source.

   c. The NFPA Code 72D Class A supervised circuits between the local panel and control room of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

2. The testing interval for smoke and heat detectors which are inaccessible due to high radiation or inerting may be extended until such time as the detectors become accessible for a minimum of 36 hours. Such detectors shall be functionally tested at a maximum interval of once per refueling cycle.

Table 3.14.C.1

FIRE DETECTORS

Location	Detector Type/ Designation (1)	Minimum Detectors Operable
Unit 2:		
Primary Containment (2) (3)	S1, S2, S8	3
Recirculation Pump MG Set Room	S15, S16, S17 S18, S19, S20	5
Emergency Switchgear Rooms	S11, S12, S13 S14	1 per room
Unit 3:		
Primary Containment (2) (3)	S103, S104, S106	3
Recirculation Pump MG Set Room	S111, S112, S113 S114, S116, S117	5
Emergency Switchgear Rooms	S107, S108, S109 S110	1 per room
Common:		
Control Room	S21, S22, S23, S24	4
Cable Spreading Room	S4, S7, S9, S10	4
Computer Room	S5, S6	2
Laboratory Area	H1, H2, H3, H4	4
Fan Area	S3, S105	2
Emergency Cooling Tower Switchgear Rooms	H562, H563, H564 H565	1 per room
HPSW Pump Structure	H397, H398	2
Recombiner Building	H566, H567, H568	3
Start-up Switchgear Building	H558, H559 H560, H561	2

(1) S = Smoke Detector   H = Heat Detector

(2) Detector(s) inaccesible during normal operation due to inerting.

(3) May be disabled during ILRT.

4.14 BASES

A. Water Fire Protection System

The monthly test of the fire pumps is conducted to check for equipment failures and deterioration. The fire pump minimum capacity is based on a design load of 2400 gpm for the largest sprinkler plus 300 gpm for manual hose lines.

When it is determined that a fire pump is inoperable, the increased surveillance required by 4.14.A.2 provides adequate assurance that the remaining pump will be operable when required.

B. CO2 Fire Protection Systems

Weekly checking of the storage tank level and pressure is deemed adequate to provide assurance that sufficient CO2 will be available in the event of a fire occurrence.

Semi-annual testing of the heat detectors in the automatic discharge systems is in accordance with NFPA-72E-1974.

Testing of the discharge initiation logic, injection valve, damper closings, and fan trippings without actual discharge of CO2 into a room demonstrates operability of the active components of the systems. System operability is demonstrated by both manual and automatic initiation for automatic discharge systems (HPCI and diesel generators). Testing of the headers and nozzles by an air flow test will detect buildups of material which may affect continued availability.

C. Fire Detection

Semi-annual testing of fire detectors is in accordance with NFPA-72E-1974.

D. Fire Barrier Penetrations

Penetration fire barrier seals are visually inspected to verify that they are functional.

6.5.2 Operation and Safety Review Committee

Function

6.5.2.1 The Operation and Safety Review Committee shall function to provide independent review and audit of designated activities in the area of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

(the members of the OSR Committee will be competent in the area of quality assurance practice and cognizant of the Quality Assurance requirements of 10 CFR 50, Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance Organization available to them.)

Organization

6.5.2.2 The Chairman, Members and Alternate Members of the ONSR Committee shall be appointed in writing by the Vice President, Electric Production, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in one or more areas given in 6.5.2.1.

Alternates

6.5.2.3 Each permanent member shall have a designated alternate to serve in his absence, and a current list of these alternates shall be maintained in Committee records. Each alternate member will serve on a continuing basis.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the OSR Committee Chairman to provide expert advise to the OSR Committee.

Meeting Frequency

6.5.2.5 The OSR Committee shall meet at least once per six months.

Quorum

6.5.2.6 A quorum of the OSR Committee shall consist of the Chairman or Vice Chairman or their designated alternates and four members or their alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

6.5.2.7 The OSR Committee shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

6.9.2 Continued

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

6.9.3 Unique Reporting Requirements

Special reports shall be submitted to the Director of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Loss of shutdown margin, Specification 3.3.A and 4.3.A within 14 days of the event.

b. Reactor vessel inservice inspection, Specification 3.6.G and 4.6.G within 90 days of the completion of the reviews.

c. (Deleted)

d. Primary containment leak rate testing approximately three months after the completion of the periodic integrated leak rate test (Type A) required by Specification 4.7.A.2.c.2. For each periodic test, leakage test results from Type A, B and C tests shall be reported. B and C tests are local leak rate tests required by Specification 4.7.A.2.f. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test.

6.9.3 Continued

e. Release rate of Radioactive Effluents, Specification 3.8.B.7, 3.8.C.3.b, 3.8.C.5.

f. Sealed source leakage in excess of limits, Specification 3.13.3.

g. Effluent Releases

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

(1) Gaseous Releases

(a) Total radioactivity released (in curies) of noble and activation gases.

(b) Maximum noble gas release rate during any one-hour period.

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

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 NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 62
License No. DPR-56

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Philadelphia Electric Company, et al., (the licensee) dated May 8 and August 2, 1979 as supplemented October 2, 1979, comply 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;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-56 is amended by revising paragraph 2.C(2) and adding paragraph 2.C(5) to read as follows:

   (2) The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 62, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (5) Operation beyond the end-of-cycle (all rods out condition) thermal power is limited to seventy (70) percent minimum.

Increasing core power level via reduced feedwater heating, once operation in the coastdown mode has begun, is not permitted unless the licensee has performed an analysis of this operating condition that confirms that this condition is bounded by the analysis for the particular cycle of operation.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. P. Gammill

W. P. Gammill, Acting Assistant Director
for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 24, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 62

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iv	iv
v	v
1	1
3	3
7	7
10	10
11	11
14	14
15	15
15a	(deleted)
15b	(deleted)
18	18
19	19
20	20
33	33
35	35
37	37
40	40
54	54
73	73
74	74
108	108
111	111
115	115
119	119
120	120
121*	121*
122	122
133a	133a
133c	133c
137*	137*
138	138
140	104
140a	140a
140c	140c
140d	140d
140e	142e
-	142g (added)
144	144
152a	152a
157	157

*Overleaf

Remove	Insert
240e	240e
240f	240f
240h	240h
240k	240k
240m	240m
241	241
249	249
250	250
257	257
257a	257a

LIST OF FIGURES

LIST OF TABLES

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud.

Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

Cold Condition - Reactor coolant temperature equal to or less than 212°F.

Cold Shutdown - The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212°F, and the reactor vessel is vented to atmosphere.

Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958)

Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

Fraction of Limiting Power Density (FLPD) - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a

1.0 DEFINITIONS (Cont'd)

the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

Logic - A logic is an arrangement of relays, contacts and other components that produces a decision output.

(a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

Maximum Fraction of Limiting Power Density (MFLPD) - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core.

Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided: Refuel Mode, Run Mode, Shutdown Mode, Startup/Hot Standby Mode.

Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

1.0 DEFINITIONS (Cont'd)

operable or are tripped, then they shall be performed prior to returning the system to an operable status.

Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation or protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A (Cont'd)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)\ \left(\frac{FRP}{MFLPD}\right)$$

where,

FRP = fraction of rated thermal power (3293 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for all 7X7 fuel and 13.4 KW/ft for all 8X8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

2. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

4. When the reactor mode switch is in the STARTUP or RUN position, the reactor shall not be operated in the natural circulation flow mode.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia) When the reactor pressure is ≤ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.	B. APRM Rod Block Trip Setting SRB ≤ 0.66W + 42% where: SRB= Rod block setting in percent of rated thermal power (3293 MWt) W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 x 10^6 lb/hr). In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows: $SRB \leq (0.66\ W + 42\%) \left(\frac{FRP}{MFLPD}\right)$ where: FRP = fraction of rated thermal power (3293 MWt). MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for all 7X7 fuel and 13.4 KW/ft for all 8X8 fuel The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.
C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.1 in. above the top of the normal active fuel zone.	C. Scram and isolation reactor low water level -- ≥538 in. above vessel zero (0" on level instruments)

1.1.A BASES (Cont'd)

The required inputs to the statistical model are the uncertainties listed on Table 5-1 of Reference 3, the nominal values of the core parameters listed in Table 5-2 of Reference 3, and the relative assembly power distribution shown in Figure 5-1a of Reference 3.

The bases for the uncertainties in the core parameters are given in Reference 2 and the basis for the uncertainty in the GEXL correlation is given in Reference 1. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Peach Bottom Atomic Power Station Unit 3 during any fuel cycle would not be as severe as the distribution used in the analysis.

B. Core Thermal Power Limit (Reactor Pressure < 800 psia on Core Flow < 10% of Rated)

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is established by other means. This is done by establishing a limiting condition of core thermal power operation with the following basis.

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Therefore a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage.

1.1.C BASES (Cont'd.)

However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit, provided scram signals are operable, is supported by the extensive plant safety analysis.

The computer provided with Peach Bottom Unit 3 has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied upon to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 17.7 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

E. References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, January 1977 (NEDO-10958-A).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June 1974 (NEDO-20340)

3. "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDE-24011-P-A.

2.1 BASES (Cont'd.)

For analyses of the thermal consequences of the transients a MCPR equal to or greater than the operating limit MCPR given in Specification 3.5.K is conservatively assumed to exist prior to initiation of the limiting transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady state operation without forced recirculation will not be permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculating pumps.

In summary:

i. The abnormal operational transients were analyzed to a power level of 3440 MWt (104.5% rated power) to determine operating limit MCPR's.

ii. The licensed maximum power level is 3293 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Scram

The Average Power Range Monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (3293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

2.1.A <u>BASES</u> (Cont'd.)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to assure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the MFLPD is greater than the fraction of rated power (FRP).

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than 1.07 when the transient is initiated from MCPR greater than the operating limit given in Specification 3.5.K.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the Safety Limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the Rod Worth Minimizer and Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when the reactor pressure is greater than 850 psig.

2.1.A <u>BASES (Cont'd.)</u>

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5-decades are covered by the IRM by means of a range switch and the 5-decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to assure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scramed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in-sequence and provides backup protection for the APRM.

B. <u>APRM Rod Block Trip Setting</u>

The APRM system provides a control rod block to avoid conditions which would result in an APRM scram trip if allowed to proceed. The APRM rod block trip setting, like the APRM scram trip setting, is automatically varied with recirculation loop flow rate. The flow variable APRM rod block trip setting provides margin to the APRM scram trip setting over the entire recirculation flow range. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The pressure relief system for each unit at the Peach Bottom Atomic Power Station has been sized to meet two design bases. First, the total capacity of the safety/relief valves and safety valves has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4.1 of subsection 4.4 of the FSAR which states that the nuclear system safety/relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements are presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report submitted in Appendix K.

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit 3. The analysis of the worst overpressure transient, (3-second closure of all main steamline isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1301 psig if a neutron flux scram is assumed. This results in a 74 psig margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient (Load Rejection with bypass valve failure to open) assuming a turbine trip scram is presented in NEDO-24204A for Peach Bottom Unit 3. This analysis shows that the 11 safety/relief valves limit pressure at the safety valves to 25 psi below the setting of the safety valves. Therefore, the safety valves will not open.

The safety/relief valve settings satisfy the Code requirements that the lowest valve set point be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients.

The results of postulated transients where inherent safety/relief valve actuation is required are given in Section 14.0 of the Final Safety Analysis Report.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 75 psig.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.1 REACTOR PROTECTION SYSTEM	3.1 REACTOR PROTECTION SYSTEM
Applicability:	Applicability:
Applies to the instrumentation and associated devices which initiate a reactor scram.	Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.
Objective:	Objective:
To assure the operability of the reactor protection system.	To specify the type and frequency of surveillance to be applied to the protection instrumentation.
Specification:	Specification:
The setpoint, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The designed system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 100 milli-seconds	A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2 respectively. B. Daily during reactor power operation, the maximum fraction of limiting power density shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specification 2.1.A.1 and 2.1.B shall be calculated if maximum fraction of limiting power density exceeds the fraction of rated power.

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: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
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) FRP/MFLPD (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

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NOTES FOR TABLE 3.1.1 (Cont'd)

10. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

11. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 14 LPRM inputs of the normal complement.

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), where:

    FRP = fraction of rated thermal power (3293MWt).

    MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for all 7x7 fuel and 13.4 KW/ft for all 8x8 fuel.

    The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

    W = Loop Recirculation flow in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

    Trip level setting is in percent of rated power (3293 MWt).

13. See Section 2.1.A.1.

4.1 BASES (Cont'd)

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4% month; i.e., in the period of a month a maximum drift of 0.4% could occur, thus providing for adequate margin.

For the APRM systems, drift of electronic aparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivty dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.2 and 4.1.3 indicates that two instrument channels have not been included in the latter tables. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches, and, hence, calibration during operation is not applicable.

B. The MFLPD is checked once per day to determine if the APRM scram requires adjustment. Only a small number of control rods are moved daily and thus the MFLPD is not expected to change significantly. Therefore, a daily check of the MFLPD is adequate.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating twice a week using heat balance data and by calibrating individual LPRM's every 6 weeks, using TIP traverse data.

TABLE 3.2.C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. Of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq(0.66W+42)\text{x}\frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Startup Mode)	$\leq$12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq(0.66W+41)\text{x}\frac{FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (3)	$\geq$2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$108 indicated on scale	8 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP) where:

   FRP = fraction of rated thermal power (3293 MWt)

   MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for all 7x7 fuel and 13.4 KW/ft for all 8x8 fuel.

   The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

   W = Loop Recirculation flow in percent of design.
   W is 100 for core flow of 102.5 million lb/hr or greater

   Trip level setting is in percent of rated power (3293 MWt).

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is $\leq$ 30%.

8. This function is bypassed when the mode switch is placed in Run.

3.3 and 4.3 BASES (Cont'd.)

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the rod sequence control system and the rod worth minimizer (RWM).

2. The control rod housing support restricts the outward movement of a control rod to less then 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) and sequence mode of the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. The group notch mode of the RSCS restricts movement of rods assigned to each notch group to notch withdrawal and insertion. All patterns associated with these restrictions have the characteristic that, assuming the worst single deviation from the restrictions, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in the peak enthalpy of any pellet exceeding 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Sections 3.6.6, 14.6.2 and 7.16.3.3 of the FSAR, NEDO-10527 and supplements thereto, and NEDE-24011-P-A.

3.3 and 4.3 BASES (Cont'd)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. Analysis of the limiting power transients shows that the negative reactivity rates resulting from the scram (Ref. NEDO-24204A) with the average response of all drives as given in the above Specification, provide the required protection, and the MCPR remains greater than 1.07.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Peach Bottom.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds one control rod of a (5x5) twenty-five control array.

In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C. In addition the control rod drop accident has been analyzed in NEDO-10527 and its supplements 1 & 2 for the scram times given in Specification 3.3.C.

Surveillance requirement 4.3.C was originally written and used as a diagnostic surveillance technique during pre-operational and startup testing of Dresden 2 & 3 for the early discovery and identification of significant changes in drive scram performance following major changes in plant operation. The reason for the application of this surveillance was the unpredicatable and degraded scram performance of drives at Dresden 2. The cause of the slower scram performances has been conclusively

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. Normal System Availability

1. During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B below. This system need not be operable when the reactor is in the Cold Condition and all control rods are fully inserted and Specification 3.3.A is met.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the surveillance requirements of the Standby Liquid Control System

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal System Availabilitt

The operability of the Standby Liquid Control System is verified by the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested by recirculating demineralized water to the test tank.

2. At least once during each operating cycle:

a. Check that the setting of the system relief valves is $1400<P<1680$ psig.

b. Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 43 gpm against a system head of 1225 psig shall be verified. After pumping boron solution the system will be flushed and demineralized water.

3.4 BASES

STANDBY LIQUID CONTROL SYSTEM

A. The conditions under which the Standby Liquid Control System must provide shutdown capability are identified via the Plant Nuclear Safety Operational Analysis (Appendix G). If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control system is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the liquid control system is required.

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 660 ppm of boron in the reactor core in less than 125 minutes. The 660 ppm concentration in the reactor core will bring the reactor from full power to at least a 3.0%Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poision peak.

The minimum limitation on the relief valve setting is intended to prevent the recycling of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long term average availability of the system is not reduced is obtained for a one out of two system by an allowable equipment out of service time of one third of the normal surveillance frequency. This method determines an equipment out of service time of ten days. Additional conservatism is introduced by reducing the allowable out of service time to seven days, and by increased testing of the operable redundant component.

3.4 BASES (Cont'd.)

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10°F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

The volume versus concentration requirement of the solution is such that, should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature versus concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 3080 gallons of solution having a 19.3 percent sodium pentaborate concentration, or the equivalent as shown in Figure 3.4.1, is required to meet this shutdown requirement. The minimum required pumping rate of 43 gpm is based on the injection of the maximum volume permitted in Figure 3.4.1 in less than 125 minutes.

4.4 BASES

STANDBY LIQUID CONTROL SYSTEM

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. The only practical time to fully test the liquid control system is during a refueling outage. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The bases for the surveillance requirements are given in subsection 3.8.6 of the Final Safety Analysis Report, and the details of the various tests are discussed in subsection 3.8.5. The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

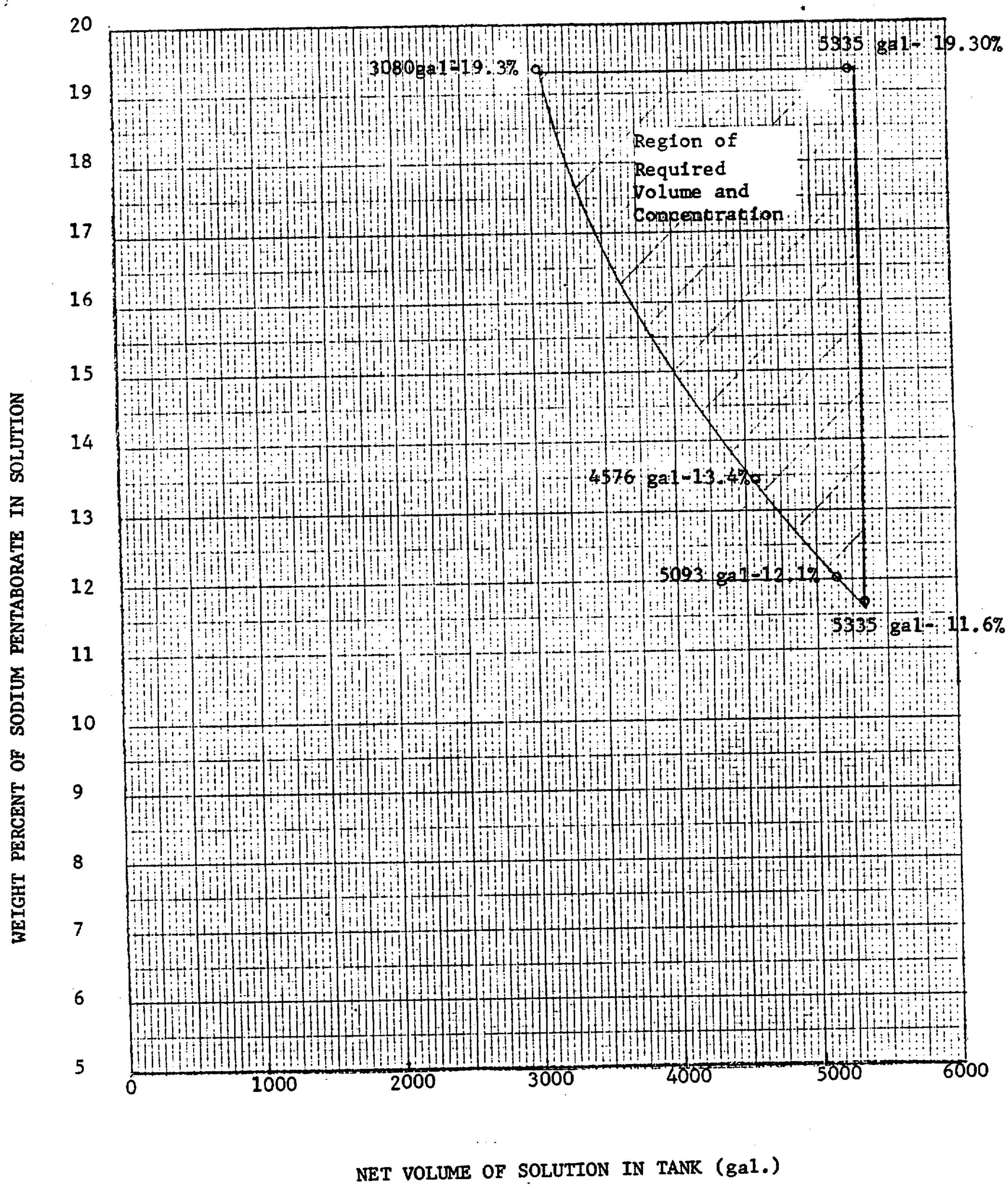

Figure 3.4.1

Amendment No. 62

LIMITING CONDITIONS FOR OPERATION

3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.5.1.A, B, C, D, F, G & H as applicable. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.J Local LHGR

During 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:

$$LHGR \leq LHGRd\ [1-(\Delta P/P)max\ (L/LT)]$$

LHGRd = Design LHGR
= 18.5 kW/ft for 7x7 fuel
13.4 kW/ft for all 8x8 fuel

$(\Delta P/P)$max = Maximum power spiking penalty
= 0.026 for 7x7 fuel
= 0.000 for 8x8 fuel

LT = Total core length
= 12.167 ft for 7x7 & 8x8 fuel
= 12.5 ft for 8x8R, 8x8 PTA and P 8x8R fuel

L = Axial position above bottom of core

SURVEILLANCE REQUIREMENTS

4.5.I Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at ≥25% rated thermal power.

4.5.J Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.

Table 3.5-2

OPERATING LIMIT MCPR VALUES AS DETERMINED FROM INDICATED TRANSIENTS FOR VARIOUS CORE EXPOSURES

Fuel Type	MCPR Operating Limit For Incremental Cycle 4 Core Average Exposure	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
7x7	1.23 (LH)	1.23 (LR)
8x8	1.24 (LH)	1.30 (LR)
PTA &P 8X8R	1.27 (RWE)	1.32 (LR)
8x8R	1.27 (RWE)	1.30 (LR)

RWE - Rod Withdrawal Error
LR - Load Rejection with failure of bypass valves to open
LH - Loss of 100°F Feedwater Heating

3.5 BASES (cont'd.)

C. HPCI

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (Appendix G) and a detailed functional analysis of the HPCI System (Section 6.0).

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. The HPCIS permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1100 and 150 psig. Two sources of water are available. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

The analysis in the FSAR, Appendix G, shows that the ADS provides a single failure proof path for depressurization for postuated transients and accidents. The RCIC serves as an alternate to the HPCI only for decay heat removal when feed water is lost. Considering the HPCI and the ADS plus RCIC as redundant paths, reference (1) methods would give an estimated allowable repair time of 15 days based on the one month testing frequency. However, a maximum allowable repair time of 7 days is selected for conservatism.

3.5 BASES (Cont'd).

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be operable when starting up from a Cold Condition. It is realized that the HPCI is not designed to operate until reactor pressure exceeds 150 psig and is automatically isolated before the reactor pressure decreases below 100 psig. It is the intent of this specification to assure that when the reactor is being started up from a Cold Condition, the HPCI is not known to be inoperable.

D. RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the main condenser is unavailable. The nuclear safety analysis, FSAR Appendix G, shows that RCIC also serves for decay heat removal when feed water is lost. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgements on the reliability of the HPCI system, an allowable repair time of 1 month is specified. Immediate and weekly demonstrations of HPCI operability during RCIC outage is considered adeuquate based on judgement and practicality.

E. Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Operational Analysis (Appendix G) and a detailed functional analysis of the ADS (Section 6).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considred only with respect to its depressurizing effect in conjunction with LPCI or Core Spray and is based on 4 valves. There are five valves provided.

3.5 BASES (Cont'd.)

H. Engineering Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Ventilation associated with the High Pressure Service Water Pumps is also associated with the Emergency Service Water pumps, and is specified in Specification 3.9.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR Part 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figure 3.5.1.A, B, C, D, F, G, and H.

The calculational procedure used to establish the APLHGR shown on Figures 3.5.1.A, B, C, D, F, G, and H is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 4. Input and model changes in the Peach Bottom loss-of-coolant analysis which are different from the previous analyses performed with Reference 4 are described in detail in Reference 8. These changes to the analysis include: (1) consideration of the counter current flow limiting (CCFL) effect, (2) corrected code inputs, and (3) the effect of drilling alternate flow paths in the bundle lower tie plate.

3.5.I <u>BASES</u> (Cont'd.)

A list of the significant plant parameters to the loss-of-coolant accident analysis is presented in Table 3.5-1.

J. <u>Local LHGR</u>

This specification assures that the linear heat generation rate in any 7X7 fuel rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. The densification power spiking penalty for the 8X8 fuel types is applied to the calculated LHGR's for the fuel loading error accident and the rod withdrawal error event in reference 6.

K. <u>Minimum Critical Power Ratio (MCPR)</u>

<u>Operating Limit MCPR</u>

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.5.K are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and analyses of the abnormal operational transients presented in References 6 and 7. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

3.5.K <u>BASES</u> (Cont'd.)

A brief summary of the analytical method used to determine the nuclear characteristics is given in Section 3 of Reference 7.

Analysis of the abnormal operational transients is presented in Section 5.2 of Reference 7. Input data and operating conditions used in this analysis are shown in Table 5-8 of Reference 7 and Section 7 of Reference 6.

L. <u>Average Planar LHGR (APLHGR), Local LHGR, and Minimum Critical Power Ratio (MCPR)</u>

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective action to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-core Probe-TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alterations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution for up to 43 incore locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication etc., corrective action is initiated within one hour of an indicated value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.L BASES (Cont'd.)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominately occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss of Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate.

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7, and 8 NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

5. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

6. Supplemental Reload Licensing Submittal For Peach Bottom Atomic Power Station Unit 3 Reload No. 3, NEDO-24204A, July, 1979.

7. General Electric Boiling Water Reactor Generic Reload Fuel Application. NEDO-24011-P-A.

8. Loss-of-Coolant Accident Analysis For Peach Bottom Atomic Power Station Unit 3, NEDO-24082, December 1977.

TABLE 3.5-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power	3440 MWt which corresponds to 105% of rated steam flow
Vessel Steam Output	14.05×10^6 lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area For Large Breaks -	
Discharge	1.9 ft^2 (DBA)
Suction	4.1 ft^2
Assumed Number of Drilled Bundles	432

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (KW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio
7x7, Type 2	7 x 7	18.5	1.5	1.2
7x7, Type 3	7 x 7	18.5	1.5	1.2
8x8, Type H	8 x 8	13.4	1.4	1.2
8x8, Type L	8 x 8	13.4	1.4	1.2
8x8 PTA	8 x 8	13.4	1.4	1.2
8x8R	8 x 8	13.4	1.4	1.2
P 8x8R Type H	8 x 8	13.4	1.4	1.2

A more detailed list of input to each model and its source is presented in Section II of Reference 5.

PEACH BOTTOM UNIT 3

P8X8R FUEL

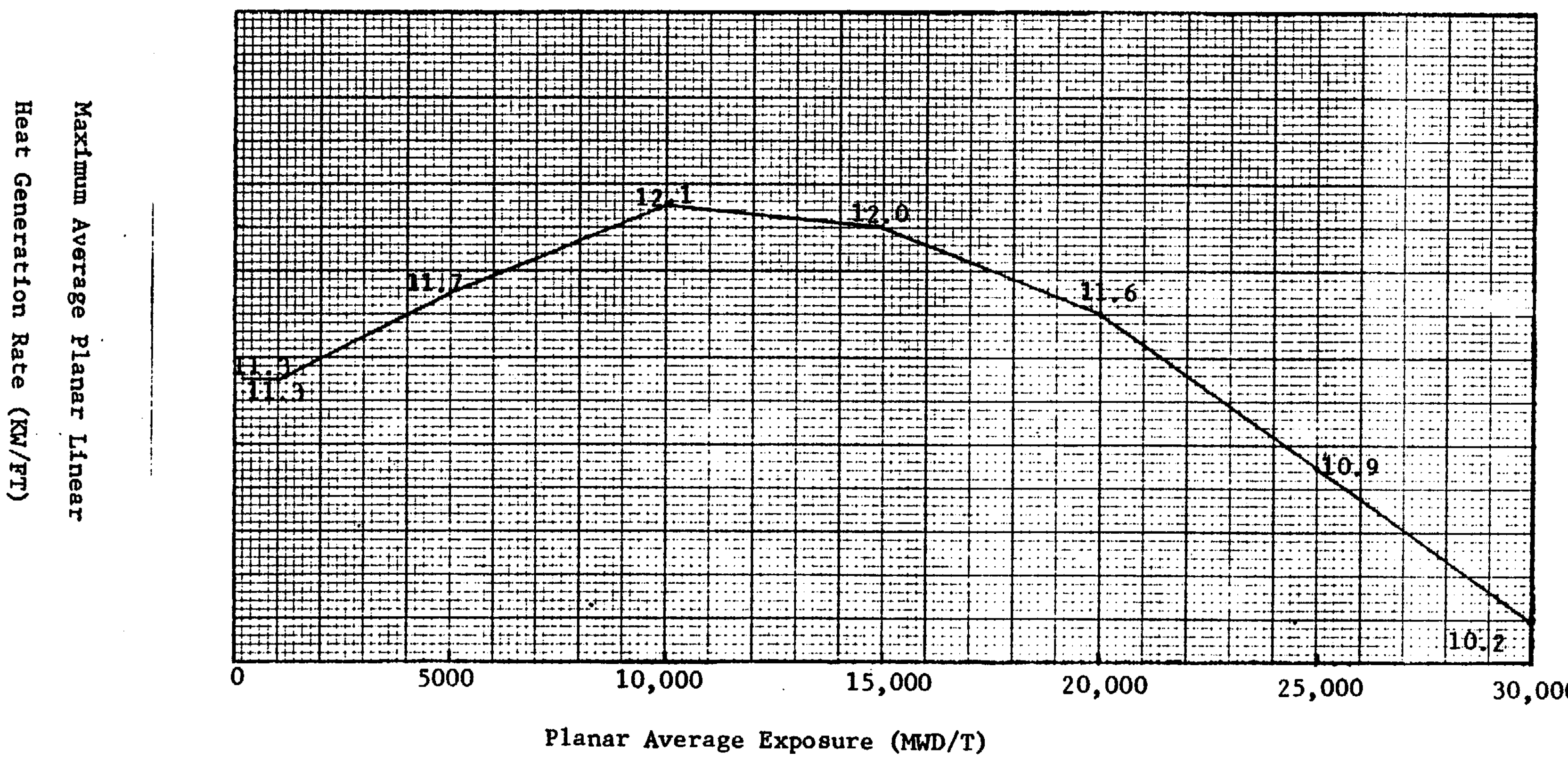

Figure 3.5.1.H Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

3.6.A Thermal and Pressurization Limitations (Cont'd)

Figures 3.6.1, 3.6.2, and 3.6.3 will be updated to account for radiation damage prior to 9 effective full power years of operation.

3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

4.6.A. Thermal and Pressurization Limitations (Cont'd)

Selected neutron flux specimens shall be removed* and tested to experimentally verify or adjust the calculated values of integrated neutron flux that are used to determine the RT_{NDT} for Figure 3.6.4,

3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shall temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

*Specimen 1 7-9 EFPY
2 15-18 EFPY
3 Standby

6.5.2 Operation and Safety Review Committee

Function

6.5.2.1 The Operation and Safety Review Committee shall function to provide independent review and audit of designated activities in the area of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

(the members of the OSR Committee will be competent in the area of quality assurance practice and cognizant of the Quality Assurance requirements of 10 CFR 50, Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance Organization available to them.)

Organization

6.5.2.2 The Chairman, Members and Alternate Members of the ONSR Committee shall be appointed in writing by the Vice President, Electric Production, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in one or more areas given in 6.5.2.1.

Alternates

6.5.2.3 Each permanent member shall have a designated alternate to serve in his absence, and a current list of these alternates shall be maintained in Committee records. Each alternate member will serve on a continuing basis.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the OSR Committee Chairman to provide expert advise to the OSR Committee.

Meeting Frequency

6.5.2.5 The OSR Committee shall meet at least once per six months.

Quorum

6.5.2.6 A quorum of the OSR Committee shall consist of the Chairman or Vice Chairman or their designated alternates and four members or their alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

6.5.2.7 The OSR Committee shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

3.6.A & 4.6.A. Bases (Cont'd)

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figure 3.6.1, 3.6.2, or 3.6.3 in conjunction with Figure 3.6.4. Note: Figure 3.6.3 includes an additional 40°F margin required by 10 CFR 50 Appendix G.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be removed and tested to experimentally verify the values used for Figure 3.6.4.

As described in paragraph 4.2.5 of the Safety Analysis report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The plant safety analyses (Ref: NEDO-24204A) states that all MSIV valve closure - Flux scram is the event which satisfies the ASME Boiler and Pressure Code requirements for protection from the consequences of pressure in excess of the vessel design pressure. The reactor vessel pressure code limit of 1375 psig, given in Subsection 4.2 of the FSAR, is well above the peak pressure produced by the above overpressure event.

3.6.D & 4.6.D BASES

Safety and Relief Valves

The safety/relief and safety valves are required to be operable above the pressure (122 psig) at which the core spray system is not designed to deliver full flow. The pressure relief system for each unit at the Peach Bottom APS has been sized to meet two design bases. First, the total capacity of the safety/relief and the safety valves has been established to meet the overpressure protection criteria of the ASME code. Second, the distribution of this required capacity between safety/relief valves and safety valves has been set to meet design basis 4.4.4.1 of subsection 4.4 of the FSAR which states that the nuclear system safety/relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report presented in Appendix K of the FSAR.

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit 3 with a total capacity of 79.51% of rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1301 psig if a neutron flux scram is assumed. This results in a 74 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the power generation design basis, the total pressure relief system capacity of 79.51% has been divided into 65.96% safety/relief (11 valves) and 13.55% safety (2 valves). The analysis of the plant isolation transient (Load Rejection with bypass valve failure to open) assuming a turbine trip scram is presented in NEDO-24204A. This analysis shows that the 11 safety/relief valves limit pressure at the safety valves to 25 psi below the setting of the safety valves. Therefore, the safety valves will not open.

Experience in safety/relief and safety valve operation shows that a testing of 50 per cent of the valves per year is adequate to detect failure or deteriorations. The safety/relief and safety valves are benchtested every second

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. With one fire pump or logic inoperable, restore the equipment to an operable status within 7 days, or in lieu of any other report required by Specification 6.9.2, submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the equipment to an operable status. Reactor startup and/or continued reactor operation is permissible.	2. With one fire pump inoperable, the remaining fire pump shall be demonstrated to be operable immediately and at least every 72 hours thereafter until the inoperable pump is restored to an operable status.
3. With two fire pumps inoperable, a. establish a back-up water supply within 24 hours, b. notify the Commission pursuant to Specification 6.9.2.a within 24 hours, by telephone and in writing no later than the first working day following the event. Submit a report within 14 days outlining the actions taken and the plans and schedule for restoring the equipment to an operable status and, c. restore the equipment to an operable status within 14 days. d. If a. above cannot be fulfilled, place the reactor in Hot Standby within the next six hours and in Cold Shutdown within the following thirty hours.	3. None
4. Except as specified in 3.14.A.6 below, the fire hose stations serving the following structures shall be operable: a. Reactor Buildings b. Radwaste Building	4. The fire hose station inspections shall be performed as follows: a. Visual inspection of hose station equipment availability - once every 31 days. b. Hose and gasket inspection - once every 18 months.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.14.A (Cont'd) c. Turbine Building d. Circulating Water Pump Structure	c. Hose station valve operability and blockage check - once every 3 years. d. Hose hydrostatic test at a pressure of 250 psig or replace with an appropriately tested hose every 3 years.
6. When a hose station serving an area which contains equipment which is required to be operable becomes inoperable, establish a continuous fire watch equipped with portable fire suppression equipment within 1 hour and provide equivalent protection to the area served by the inoperable station from an operable hose station within 6 hours.	6. None
7. Except as specified in 3.14.A.8 below, the fire suppression spray system serving a Standby Gas Treatment System charcoal filter train shall be operable when a train is required to be operable.	7. The SGTS fire suppression spray system testing shall be performed as follows: a. Simulated automatic actuation test- Once every 18 months. b. Inspection of nozzles and spray header- Once every 18 months. c. Header and nozzle air flow test- Once every 3 years.
8. If the requirements of 3.14.A.7 cannot be met, a. establish a fire watch patrol to inspect the area with inoperable fire suppression equipment at least once per shift, b. restore the system to an operable status within 14 days, or in lieu of any other report required by Specification 6.9.2 submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the system to an operable status. The SGTS may be considered operable for the purposes of Specification 3.7.B.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.14.B.3 (Cont'd) b. an operable flow path to each room and c. four heat detectors, except that one detector may be inoperable for a period not to exceed 7 days. 4. If the requirements of 3.14.B.1, 2, or 3 cannot be met, a. establish a continuous fire watch with back-up fire suppression equipment for the unprotected area (HPCI, Cable Spreading, Computer, Diesel Generator) within 1 hour b. restore the system to an operable status within 14 days, or in lieu of any other report required by Specification 6.9.2, submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the system to an operable status. Reactor startup and/or continued reactor operation is permissible.	

Table 3.14.C.1

FIRE DETECTORS

Location	Detector Type/ Designation (1)	Minimum Detectors Operable
Unit 2:		
Primary Containment (2)(3)	S1, S2, S8	3
Recirculation Pump MG Set Room	S15, S16, S17 S18, S19, S20	5
Emergency Switchgear Rooms	S11, S12, S13 S14	1 per room
Unit 3:		
Primary Containment (2)(3)	S103, S104, S106	3
Recirculation Pump MG Set Room	S111, S112, S113 S114, S116, S117	5
Emergency Switchgear Rooms	S107, S108, S109 S110	1 per room
Common:		
Control Room	S21, S22, S23, S24	4
Cable Spreading Room	S4, S7, S9, S10	4
Computer Room	S5, S6	2
Laboratory Area	H1, H2, H3, H4	4
Fan Area	S3, S105	2
Emergency Cooling Tower Switchgear Rooms	H562, H563, H564 H565	1 per room
HPSW Pump Structure	H397, H398	2
Recombiner Building	H566, H567, H568	3
Start-up Switchgear Building	H558, H559 H560, H561	2

(1) S = Smoke Detector H = Heat Detector

(2) Detector(s) inaccesible during normal operation due to inerting.

(3) May be disabled during ILRT.

4.14 BASES

A. Water Fire Protection System

The monthly test of the fire pumps is conducted to check for equipment failures and deterioration. The fire pump minimum capacity is based on a design load of 2400 gpm for the largest sprinkler plus 300 gpm for manual hose lines.

When it is determined that a fire pump is inoperable, the increased surveillance required by 4.14.A.2 provides adequate assurance that the remaining pump will be operable when required.

B. CO2 Fire Protection Systems

Weekly checking of the storage tank level and pressure is deemed adequate to provide assurance that sufficient CO2 will be available in the event of a fire occurence.

Semi-annual testing of the heat detectors in the automatic discharge systems is in accordance with NFPA-72E-1974.

Testing of the discharge initiation logic, injection valve, damper closings, and fan trippings without actual discharge of CO2 into a room demonstrates operability of the active components of the systems. System operability is demonstrated by both manual and automatic initiation for automatic discharge systems (HPCI and diesel generators). Testing of the headers and nozzles by an air flow test will detect buildups of material which may affect continued availability.

C. Fire Detection

Semi-annual testing of fire detectors is in accordance with NFPA-72E-1974.

D. Fire Barrier Penetrations

Penetration fire barrier seals are visually inspected to verify that they are functional.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

The site is located partly in Peach Bottom Township, York County, partly in Drumore Township, Lancaster County, and partly in Fulton Township, Lancaster County, in southeastern Pennsylvania on the westerly shore of Conowingo Pond at the mouth of Rock Run Creek. It is about 38 miles north-northeast of Baltimore, Maryland, and 63 miles west-southwest of Philadelphia, Pennsylvania. Figures 2.2.1 through 2.2.4 of the FSAR show the site location with respect to surrounding communities.

5.2 REACTOR

A. The core shall consist of not more than 764 fuel assemblies. 7 x 7 fuel assemblies shall contain 49 fuel rods and 8 x 8 fuel assemblies shall contain 62 or 63 fuel rods.

B. One Pressurized Test Assembly may be inserted in the Core for up to four full fuel cycles.

C. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of the theoretical density.

D. One Fast Scram Control Rod Drive may be utilized during the fourth fuel cycle operation.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2.2 of the FSAR. The applicable design codes shall be as described in Table 4.2.1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2.1 of the FSAR. The applicable design codes shall be as described in Appendix M of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

Alternates

6.5.2.3 Each permanent member shall have a designated alternate to serve in his absence, and a current list of these alternates shall be maintained in Committee records. Each alternate member will serve on a continuing basis.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the OSR Committee Chairman to provide expert advise to the OSR Committee.

Meeting Frequency

6.5.2.5 The OSR Committee shall meet at least once per six months.

Quorum

6.5.2.6 A quorum of the OSR Committee shall consist of the Chairman or Vice Chairman or their designated alternates and four members or their alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

6.5.2.7 The OSR Committee shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

6.5.2 Operation and Safety Review Committee

Function

6.5.2.1 The Operation and Safety Review Committee shall function to provide independent review and audit of designated activities in the area of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

(the members of the OSR Committee will be competent in the area of quality assurance practice and cognizant of the Quality Assurance requirements of 10 CFR 50, Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance Organization available to them.)

Organization

6.5.2.2 The Chairman, Members and Alternate Members of the ONSR Committee shall be appointed in writing by the Vice President, Electric Production, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in one or more areas given in 6.5.2.1.

6.9.2 Continued

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

6.9.3 Unique Reporting Requirements

Special reports shall be submitted to the Director of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Loss of shutdown margin, Specification 3.3.A and 4.3.A within 14 days of the event.

b. Reactor vessel inservice inspection, Specification 3.6.G and 4.6.G within 90 days of the completion of the reviews.

c. (Deleted)

d. Primary containment leak rate testing approximately three months after the completion of the periodic integrated leak rate test (Type A) required by Specification 4.7.A.2.c.2. For each periodic test, leakage test results from Type A, B and C tests shall be reported. B and C tests are local leak rate tests required by Specification 4.7.A.2.f. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test.

6.9.3 Continued

e. Release rate of Radioactive Effluents, Specification 3.8.B.7, 3.8.C.3.b, 3.8.C.5.

f. Sealed source leakage in excess of limits, Specification 3.13.3.

g. Effluent Releases

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

(1) Gaseous Releases

(a) Total radioactivity released (in curies) of noble and activation gases.

(b) Maximum noble gas release rate during any one-hour period.

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

EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 63 AND 62 TO FACILITY LICENSE NOS. DPR-44 AND DPR-56

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

I. INTRODUCTION

Philadelphia Electric Company (licensee) has requested amendments to Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Unit Nos. 2 and 3. The amendments involve (a) licensing action for Peach Bottom Unit 3 authorizing operation for Cycle 4, (b) revision of the material surveillance program for both Units 2 and 3 and (c) administrative changes for both Units 2 and 3. The identification of each change, background information and our evaluation are discussed separately.

II. PEACH BOTTOM 3 - CYCLE 4 OPERATION

1. Introduction

The Philadelphia Electric Company has proposed changes to the Technical Specifications of the Peach Bottom Atomic Power Station, Unit No. 3, (Reference 1). The proposed changes relate to the replacement of 272 fuel assemblies constituting refueling of the reactor core for 4th cycle operation at power levels up to 3293 Mwt (100% power).

Specific items for which the licensee has requested approval include: (1) modification of the APRM and RBM setpoint equations, (2) deletion of the fuel densification power spiking penalty for the 8x8 fuel, (3) deletion of the reactor vessel pressure operating limit, (4) increasing the Standby Liquid Control System capacity and (5) approval for continued use of a Fast Scram Control Rod Drive during cycle 4.

In support of these requests the licensee provided References 2 and 3 as part of the reload application. The licensee's proposed reload with 272 fuel assemblies of the type P8DRB284H represents the first reload application consisting entirely of the pressurized retrofit, P8x8R, fuel design. The remainder of the 764 fuel assemblies in the core will be of mixed fuel types irradiated during the previous cycle(s).

A large number of generic considerations related to the General Electric 7x7, PTA, 8x8, 8x8R and P8x8R fuel types and mixed cores containing these fuel types, were approved by the NRC in References 4, 5 and 6. Only the additional areas of review are discussed in this safety evaluation report.

The GE topical reports, Reference 7 and Reference 8, provide comprehensive summaries of GE BWR reload related issues, requirements and limitations. NEDE-24011-P (Reference 7) which was approved by Reference 5 also contains values for each plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other various design parameters. Additional plant and cycle dependent information are provided in the reload analysis, (Reference 2), which closely follows the outline of Appendix A of NEDE-24011-P (Reference 7). The above mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

2.0 Evaluation

2.1 Nuclear Characteristics

The reference core loading for cycle 4 is shown in Figure 1 of Reference 2. This core loading scheme results in quarter core symmetry, which is consistent with previous cycles. Section 4 of Reference 2 provided the calculated core effective multiplication and control system worth under a cold, xenon-free condition with the strongest control rod out. The minimum shutdown margin for this condition was calculated to be 1.31% $\Delta k/k$. This exceeds the minimum Technical Specification requirement of 0.38% $\Delta k/k$ for this condition.

The Standby Liquid Control System (SLCS) with its present capability (600 ppm boron) would bring the reactor to 2.2% $\Delta k/k$ subcritical. To increase the shutdown capability of the alternate shutdown system above the Technical Specification requirement of 3.0% $\Delta k/k$ subcritical, the licensee has proposed in section 5 of Reference 2 to increase the SLCS concentration to 660 ppm boron. At this increased concentration the SLCS will bring the core to at least 3.2% $\Delta k/k$ subcritical.

Based on the data presented in sections 4 and 5 of Reference 2, both the control rod system and the standby liquid control system (660 ppm boron) will have acceptable shutdown capability during cycle 4.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limits

As noted in our evaluation (Reference 5) of NEDE-24011-P, GE utilizes two transient criteria in connection with fuel performance during abnormal operational transients. These criteria, or safety limits (cf GDC 10, 10 CFR 50, Appendix A), are intended to protect against either overstraining or overheating of the cladding during transient events.

To preclude fuel rod failure from excessive strain during transients, GE has established a 1.0 percent cladding plastic strain limit. The determinable core variable used to monitor the cladding strain during reactor operations is the Linear Heat Generation Rate (LHGR) of the fuel. Maximum LHGR conditions which effect the fuel locally can occur during abnormal operational conditions such as the Rod Withdrawal Error (RWE) and the Fuel Loading Error (FLE). A more detailed discussion on this safety limit, and its applicability to Peach Bottom Unit 3, cycle 4 operations, is provided in Section 2.5.3.

To provide assurance that the fuel rods will not overheat during reactor operations, the Critical Power Ratio (CPR) is monitored. The Safety Limit Minimum Critical Power Ratio (SLMCPR) which may be allowed to result from core-wide or localized transients (or from undetected fuel loading errors) is 1.07. This limit has been imposed to assure that during transients, 99.9% of the fuel rods will avoid transition boiling, and that transition boiling will not occur during steady state operation as the result of the worst possible fuel loading error. The dependence of the operating limit MCPR on the SLMCPR is discussed in Section 2.2.2.

2.2.2. Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee to determine which event results in the largest (ΔCPR) reduction in the minimum critical power ratio. These events have been analyzed for both the exposed fuel and the new reload fuel. Addition of the largest (ΔCPR) ratio to the SLMCPR establishes the operating limit MCPRs for each fuel type.

2.2.2.1 Abnormal Operational Transient Analysis Methods

As stated in Section 1.0, this is the first Peach Bottom Unit 3 reload which consists entirely of the pressurized retrofit P8x8R fuel design. However, cycle 3 reload consisted of 252 retrofit 8x8R fuel assemblies. The only difference between the P8x8R fuel and the 8x8R fuel is the prepressurization to three atmospheres with helium in the P8x8R fuel as opposed to one atmosphere of helium in the 8x8R fuel.

Our evaluation of the transient analysis methods used for second cycle, non-equilibrium cores, of the retrofit fuel design was provided in Reference 21. In Reference 21, the staff concluded that the 8x8R GEXL correlation used by GE in the reload analysis for non-equilibrium cores has conservatisms which are equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the staff. The staff also concluded that as equilibrium cores are approached, the conservatism in the analysis methods associated with non-equilibrium cores will diminish. To assure that this conservatism is not substantially eroded, we require that this issue be resolved prior to the next reload cycle of Peach Bottom Unit 3.

The subject analysis for the retrofit fuel incorporated the local R-Factor distribution which appears in Table 5-2B of Reference 22. The R-Factors shown in the table were calculated using a local peaking factor distribution applicable to the unpressurized 8x8R fuel. The use of pressurized rods will have the effect of slightly reducing fuel temperatures during power operation which will result in a small reduction in the local Doppler feedback effect on local (pinwise) power peaking. GE states (References 15, 23) that the resulting difference between unpressurized 8x8R and pressurized P8x8R local power peaking is insignificant. Moreover, higher peaking in the P8x8R assemblies would tend to reduce the flatness of intrabundle peaking. Since decreased peaking (flatter power distribution) results in more rods in boiling transition in the GETAB statistical analysis, the use of the 8x8R R-Factor distribution for P8x8R reloads is considered conservative. Thus, the staff finds the statistical safety limit, originally derived for 8x8R reloads, to be equally acceptable for P8x8R BWR reloads.

However, the non-conservative adverse effect of high flow quality (void fraction) within the P8x8R fuel assembly channels which results from the same reduction in fuel time constant will still be present whenever P8x8R assemblies are in the core. Thus, the transient critical bundle power in the pressurized P8x8R fuel assemblies will be decreased relative to the unpressurized 8x8R and unpressurized 8x8 assemblies. GE sensitivity studies (Reference 23) indicate that for core-wide events the P8x8R assemblies will have a slightly larger transient ΔCPR (0.1) than the unpressurized 8x8 and retrofit unpressurized 8x8R fuel types. Thus, as a result of the reduced fuel time constant, the P8x8R assemblies will require a correspondingly higher operating limit MCPR than the 8x8R/8x8 assemblies whenever the limiting transient is a rapid pressurization transient.

Therefore, considering the above discussion, when operating MCPR limits for mixed (P8x8R, 8x8R and 8x8) reload cores are established based on rapid core-wide transient events, the staff finds it acceptable to either: (1) perform separate GETAB transient analyses (separate operating limits) for the pressurized and unpressurized fuel assemblies, or (2) perform a single GETAB transient analysis (a single operating limit) which conservatively incorporates the fuel rod thermal characteristics of the P8x8R fuel assembly. In the reload analysis for cycle 4 of Peach Bottom Unit 3, the licensee has selected option 1, which is acceptable. The results of the licensee's analysis are shown in Section 2.2.2.2.

2.2.2.2 Abnormal Operational Transient Analysis Results

The transients evaluated were the generator load rejection without bypass, feedwater controller failure at maximum demand, loss of 100°F feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6, 7 and Figure 2 of Reference 1 were assumed.

The calculated systems responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Listed below are the limiting MCPRs for the various fuel types at the specified cycle exposure.

Transient	Limiting Exposure Time	OLMCPR (7x7)	(8x8)	(8x8R)	(PTA & P8x8R)
Load Rejection Without Bypass	(EOC4-2 GWD/T) to (EOC4)	(1.23)	(1.30)	(1.30)	(1.32)
Loss of 100°F Feedwater Heater	(BOC4) to (EOC4-2GWD/T)	(1.23)	(1.24)	(*)	(*)
Rod Withdrawal Error **	(BOC4) to (EOC4-2 GWD/T)	(*)	(*)	(1.27)	(1.27)

* Not Limiting

** Includes the effects of densification power spiking

Addition of the most severe ΔCPR to the safety limit (1.07) gives the appropriate operating limit MCPR for each fuel type. This sum will assure that the safety limit MCPR is not violated.

We have determined that the operating limit MCPRs listed above are acceptable for cycle 4 operation at Peach Bottom Unit No. 3.

2.3 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 5. The faster fuel time constant of the reload pressurized P8x8R fuel results in more (thermal) energy being deposited in the fuel channel (within the reactor coolant pressure boundary) in a shorter period of time when compared with unpressurized fuel. However, GE sensitivity studies show that this more rapid energy transfer has a negligible effect on the peak system pressure associated with pressurization type transients. Nevertheless, current GE BWR system transient methods for mixed reload cores will account for this small effect via the dominant fuel type selection procedure discussed in Reference 7. Thus, the staff finds that the effects of fuel prepressurization are adequately accounted for in vessel overpressurization analyses. Also as specified in Reference 5, the sensitivity of peak vessel pressure to failure of one safety valve has been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the overpressure design limit (1375 psi) to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is acceptable.

2.4 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 7. The results show that the fuel type dependent channel hydrodynamic stability decay ratios and reactor core stability decay ratio at the least stable operating state (corresponding to the intersection of the natural circulation power curve and the 105% rod line) are 0.29 (8x8R/P8x8R/PTA), 0.40 (8x8), 0.01 for the (7x7) and 0.90 respectively. These predicted decay ratios are all below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

Because the pressurized fuel has a shorter thermal time constant, reactor core thermal-hydraulic stability will also be affected since it involves coupled neutronic thermal-hydraulic dynamic behavior. Sensitivity studies (Reference 14) performed with GE's licensing basis stability methods indicate that the core stability decay ratio monotonically increases with increasing fuel rod gap conductance. Thus, it is to be expected that actual core stability at the least stable operating state will decrease somewhat (increased decay ratio) during the transition from unpressurized to pressurized fuel. Additional stability studies (Reference 15) have been performed by GE more recently, utilizing their licensing basis stability code and gap conductance input from their approved GEGAP-III computer code. These studies indicate that prepressurizing 8x8R fuel to three atmospheres will cause the actual core stability decay ratio to increase by approximately 0.08 for operating BWR/2&3s and approximately 0.10 for BWR/4s. However, GE has historically utilized a constant gap conductance value of 1000 Btu/hr-ft^2-°F for licensing calculations. This conservatively bounds the gap conductance values predicted by GEGAP-III for both unpressurized and pressurized fuel designs. Moreover, GE states (Reference 15) that a significant decrease in calculated decay ratios (0.2 to 0.3) would be realized if GEGAP gap conductance values were used instead of a constant value of 1000 Btu/hr-ft-^{2}°F. Thus, although no change in decay ratios will be predicted on a licensing basis for core reloads with pressurized fuel compared to core reloads with unpressurized fuel, GE believes that adequate conservatisms will be retained in P8x8R core stability calculations.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating with a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

Prior to cycle 2 operation, the staff as an interim measure added a requirement to the Technical Specifications which restricted planned plant operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins during cycle 4. On the basis of the foregoing, the staff considers the thermal-hydraulic stability during cycle 4 to be acceptable.

2.5 Accident Analysis

Our generic evaluation of the applicability of GE's accident analysis models and methods to pressurized (P8x8R) fuel as well as our evaluation of the effects of prepressurization on previously reviewed BWR accident analysis results is contained in Reference 11. Events considered by GE included the Control Rod Drop, Fuel Loading Error, and Loss of Coolant accidents. Based on our review (Reference 6) of the information provided by GE, we agree that the methods and results for the Control Rod Drop Accident and Fuel Loading Error, contained in Reference 7, remain valid and acceptable for pressurized (P8x8R) fuel.

2.5.1 ECCS Appendix K Analysis

Input data and results for the ECCS analysis have been given in References 2, 12 and 13. The information presented fulfills the requirements for such analyses outlined in Reference 5. In connection with the Loss of Coolant Accident (LOCA) we concluded that the existing approved LOCA-ECCS models and methods remain valid for P8x8R fuel prepressurized with helium to three atmospheres. In addition, based on sensitivity studies performed by GE, we also concluded that prepressurizing the fuel to three atmospheres results in lower calculated peak cladding temperature for all BWR classes.

We have reviewed the analyses and information submitted for the reload and conclude that the Peach Bottom Unit 3 plant will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when it is operated at a Minimum Critical Power Ratio (MCPR) greater than or equal to 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss of Coolant Accident, as described in Section 2.2).

2.5.2 Control Rod Drop Accident

The Peach Bottom Unit 3 Scram Reactivity Functions at 20°C and 286°C did not satisfy the requirements for the bounding analysis described in Reference 7. Therefore, it was necessary for the licensee to perform plant and cycle specific analysis for the control rod drop accident. The results of this analysis indicate that the CRDA peak enthalpies under cold and hot conditions are 174 cal/gm and 175 cal/gm respectively. These values are well below the 280 cal/gm design limit approved in Reference 5. The staff finds these results acceptable.

2.5.3 Fuel Load Error

The licensee has considered the effects of postulated Fuel Loading Errors (FLE) in the reload analysis. The FLE analysis for the most severe misloadings were performed using GE's revised analysis methods (References 16 and 17), which have previously been reviewed and approved by the staff (Reference 18). The results show that the worst possible FLE will not cause violation of the 1.07 safety limit MCPR. The staff finds that these results, which include the 0.02 ΔCPR allowance required by NRC to allow for the axially varying water gap for a misoriented fuel bundle, are acceptable. The FLE limiting LHGR is for a UO_2 rod in the misloaded fresh fuel bundle of prepressurized fuel (Reference 20). This value was calculated to be 18.4 Kw/ft which includes the effects of densification power spiking as required by Reference 19.

Using previously accepted methods, GE calculated exposure-dependent linear heat generation rates (LHGRs), which would result in one percent cladding plastic strain for the unpressurized standard 8x8 and unpressurized 8x8R fuel types. These calculated safety limit LHGRs, which appear in Reference 7, were found to be acceptable in connection with our evaluation of the generic reload topical report. One of the principal effects of prepressurization with helium to three atmospheres is to increase the fuel-to-cladding gap conductance. Thus, for the same local linear heat generation rate, pressurized P8x8R fuel temperatures, and hence fuel thermal expansion strains, will be less than for unpressurized 8x8R fuel. Put another way, pressurized P8x8R fuel could attain a somewhat higher LHGR at which one percent cladding strain occurs. However, GE has referenced the safety limit LHGRs previously calculated for unpressurized 8x8 and unpressurized 8x8R fuel for the Peach Bottom Unit 3 reload licensing application which includes a mixture of GE fuel types in addition to the P8x8R fuel in the refueled core.

Based on comparison of the approved safety limit LHGRs related to the 1% strain criteria which appears in Reference 7, and the calculated LHGR of 18.4 Kw/ft from the FLE analysis, the limiting LHGR calculated for the misloaded pressurized P8x8R fuel is acceptable.

3.0 Fast Scram Control Rod Drive

The licensee has proposed continued operation through cycle 4 of Fast Scram Control Rod Drive (FSCRD) S/N 7464 which was initially installed for cycle 3 operation. The second cycle operation of FSCRD S/N 7464 is supported by disassembly and inspection results of the first FSCRD S/N 7067 which was installed and operated during cycle 2. Our approval for the use of the FSCRD is reported in Reference 9.

Based on our review of References 3, 9, and 10, we approve the use of FSCRD S/N 7464 in Peach Bottom Unit 3 during cycle 4. The approval does not include use of similar FSCRD's in this reactor or in other reactors without further staff review of their specific application. To facilitate future reviews in which FSCRD's are to be more extensively used for the control rod drive system, the licensee should first report the results of his findings concerning the performance of FSCRD S/N 7464 at Peach Bottom Unit No. 3 during cycle 4.

4.0 End-of-Cycle Power Coastdown

The staff has observed that several BWR licensees have stated in their reload applications that thermal power coastdown beyond EOC ARO is permissible based on reference to Section 5.2 of the Generic Reload Fuel Application (NEDE-24011-P). Although several paragraphs on coastdown appear in the topical, the subject was never explicitly addressed in our SER on the topical. However, we have been approving requests for coastdown operation via explicit plant-specific evaluations for core reloads. Our approvals have been limited to not less than 70% coastdown core power level which is the limit of our acceptance of the safety analyses generally referenced for such purposes. This 70% floor appears as a license condition for coastdown operation in our approvals.

We have discussed our concerns on EOC coastdown operations with the licensee (Reference 24). The licensee has agreed to the following conditions related to EOC coastdown operations:

1. Operation beyond the end-of-cycle (all rods out condition) thermal power is limited to seventy (70) percent minimum.

2. Increasing core power level via reduced feedwater heating, once operation in the coastdown mode has begun, is not permitted unless the licensee has performed an analysis of this operating condition that confirms that this condition is bounded by the analysis for the particular cycle of operation.

The staff finds these conditions acceptable for operation in the coastdown mode.

5.0 Physics Startup Testing

The safety analysis for the upcoming cycle is based upon a specifically designed core configuration. We have assumed that, after reloading, the actual core configuration will conform to the designed configuration. A startup test program can provide the assurance that the core conforms to the design. We require that a startup test program be performed and the minimum recommended tests are:

1. Visual inspection of the core using a photographic or videotape record.

2. A check of core power symmetry by checking for mismatches between symmetric detectors.

3. Withdrawal and insertion of each control rod to check the criticality and mobility.

4. Comparison of predicted and measured critical in-sequence rod pattern for nonvoided conditions.

The startup test program submitted by the licensee for cycle 3 remains acceptable for cycle 4.

In the future, as a result of our ongoing generic review of BWR startup tests, we anticipate requiring a description of each test sufficient to show how it provides assurance that the core conforms to the design. The description is anticipated to include both the acceptance criteria and the actions to be taken in case the acceptance criteria are not obtained.

In addition to the requirements above, we request that a brief written report of the startup tests be submitted to the NRC within 45 days of the completion of the tests.

6.0 Technical Specifications

The proposed Technical Specification changes (Reference 1) for cycle 4 include revised operating limit MCPRs for each fuel type in the core and changes to specific items identified in Section 1.0.

Based on our evaluation described in Section 2.2, the staff finds the MCPRs therein listed to be consistent with and adequately supported by the cycle 4 reload analysis.

The proposed modification of the APRM and RMB setpoint equations are consistent with GE's recommended changes appearing in Section 5.2.1.5 of Reference 7. The new factors used in the equations eliminate the need to redefine the peaking factor limit with every fuel change. Because the resulting equations are equivalent and they reduce the potential for error in redefining peaking factors from cycle to cycle, the staff finds the proposed modifications to the setpoint equations acceptable.

Deletion of the fuel densification power spiking penalty from the Technical Specification for the 8x8 fuel types has been approved by the staff in Reference 19. This approval is contingent on augmenting abnormal operational conditions which effect the fuel locally, e.g. Rod Withdrawal Error and the Fuel Loading Error by the fuel densification power spike allowance. The licensee, as shown in Sections 2.2.2.2 and 2.5.3, has met this requirement. Therefore, the staff finds the requested deletion acceptable.

The design basis overpressure transient analysis found acceptable in Section 2.3 provides sufficient margin between the reactor vessel high pressure setpoint (1055 psi) and the overpressure design limit (1375 psi) to accommodate the most severe pressurization transient. Additional conservatism is inherent in this comparison because the trend is for the pressure increase from the transient to be much less than directly proportional to the increase in initial dome pressure (Reference 5).

The staff approval for increasing the Standby Liquid Control System capacity and for continued use of the Fast Scram Control Rod Drive during cycle 4 is provided in Sections 2.1 and 3.0 respectively.

III. MATERIAL SURVEILLANCE PROGRAM

The licensee's application dated May 8, 1979 proposed an integrated material surveillance program for Units 2 and 3. The proposed program consisted of a common neutron flux specimen removal program. From review of the licensee's proposal and from available information, we conclude that the materials in these reactor vessels are different and that an integrated materials surveillance program is not appropriate. The staff recommended that compliance with 10 CFR 50, Appendix H could be achieved with the following withdrawal schedule:

Material surveillance capsules should be removed from Peach Bottom Units 2 and 3 in accordance with the following schedule:

First capsule - During a refueling outage at 7 to 9 EFPY.

Second capsule - During a refueling outage at 15 to 18 EFPY.

Third capsule - Standby

The results of tests on these specimens will be used to verify and/or adjust the changes in RT_{NDT} due to irradiation.

The acceptability of the above schedule is discussed below.

10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements" requires a material surveillance program for reactor vessels to monitor changes in the fracture toughness properties of ferritic materials in the vessel beltline region resulting from their exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens periodically withdrawn from the reactor vessel. This Appendix gives withdrawal schedules based on the amount of radiation damage predicted at the end of the service lifetime of the vessel. At the end of the service life the materials in the Peach Bottom reactor vessels are expected to have RT_{NDT} values of less than 100°F. Thus, a three capsule program is required.

The above withdrawal schedule is based on the assumption that the adjusted value of RT_{NDT} at end of life will not exceed 100°F. We feel that RT_{NDT} will not exceed 100°F because of the relatively low fluences on the vessel wall, approximately 1×10^{18} n/cm^2 at end of life. In case the amount of radiation damage exceeds this estimate, the withdrawal schedule will be revised in accordance with paragraph II.C.3 of Appendix H.

The proposed surveillance program as modified is acceptable and is in accordance with Appendix H, 10 CFR Part 50. This program was discussed with the licensee and he agreed.

IV. ADMINISTRATIVE CHANGES

Included in the licensee's application dated May 8, 1979 were a number of administrative and typographical changes to the Technical Specifications consisting of: (1) clarification of the staff's requirements for reporting of primary and secondary leak rate test results; (2) revision of certain titles of members of the Operation and Safety Review Committee to reflect recent changes to the Engineering and Research Department organization; (3) revision of the table specifying fire detectors by deleting reference to a non-existent heat detector, and (4) correction of certain typographical errors.

1. Reporting of Leak Rate Test Results

The staff's requirements for reporting of periodic leak rate tests for containments are identified in two documents: (a) paragraph V. of Appendix J to 10 CFR 50; and (b) Specifications 4.6.1.5 and 6.9.1 of the Standard Technical Specifications for General Electric Boiling Water Reactors (NUREG-0123). These requirements are limited to the periodic testing of primary containment systems. We require the submission of special reports of the test results to permit the staff to identify and evaluate any evidence of structural deterioration which may affect either the containment structural integrity or leak-tightness. The wording of the current Peach Bottom Unique Reporting Requirements (TS 6.9.3) implies that all primary and secondary containment leak rate testing requires submission of Special Reports to the Commission. A strict interpretation of this wording would include a requirement for submission of Special Reports for all containment leak testing such as the confirmation of the ability of secondary containment to maintain 1/4 inch of water negative pressure after each violation of secondary containment.

In view of the above, the licensee proposed a clarification which would in effect delete this requirement for submission of unnecessary reports. We have reviewed the licensee's submittal and agree that a revised wording of the Peach Bottom TS is prudent. The staff recommended to the licensee that all references to secondary containment leak rate test results should be deleted from the Unique Reporting Requirements. This recommendation is based on the fact that the reporting of any degradation of secondary containment is already required by Specification 6.9.2 Reportable Occurrences. This vehicle for reporting secondary containment degradation is more appropriate than that of a Special Report. The licensee agreed. Based on the above, we find the request as modified by the staff to be acceptable.

2. Operation and Safety Review Committee Titles

The licensee's proposal would revise the title of one of the members of the committee to reflect a recent reorganization within the Engineering and Research Department. Since the change does not involve functions of the Committee or qualifications of its membership we find the change to be acceptable. To eliminate future administrative changes of this type, the staff recommended that the composition of the committed be specified in terms of technical qualifications and experience instead of by title. The licensee agreed.

3. Heat Detectors

By Amendment Nos. 39 and 39 to DPR-44 and DPR-56, the Technical Specifications were revised to incorporate limiting conditions for operation and surveillance requirements for existing fire protection systems. The licensee's request would delete reference to a non-existent heat detector (H 569). We have reviewed the request as well as the "Peach Bottom Atomic Power Station Units Nos. 2 and 3 Fire Protection Program Report" and verified that the change is a correction to a previously issued Amendment and is acceptable.

4. Typographical Errors

The other changes requested by the licensee have been verified to be typographical errors and are acceptable.

V. Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

VI. Conclusion

For item II which relates to Operating License DPR-56, we have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the approval of this item will not be inimical to the common defense and security or to the health and safety of the public.

For items III and IV which relate to both Operating License Nos. DPR-44 and DPR-56, we have concluded, based on the considerations discussed above, that: (1) because these items do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the approval of these items does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 24, 1979

References

1. Application For Amendment of Facility Operating License DPR-56, dated August 2, 1979.

2. Supplemental Reload Licensing Submittal For Peach Bottom Atomic Power Station Unit No. 3, Reload No. 3, NEDO-24204 A, dated July 1979.

3. General Electric Boiling Water Reactor Reload 1 Licensing Amendment For Peach Bottom Atomic Power Station Unit No. 3, Fast Scram Control Rod Drive Second Supplement, NEDO-21363-2A, dated July 1979.

4. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

5. Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-24011-P), April 1978.

6. Letter, T. A. Ippolito (NRC) to R. Gridley (GE), dated April 16, 1979, transmitting Safety Evaluation Supplement of the GE Generic Reload Fuel Application approving use of prepressurized retrofit 8x8 fuel for BWR reloads.

7. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDO-24011-P, May 1977.

8. General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1, 1976.

9. Memorandum: D. Eisenhut (NRC) to K. Goller (NRC) Review of Peach Bottom Unit No. 3 Reload 1, dated February 23, 1977.

10. General Electric Boiling Water Reactor Reload 1 Licensing Amendment For Peach Bottom Atomic Power Station Unit No. 3 Fast Scram Control Rod Drive Supplement, NEDO-21363-2, dated November 1976.

11. NRC letter (O. Parr) to General Electric (G. Sherwood) dated November 21, 1978.

12. Loss-Of-Coolant Accident Analysis Report for James A. Fitzpatrick Nuclear Power Station Plant (Lead Plant), NEDO-21662, July 1977.

13. Loss-Of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit No. 3, NEDO-24082, December 1977.

14. "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Report NEDO-21506, dated January 1977.

15. General Electric letter (E. Fuller) to NRC (O. Parr), dated August 14, 1978.

16. General Electric letter (R. Engle) to USNRC (D. Eisenhut), "Fuel Assembly Loading Error," dated June 1, 1977.

17. General Electric letter (R. Engle) to USNRC (D. Eisenhut), dated November 30, 1977.

18. USNRC letter (D. Eisenhut) to General Electric (R. Engle), dated November 30, 1977.

19. USNRC letter (D. Eisenhut) to General Electric (R. Gridley), dated June 9, 1978.

20. Letter: M. J. Cooney (Philadelphia Electric Company) to T. A. Ippolito (NRC), dated October 2, 1979.

21. Memorandum: P. Check (NRC) to T. A. Ippolito (NRC) Review of Cooper Nuclear Station Unit 1, Reload 4, dated April 11, 1979.

22. NRC letter (Eisenhut) to General Electric (Gridley) transmitting "Safety Evaluation for the General Electric Topical Report, 'Generic Reload Fuel Application, (NEDE-24011-P)'" dated May 12, 1978.

23. General Electric letter (E. Fuller) to NRC (O. Parr) dated June 8, 1978.

24. Telephone conversation: R. Riggs, D. Verelli (NRC), to L. Rubino (Philadelphia Electric Company), dated September 10, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-277 AND 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 63 and 62 to Facility Operating License Nos. DPR-44 and DPR-56, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station, Units Nos. 2 and 3 (the facility) located in York County, Pennsylvania. The amendments are effective as of the date of issuance.

The amendment for Peach Bottom Unit 3 (DPR-56) involves:

(1) use of pre-pressurized fuel for Cycle 4 operation;

(2) modification of the APRM and RBM setpoint equations;

(3) deletion of the fuel densification power spiking penalty for 8x8 fuel;

(4) deletion of the reactor vessel pressure operating limit;

(5) continued use of the fast scram control rod drive during Cycle 4;

(6) increase of the standby liquid control system capacity;

(7) addition of a license condition which governs operation during any coastdown after end-of-cycle;

(8) revision of the withdrawal schedule for the reactor vessel material surveillance program, and

(9) administrative changes relating to reporting of primary and secondary leak rate results, members of the Operation and Safety Review Committee, the table specifying fire detectors and correction of typographical errors.

The amendment for Peach Bottom Unit 2 (DPR-44) involves the reactor vessel material surveillance program and administrative changes (items 8 and 9 above).

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of an Amendment to Facility Operating License No. DPR-56 was published in the FEDERAL NOTICE on August 16, 1979 (44 FR 48000). The proposed action so noticed included items (1) through (6) above. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action. Prior public notice of items (7) through (9) and the amendment to Operating License No. DPR-44 was not required since these items do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendment dated May 8 and August 2, 1979, as supplemented by information contained in letter dated October 2, 1979, (2) Amendment Nos. 63 and 62 to License Nos. DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 24th day of October, 1979

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors