

JUNE 13 1980

REGULATORY DOCKET FILE COPY

DISTRIBUTION:

Docket File

NRC PDR

L PDR

Docket No. 50-277

TERA

NSIC

ORB#4 Rdg

NRR Rdg

DEisenhut

RPurple

TNovak

GLainas

RTedesco

RReid

DVerrelli

RIngram

IE-5

OELD

AEOD

ACRS-16

Gray File +4

RDiggs

HDenton

CMiles

BJones-4

BScharf-10

STSG

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 the enclosed Amendment No. 70 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. The amendment revises the Technical Specifications for Cycle 5 operation and is in response to your application dated March 3, 1980, as amended by your letter dated April 28, 1980.

The revision to the Technical Specifications involves: (1) modification of the average power range monitor and rod-block monitor 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) increase in the Standby Liquid Control System capacity, (5) use of two control rods containing hafnium control pins, and (6) extension of exposure times on the Lead Test Assemblies.

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

Sincerely,

Original signed by
Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:

1. Amendment No. 70 to DPR-44
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

8006270 357

No copy distribution taken of Amendment or Notice. See account.

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD-OR:DL	OELD
SURNAME	RIngram	DVerrelli/cb	RReid	TNovak	CUTCHN
DATE	6/6/80	6/6/80	6/9/80	6/10/80	6/11/80



UNITED STATES
 NUCLEAR REGULATORY COMMISSION
 WASHINGTON, D.C. 20555
June 13, 1980

TRIBUTION:
 Docket File
 ORB#4RDG
 Ringram

Docket No. **50-277**

Docketing and Service Section
 Office of the Secretary of the Commission

SUBJECT: PEACH BOTTOM UNIT NO. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- Notice of Availability of Applicant's Environmental Report.
- Notice of Proposed Issuance of Amendment to Facility Operating License.
- Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- Notice of Availability of NRC Draft/Final Environmental Statement.
- Notice of Limited Work Authorization.
- Notice of Availability of Safety Evaluation Report.
- Notice of Issuance of Construction Permit(s).

Notice of Issuance of Facility Operating License(s) or Amendment(s).

Other: Amendment No. 70
Referenced documents have been provided PDR

Division of Licensing, ORB#4
 Office of Nuclear Reactor Regulation

Enclosure:
 As Stated

OFFICE →	ORB#4:DL					
SURNAME →	RIngram/cb					
DATE →	6/13/80					



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

June 13, 1980

Docket No. 50-277

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 the enclosed Amendment No. 70 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. The amendment revises the Technical Specifications for Cycle 5 operation and is in response to your application dated March 3, 1980, as amended by your letter dated April 28, 1980.

The revision to the Technical Specifications involves: (1) modification of the average power range monitor and rod-block monitor 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) increase in the Standby Liquid Control System capacity, (5) use of two control rods containing hafnium control pins, and (6) extension of exposure times on the Lead Test Assemblies.

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

Sincerely,

A handwritten signature in cursive script, reading "Robert W. Reid".

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:

1. Amendment No. 70 to DPR-44
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Mr. Edward G. Bauer, Jr.
Philadelphia Electric Company

cc:

Eugene J. Bradley
Philadelphia Electric Company
Assistant General Counsel
2301 Market Street
Philadelphia, Pennsylvania 19101

Troy B. Conner, Jr.
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Raymond L. Hovis, Esquire
35 South Duke Street
York, Pennsylvania 17401

Warren K. Rich, Esquire
Assistant Attorney General
Department of Natural Resources
Annapolis, Maryland 21401

Philadelphia Electric Company
ATTN: Mr. W. T. Ullrich
Peach Bottom Atomic
Power Station
Delta, Pennsylvania 17314

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

Curt Cowgill
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Peach Bottom Atomic Power Station
P. O. Box 399
Delta, Pennsylvania 17314

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

Region III Office
ATTN: EIS COORDINATOR
Curtis Building (Sixth Floor)
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

M. J. Cooney, Superintendent
Generation Division - Nuclear
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

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

cc w/enclosure(s) & incoming dtd.:
3/3/80 & 4/28/80
Mr. R. A. Heiss, Coordinator
Pennsylvania State Clearinghouse
Governor's Office of State Planning
and Development
P. O. Box 1323
Harrisburg, Pennsylvania 17120



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. 70
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 March 3, 1980, as supplemented April 28, 1980, 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.

8006270361

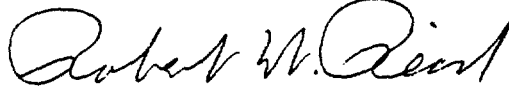
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 70, are hereby incorporated in the license. PECO 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



Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 13, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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.

<u>Remove Pages</u>	<u>Insert Pages</u>
iv	iv
1	1
3	3
4	4
7	7
10	10
11	11
15	15
18	18
19	19
20	20
21	21
31	31
33	33
35	35
37	37
40	40
54	54
73	73
74	74
91	91
92	92
108	108
111	111
115	115
119	119
120	120
121	121
122	122
133a	133a
133c	133c
140	140
140a	140a
140b	140b
140c	140c
140d	140d
140e	140e
142	Deleted
142a	Deleted

Remove Pages

142b
142e
--
--
144
152a
157
241
242

Insert Pages

142b
142e
142g (added)
142h (added)
144
152a
157
241
242

Page 121 is unchanged and is included for your convenience only.

LIST OF FIGURES
Title

<u>Figure</u>	<u>Title</u>	<u>Page</u>
1.1-1	APRM Flow Bias Scram Relationship To Normal Operating Conditions	16
4.1.1	Instrument Test Interval Determination Curves	55
4.2.2	Probability of System Unavailability Vs. Test Interval	98
3.4.1	Required Volume and Concentration of Standby Liquid Control System Solution	122
3.4.2	Required Temperature vs. Concentration for Standby Liquid Control System Solution	123
* Figures 3.5.1 A and 3.5.1 B (7x7 Fuel) deleted. (PB2 Cycle 5- all 8x8 core)		
3.5.1.C	MAPLHGR Vs. Planar Average Exposure, Unit 2, 8x8 Fuel, Type H 80 mil & 100 mil channels	142b
3.5.1.D	MAPLHGR Vs. Planar Average Exposure, Unit 2, 8x8 Fuel, Type L, 100 mil channels	142c
3.5.1.E	Kf Factor Vs. Core Flow	142d
3.5.1.F	MAPLHGR Vs. Planar Average Exposure, Unit 2, 8x8 LTA Fuel, 100 mil channels	142e
3.5.1.G	MAPLHGR Vs. Planar Average Exposure, Unit 2, 8x8R Fuel, Type 8DRB284, 100 mil channels	142f
3.5.1.H	MAPLHGR Vs. Planar Average Exposure, Unit 2, P 8X8R Fuel, Type P8DRB285, 100 mil channels	142g
3.5.1.I	MAPLHGR vs. Planar Average Exposure, Unit 2, P 8x8R Fuel, Type P8DRB284 H. 80 mil & 100 mil channel & 120 mil channels	142h
3.6.1	Minimum Temperature for Pressure Tests such as required by Section XI	164
3.6.2	Minimum Temperature for Mechanical Heatup or Cooldown following Nuclear Shutdown	164a
3.6.3	Minimum Temperature for Core Operation (Criticality)	164b
3.6.4	Transition Temperature Shift vs. Fluence	164c
6.2-1	Management Organization Chart	244
6.2-2	Organization for Conduct of Plant Operations	245

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 represent 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 produce 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)

Operating - Operating means that a system or component is performing its intended functions in its required manner.

Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.
2. At least one door in each airlock is closed and sealed.
3. All automatic containment isolation valves are operable or deactivated in the isolated position.
4. All blind flanges and manways are closed.

Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.

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 13.4 KW/ft for all 6X8 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 LIMITB. 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.

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.

LIMITING SAFETY SYSTEM SETTINGB. APRM Rod Block Trip Setting

$$SRB \leq 0.66W + 42\%$$

where:

SRB= Rod block setting in percent of rated thermal power (3293 MWt)

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

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 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. Scram and isolation-->538 in. above
reactor low water vessel zero
level (0" on level
instruments)

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 2 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 BASES (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 the fuel cladding integrity safety limit 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 change 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 BASES (Cont'd.)

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 at 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 scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. 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. APRM Rod Block Trip Setting

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.

2.1 BASES (Cont'd.)

C. Reactor Water Low Level Scram and Isolation (Except Main Steamlines)

The set point for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in PSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the fuel cladding integrity safety limit in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 31 in. below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of less than or equal to 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

E. Turbine Control Valve Scram

The turbine control valve fast closure scram anticipates the pressure, neutron flux and heat flux increase that could result from fast closure of the turbine control valves due to a load rejection exceeding the capacity of the bypass valves or a failure in the hydraulic control system which results in a loss of oil pressure. This scram is initiated from pressure switches in the hydraulic control system which sense loss of oil pressure due to the opening of the fast acting solenoid valves or a failure in the hydraulic control system piping. Two turbine first stage pressure switches for each trip system initiate automatic bypass of the turbine control valve fast closure scram when the first stage pressure is below that required to produce 30% of rated power. Control valve closure time is approximately twice as long as that for stop valve closure.

1.2 BASES

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator assures not exceeding 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel (1250 psig at 575 degrees F) and coolant system piping (suction piping: 1148 psig at 562 degrees F; discharge piping: 1326 psig at 562 degrees F). The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and ANSI B31.1.0 for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure ($110\% \times 1250 = 1375$ psig), and the ANSI Code permits pressure transients up to 20% over the design pressure ($120\% \times 1148 = 1378$ psig; $120\% \times 1326 = 1591$ psig).

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. At this time it is included in the reactor coolant system.

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 show 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 2. The analysis of the worst overpressure transient is provided in the Supplemental Reload Licensing Submittal and demonstrates margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient is provided in the Supplemental Reload Licensing submittal Safety Evaluation and demonstrates that 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 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 OPERATIONSURVEILLANCE REQUIREMENTS**3.1 REACTOR PROTECTION SYSTEM****Applicability:**

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

When there is fuel in the vessel, 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.

3.1 REACTOR PROTECTION SYSTEM**Applicability:**

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

- 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			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (7)	Startup	Run		
1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
2	APRM High Flux	(.66W+54) 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

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 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 apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that two instrument channels have not been included in the latter table. 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)	$\leq (0.66W+L_2) \times \frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Startup Mode)	$\leq 12\%$	6 Inst. Channels	(1)
2	APRM Downscale	≥ 2.5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq (0.66W+L_1) \times \frac{FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	≥ 2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (3)	≥ 2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(1)
3	IRM Upscale	≤ 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)

Amendment No. 28, 34, 36, 42, 48, 70

-73-

UNIT 2

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
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 ≥ 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.

3.2 BASES (Cont'd)

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 850 psig. The Reactor Pressure Vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the RUN Mode is less severe than the loss of feedwater analyzed in section 14.5 of the FSAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic. Temperature is monitored at four (4) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" DC control bus and two (2) by "B" DC control bus. Each pair of sensors, e.g., "A" or "B" at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves. The trip settings of $\leq 300\%$ of design flow for high flow and 200°F for high temperature are such that core uncover is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of $\leq 300\%$ for high flow and 200°F for temperature are based on the same criteria as the HPCI.

The Reactor Water Cleanup System high flow and temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that core uncover is prevented and fission product release is within limits.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the fuel cladding integrity safety limit. The trip logic for this function is 1 out of n: e.g., any trip on one of 6 APRM's, 8 IRM's, or 4 SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

3.2 BASES (Cont'd)

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection: i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit.

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in the local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in the control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point when one has an upscale.

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 than 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 the fuel cladding integrity safety limit. Analysis of the limiting power transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above Specification, provide the required protection.

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 rod array.

In the analytical treatment of the transients, which are assumed to scram on high neutron flux, 340 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. The 340 milliseconds used in the analyses consist of 140 milliseconds for sensor and circuit delay and 200 milliseconds to start of control rod motion. 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 unpredictable and degraded scram performance of drives at Dresden 2. The cause of the slower scram performances has been conclusively

LIMITING CONDITIONS FOR OPERATIONSURVEILLANCE REQUIREMENTS**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.

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 Availability**

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 with 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 poison 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 is based on the injection of the maximum net storage volume within 125 minutes.

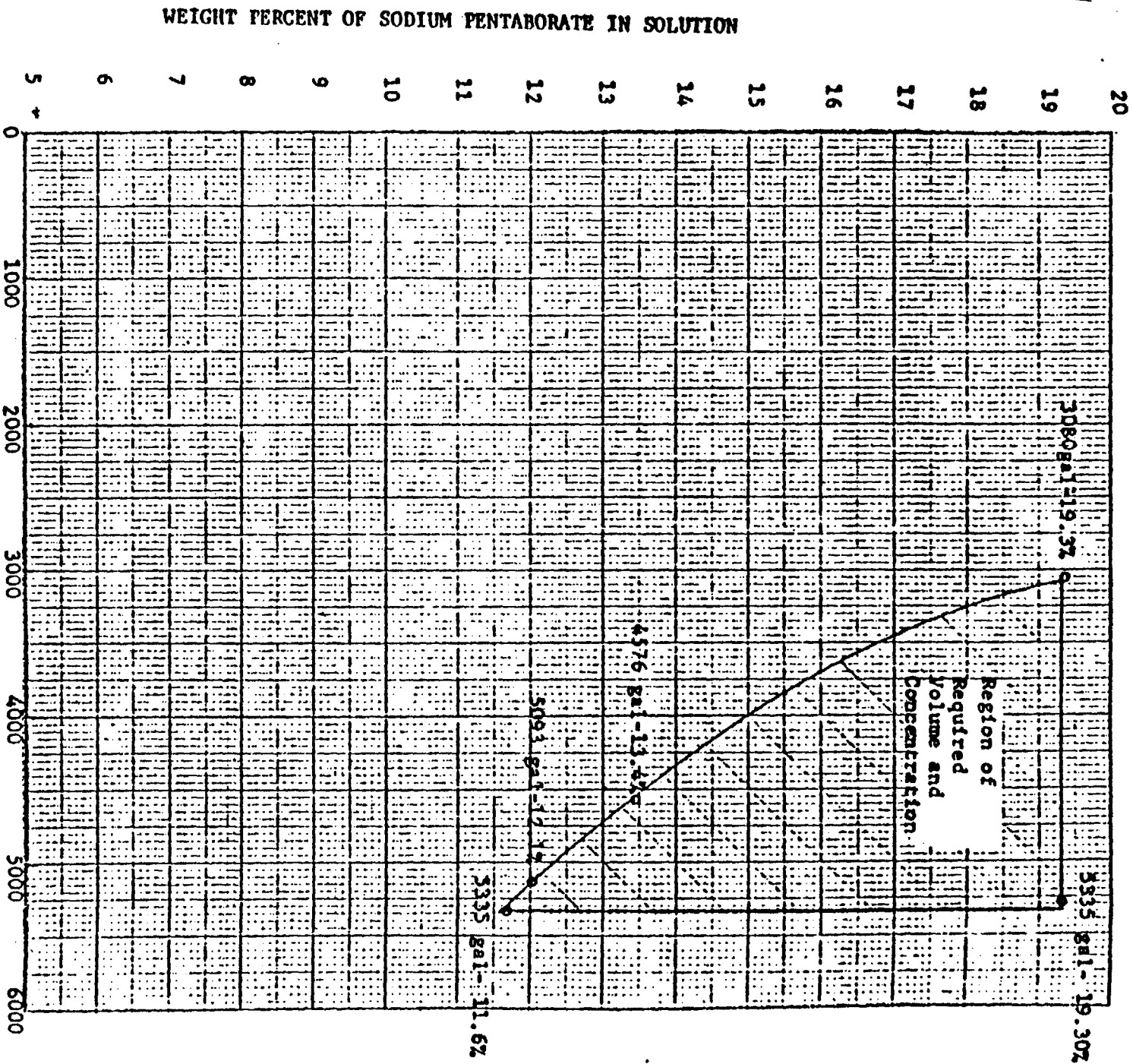
PBAPS

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.



Amendment No. 7 0

Figure 3.4.1

LIMITING CONDITIONS FOR OPERATIONSURVEILLANCE REQUIREMENTS**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 Figures 3.5.1.C,D, F, G, H & I 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 design LHGR.

$LHGR \leq LHGRd$

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

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 $\geq 25\%$ rated thermal power.

4.5.J Local LHGR

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

Table 3.5-2

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

<u>Fuel Type</u>	<u>MCPR Operating Limit For Incremental Cycle 5 Core Average Exposure</u>	
	<u>BOC to 1000 MWD/t Before EOC</u>	<u>1000 MWD/t before EOC To EOC</u>
8x8	1.28	1.31
8x8R & LTA	1.28	1.31
P8x8R	1.30	1.33

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 Figures 3.5.1.C, D, F, G, H, and I.

The calculational procedure used to establish the APLHGR shown on Figures 3.5.1.C, D, F, G, H, and I 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 BASES (Cont'd.)

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

J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR shall be checked daily during reactor operation at 225% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

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 the fuel cladding integrity safety limit, and analyses of the abnormal operational transients presented in the Supplemental Reload Licensing Submittal and Reference 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 are as described in reference 7.

3.5.K BASES (Cont'd.)

The limiting transients which determine the required steady state MCPR limits are given in Table 3.5-2. These transients yield the largest Δ CPR for each class of fuel. When added to the safety limit MCPR of the fuel cladding integrity safety limit, the required minimum operating limit MCPR's of specification 3.5.K are obtained.

Two codes are used to analyze the rod withdrawal error transient. The first code simulates the three dimensional BWR core nuclear and thermal-hydraulic characteristics. Using this code a limiting control rod pattern is determined; the following assumptions are included in this determination:

- (1) The core is operating at full power in the xenon-free condition.
- (2) The highest worth control rod is assumed to be fully inserted.
- (3) The analysis is performed for the most reactive point in the cycle.
- (4) The control rods are assumed to be the worst possible pattern without exceeding thermal limits.
- (5) A bundle in the vicinity of the highest worth control rod is assumed to be operating at the maximum allowable linear heat generation rate.
- (6) A bundle in the vicinity of the highest worth control rod is assumed to be operating at the minimum allowable critical power ratio.

The three-dimensional BWR code then simulates the core response to the control rod withdrawal error. The second code calculates the Rod Block Monitor response to the rod withdrawal error. This code simulates the Rod Block Monitor under selected failure conditions (LPRM) for the core response (calculated by the 3-dimensional BWR simulation code) for the control rod withdrawal.

The analysis of the rod withdrawal error for Peach Bottom Unit 2 considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor which is operating with the limiting control rod pattern as discussed above.

3.5.K BASES (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 in the Supplemental Reload Licensing Submittal.

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

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 APLGR, 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 APLGR, 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 APLGR, 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. - DELETED
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 2, NEDO-24081, December 1977.

TABLE 3.5-1.

SIGNIFICANT INPUT PARAMETERS TO THE
LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

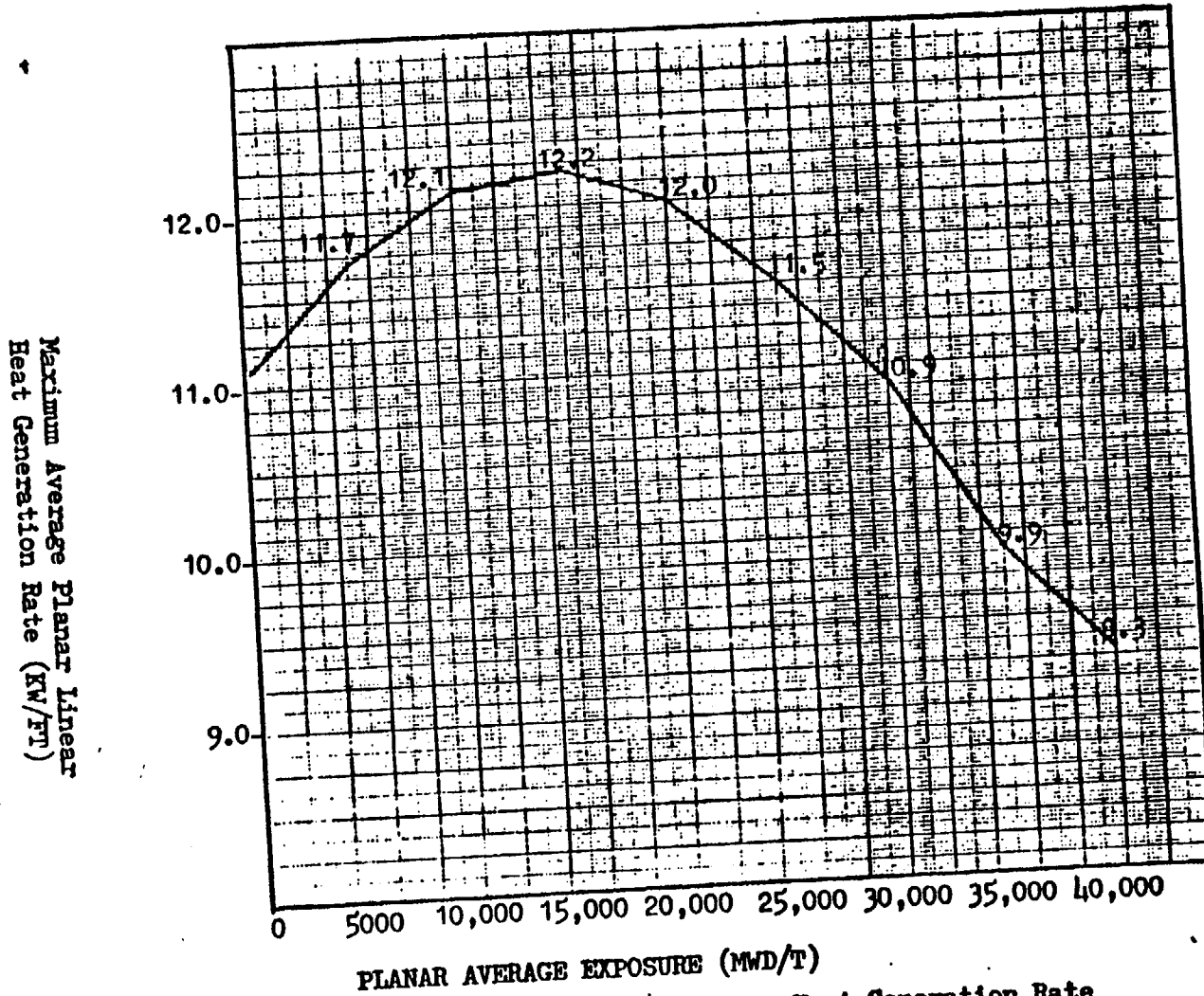
Core Thermal Power	3440 MWt which corresponds to 105% of rated steam flow
Vessel Steam Output	14.05 x 10 ⁶ lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area For Large Breaks -	
Discharge	1.9 ft ² (DBA)
Suction	4.1 ft ²
Assumed Number of Drilled Bundles	360

FUEL PARAMETERS:

<u>Fuel Type</u>	<u>Fuel Bundle Geometry</u>	<u>Peak Technical Specification Linear Heat Generation Rate (KW/ft)</u>	<u>Design Axial Peaking Factor</u>	<u>Initial Minimum Critical Power Ratio</u>
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
8x8R/LTA	8 x 8	13.4	1.4	1.2
P 8x8R Type P8DRB284H	8 x 8	13.4	1.4	1.2
P 8x8R Type P8DRB285	8 x 8	13.4	1.4	1.2

Amendment No. 16, 40, 70

PEACH BOTTOM UNIT 2
8X8 FUEL, TYPE 8D274H - 80 MIL and 100 MIL CHANNELS



- 142b -

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

PBAPS

Unit 2

PEACH BOTTOM UNIT 2

FUEL TYPE: 8X8 LTA 260 (Applicable to 100 mil channels)

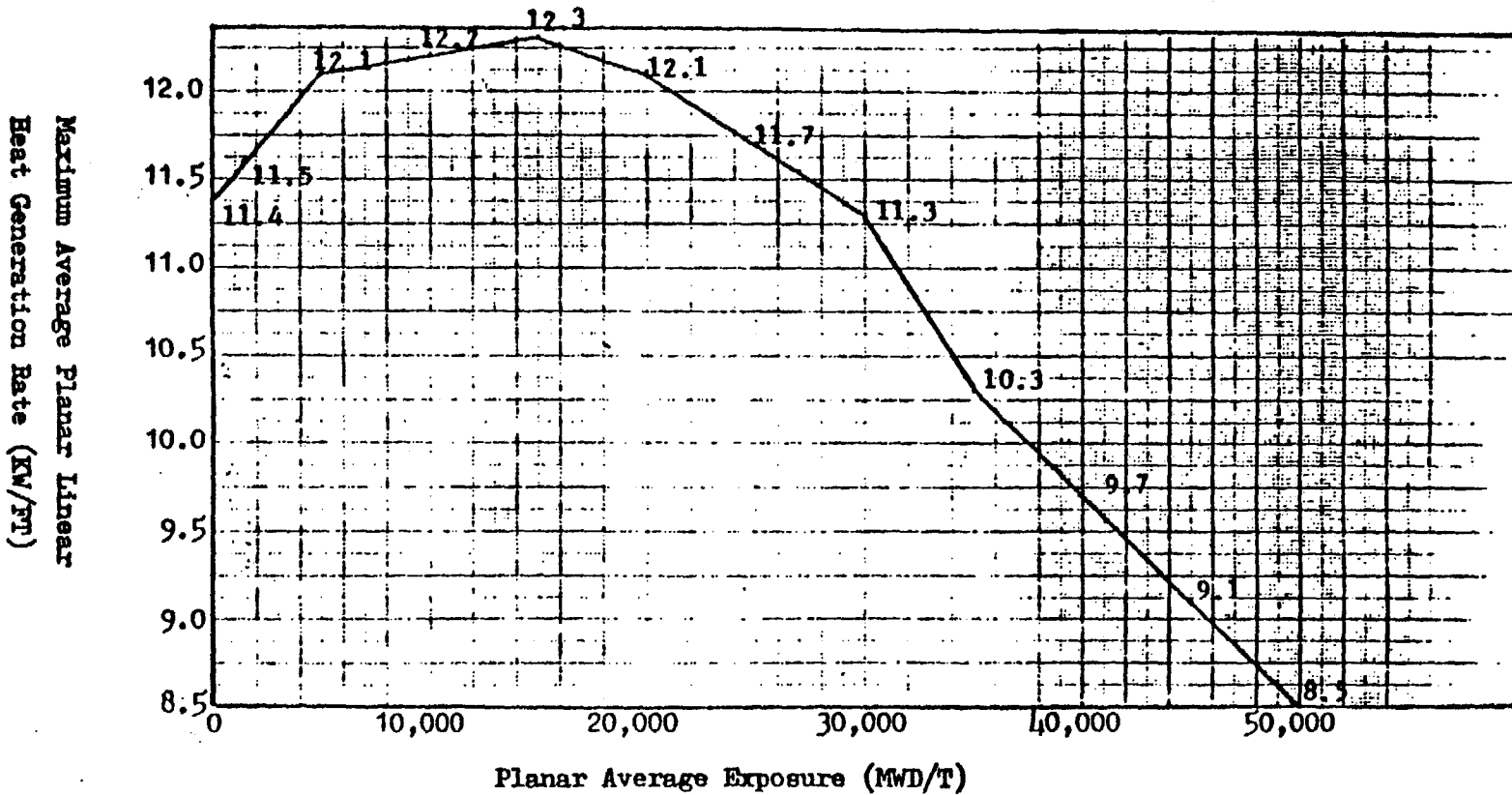


Figure 3.5.1.F Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

Maximum Average Planar Linear
Heat Generation Rate (KW/FT)

PEACH BOTTOM UNIT 2

P8X8R FUEL

TYPE P8DRB285 (Applicable to 100 mil channels)

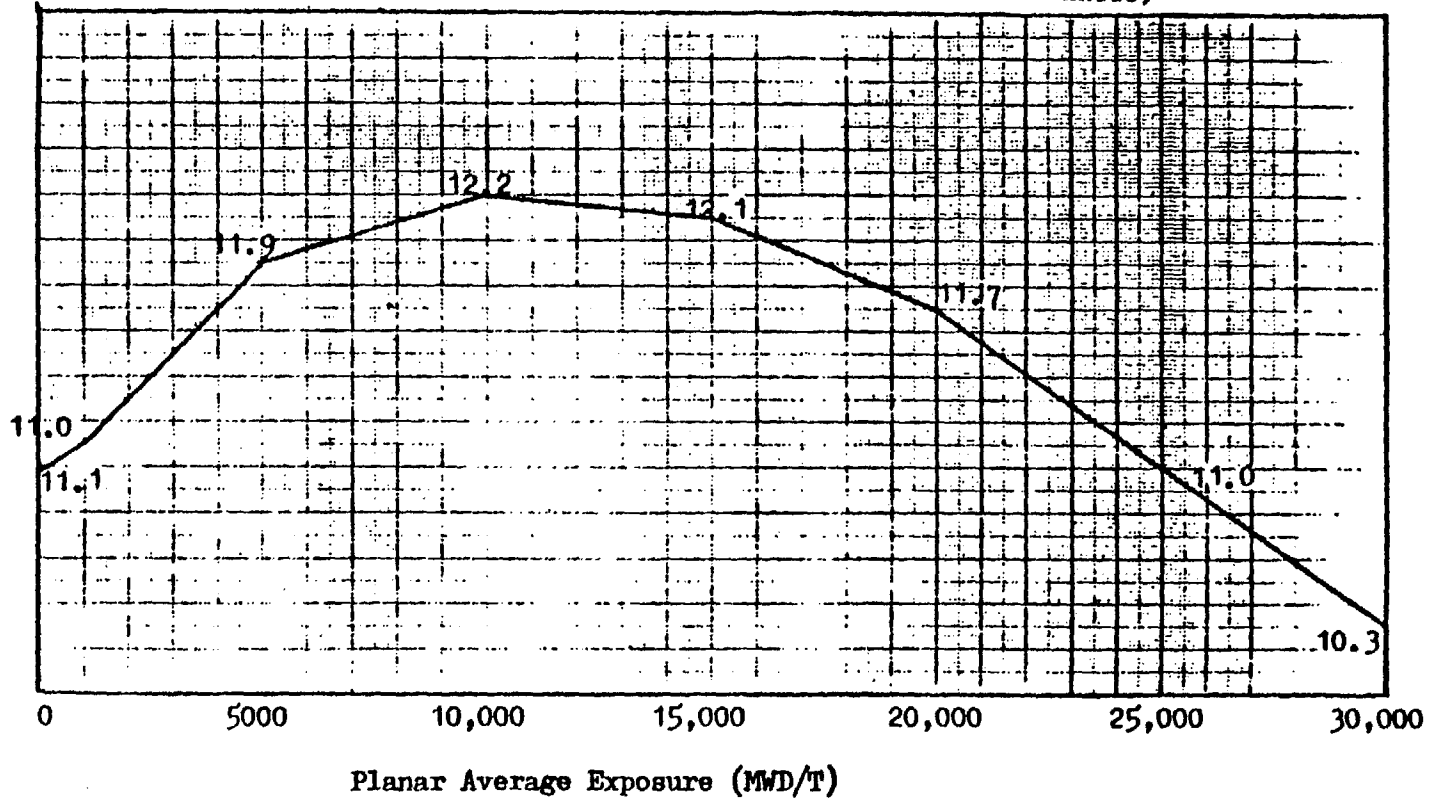


Figure 3.5.1.H Maximum Average Planar Linear Heat Generation Rate
Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

P8X8R FUEL

TYPE P8DRB284H

(Applicable to 80 mil, 100 mil, and 120 mil channels)

Maximum Average Planar Linear
Heat Generation Rate (KW/FT)

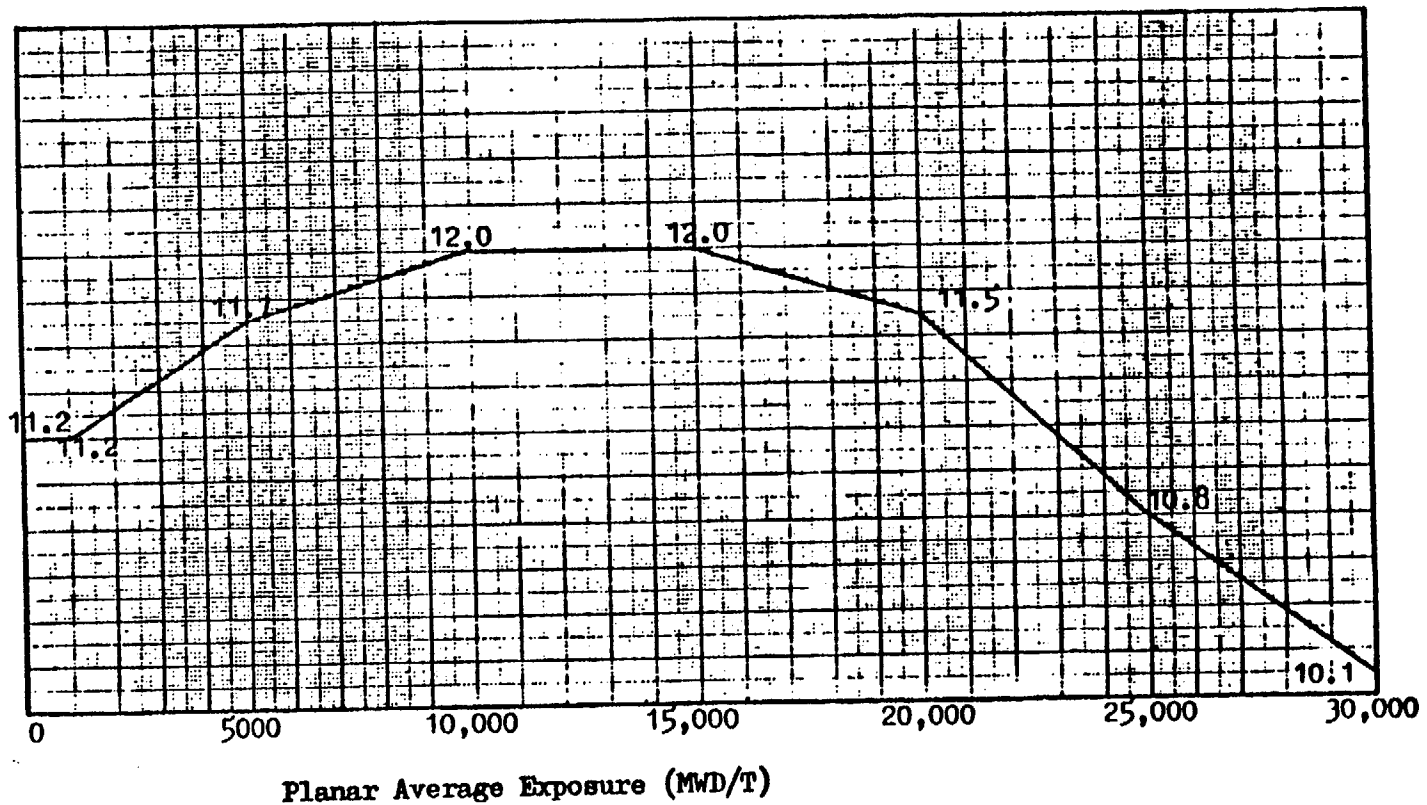


Figure 3.5.1.I Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

LIMITING CONDITIONS FOR OPERATIONSURVEILLANCE REQUIREMENTS3.6.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 for Figure 3.6.4
NDT

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. 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

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: NEDE-24011-P-A) 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 show compliance with the ASME code requirements are 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 2 with a total capacity of 79.51% of rated steam flow. The analysis of the worst overpressure transient demonstrates 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 shows that the 11 safety/relief valves limit pressure at the safety valves 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

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 8X8 fuel assemblies. 8 x 8 fuel assemblies shall contain 62 or 63 fuel rods.
- B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B₄C) compacted to approximately 70% of the theoretical density, except as described in Section 5.2.C below.
- C. Two test control rods (maximum) with up to 12 boron carbide (B₄C) pins per control rod replaced with solid hafnium metal control pins may be substituted for two B₄C control rods (Section 5.2.B above).

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.

5.5 FUEL STORAGE

- A. The new fuel storage facility shall be such that the Keff dry is less than 0.90 and flooded is less than 0.95.
- B. The Keff of the spent fuel storage pool shall be less than or equal to 0.95.
- C. Spent fuel shall only be stored in the spent fuel pool in a vertical orientation in approved storage racks.
- D. The average fuel assembly loading shall not exceed 17.3 grams U-235 per axial centimeter of total active fuel height of the assembly.

5.6 SEISMIC DESIGN

The station Class I structures and systems have been designed for ground accelerations of 0.05g (design earthquake) and 0.12g (maximum credible earthquake).



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-44

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

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

DOCKET NO. 50-277

1.0 Introduction

The Philadelphia Electric Company (the licensee) has proposed changes to the Technical Specifications of the Peach Bottom Atomic Power Station, Unit No. 2 (Reference 1). The proposed changes relate to the replacement of 292 fuel assemblies constituting refueling of the reactor core for 5th 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 average power range monitor (APRM) and rod-block monitor (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) increase in the Standby Liquid Control System (SLCS) capacity, (5) use of two control rods containing hafnium control pins, and (6) extension of exposure times on the Lead Test Assemblies (LTAs).

In support of these requests the licensee provided References 2 and 3 as part of the reload application. The licensee's proposed reload with 292 fuel assemblies consists 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, LTA, 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.

The GE topical reports, References 7 and 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 datum such as steady state and 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 is 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.

8006270 364

2.0 Evaluation

2.1 Nuclear Characteristics

The reference core loading for Cycle 5 is shown in Figure 1 of Reference 2. This core loading scheme results in quarter core symmetry. 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 1.20% $\Delta k/k$. This exceeds the minimum Technical Specification requirement of 0.38% $\Delta k/k$ for this condition.

The SLCS, with its present capability (600 ppm boron), would bring the reactor to 2.6% $\Delta k/k$ 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 3.7% $\Delta k/k$ subcritical.

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

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 (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% 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 2, Cycle 5 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 FLE. 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 (Δ CPR) reduction in the MCPR. These events have been analyzed for both the exposed fuel and the new reload fuel. Addition of the largest (Δ CPR) 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 reload consists entirely of the pressurized retrofit P8x8R fuel design. However, the Cycle 4 reload consisted of 260 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 9. In Reference 9, we 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 NRC staff. We 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 any operation approaching equilibrium cores.

The subject analysis for the retrofit fuel incorporated the local R-Factor distribution which appears in Table 5-2B of Reference 10. 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 11, 12) 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, we find 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 12) indicate that for core-wide events the P8x8R assemblies will have a slightly larger transient Δ CPR (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, we find 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 5 of Peach Bottom Unit 2, the licensee has selected option 1, which is acceptable.

During our review of Reference 1, it was noted that the licensee proposed changes in the Technical Specifications related to the analytical treatment of the transients. The proposed change affected the scram insertion times, specifically the Reactor Protection System (RPS) logic delay time. Staff discussions with the licensee and GE revealed that GE was using 50 msec for the RPS logic delay time in the reload analyses instead of the 100 msec which is consistent with the existing Technical Specifications. The proposed change was to bring the Technical Specifications into agreement with the reload analysis. This approach is inconsistent with 10 CFR 50.59 "changes, tests and experiments." Typically such changes should be supported by a written safety evaluation which provides the bases for the changes. The safety considerations involved are: (1) the proposed change reduces the End of Cycle (EOC) Δ CPR for the limiting transient, which sets the Operating Limit Minimum Critical Power Ratio (OLMCPR) (see Section 2.2.2) and (2) the proposed change decreases the vessel pressure for the Main Steam Isolation Valve (MSIV) overpressurization event. Therefore, the proposed change may be considered an unreviewed safety question as defined in §50.59(2)(iii).

Until such time that the 50 msec RPS logic delay time is specifically approved for use in reload analyses, the calculated Δ CPR for the transient analysis will be augmented with an additional Δ CPR of 0.03 (Reference 1). Likewise, the peak calculated pressure for the MSIV overpressurization event will be increased by 5 psi.

The licensee and the NRC staff have discussed this position and both are in agreement with these determinations. Results of the licensee's analyses which include the above adjustments are discussed 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 Δ CPRs 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.

<u>Transient</u>	<u>Limiting Exposure Time</u>	<u>(8x8)</u>	<u>OLMCPR (8x8R/LTA)</u>	<u>(P8x8R)</u>
Rod Withdrawal Error**	BOC 5 to EOC 5 - 1000 Mwd/t	(1.28)	(1.28)	(*)
Fuel Loading Error***	BOC 5 to EOC5 - 1000 Mwd/t	(*)	(*)	(1.30)
Load Rejection Without Bypass****	EOC5 - 1000 Mwd/t to EOC5	(1.31)	(1.31)	(1.33)

* Not Limiting

** Includes the effects of densification power striking (see Section 6.0)

*** Includes 0.02 Δ CPR allowance (see Section 2.5.3)

**** Includes 0.03 Δ CPR augmentation (see Section 2.2.2.1)

Addition of the most severe Δ CPR 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 5 operation at Peach Bottom Unit No. 2.

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, we find 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, and adjusted in accordance with Section 2.2.2.1, 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 operating state curve and the 105% rod line) are 0.29 for the 8x8R/P8x8R, 0.39 for the 8x8 and 0.85 respectively. These predicted decay ratios are all below the 1.0 Ultimate Performance 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 13) 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 11) 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 atmosphere 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-ft²-°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 11) 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-ft²-°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 conservatism will be retained in P8x8R core stability calculations.

We have 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. Our 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 NRC staff concerns through meetings, topical reports and 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 5 operation, as an interim measure, we 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 5. On the basis of the foregoing, we consider the thermal-hydraulic stability during Cycle 5 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 Emergency Core Cooling System (ECCS) Appendix K Analysis

Input data and results for the ECCS analysis have been given in References 2, 15, and 16. 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 conclude 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 2 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 MCPR greater than or equal to 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the LOCA, as described in Section 2.2).

2.5.2 Control Rod Drop Accident (CRDA)

The Peach Bottom Unit 2 Scram Reactivity Function at 20°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 CRDA. The results of this analysis indicate that the CRDA peak enthalpy under cold (20°C) conditions is 207 cal/gm. Therefore both cold (20°C) and hot (286°C) values are well below the 280 cal/gm design limit approved in Reference 5. We find these results acceptable.

2.5.3 Fuel Loading Error (FLE)

The licensee has considered the effects of postulated FLEs in the reload analysis. The FLE analysis for the most severe misloadings was performed using GE's revised analysis methods (References 17 and 18), which have previously been reviewed and approved by the NRC staff (Reference 19). The results show that the worst possible FLE will not cause violation of the 1.07 safety limit MCPR. We find 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 Linear Heat Generation Rate (LHGR) was calculated to be 17.3 Kw/ft which includes the effects of densification power spiking as required by Reference 20. Using previously accepted methods, GE calculated exposure-dependent LHGRs, which would result in 1% 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 principle effects of prepressurization with helium to three atmospheres is to increase the fuel-to-cladding gap conductance. Thus, for the same local LHGR, 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 1% 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 2 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 17.3 Kw/ft from the FLE analysis, the limiting LHGR calculated for the misoriented pressurized P8x8R fuel is acceptable.

3.0 Control Rods With Hafnium Pins

The licensee has proposed use of two demonstration control rods containing three solid hafnium absorber pins in each wing. The hafnium absorber pins will replace standard B_4C absorber pins. The purpose of the demonstration hafnium control rods is to obtain information on the performance of hafnium in a BWR environment.

The mechanical design of the hafnium control rod is the same as the standard B_4C control rod currently in use. However, because hafnium is heavier than B_4C , each demonstration control rod will weigh 16 pounds more than the standard B_4C control rod. The effect of this increased weight will be a slight increase in the two rod scram times and a negligible increase in the core average scram times. Therefore the Δ CPR results for all abnormal operational transients, as described in Section 2.2, remain unchanged.

The licensee's use of the hafnium control rods and proposed changes to the Technical Specifications required for their use are supported by the safety evaluation provided in Reference 3. Therefore we have concluded that the licensee has met the requirements under the provisions of 10 CFR 50.59, and that the proposed use of the hafnium control rods is acceptable.

4.0 Lead Test Assemblies (LTAs)

The LTAs to be operationally extended were first inserted into the core at the beginning of Cycle 2. The licensee has stated that the four LTAs will be inspected prior to insertion for Cycle 5 to ascertain fuel bundle integrity. The information obtained from the LTA demonstration program will be used to systematically determine the impact of fuel reliability and weigh the advantages of extended exposures relative to other uranium utilization improvement methods.

Results of the safety evaluation supporting Cycle 5 operation of the reconstituted and non-reconstituted LTA fuel were provided in Reference 2. Based on results of the evaluations and analysis, the accident and transient analyses of Cycle 5 are insignificantly affected and the operating limits of Cycle 5 are also unaffected.

Therefore, we support continuation of the LTA program during Cycle 5 operation in the Peach Bottom Unit 2 reactor.

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 design 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 insequence rod pattern for nonvoided conditions.

The startup test program submitted by the licensee for Cycle 4 remains acceptable for Cycle 5.

The licensee will submit to the NRC a brief written report of the startup tests within 90 days of the completion of the tests as required by the Peach Bottom specifications.

6.0 Technical Specifications

The proposed Technical Specification changes (Reference 1) for Cycle 5 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, we find the MCPRs therein listed to be consistent with and adequately supported by the Cycle 5 reload analysis, when augmented by the adjustments described in Section 2.2.2.1.

The proposed modification of the APRM and RBM 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, we find 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 NRC staff in Reference 20. This approval is contingent on augmenting abnormal operational conditions which affect the fuel locally, e.g., Rod Withdrawl Error and the Fuel Loading Error by the fuel densification power spike allowance. The licensee, as shown in Section 2.2.2.2 and 2.5.3, has met this requirement. Therefore, we find the requested deletion acceptable.

The design basis overpressure transient analysis found acceptable in Section 2.3 when augmented by the 5 psi specified in Section 2.2.2.1 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). Therefore, deletion of the reactor pressure vessel operating limit is acceptable.

Our evaluation for increasing the SLCS capacity, use of the two hafnium control rods, and continuation of the LTA program during Cycle 5 are provided in Sections 2.1, 3.0, and 4.0 respectively.

7.0 Environmental Considerations

We have determined that the amendment does 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 amendment involves 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 this amendment.

8.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

References

1. Application for Amendment of Facility Operating License DPR-44 dated March 3, 1980. Supplement dated April 28, 1980.
2. Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit No. 2, Reload No. 4, NEDO-24237 A, dated February 1980.
3. General Electric Proposed Peach Bottom Atomic Power Station Unit 2 Alternate Absorber Control Blade Test Program, NEDO-24231, Revision 1, January, 1980.
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: P. Check (NRC) to T. A. Ippolito (NRC) Review of Cooper Nuclear Station Unit 1, Reload 4, dated April 11, 1979.
10. 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.
11. General Electric letter (E. Fuller) to NRC (O. Parr), dated August 14, 1978.
12. General Electric letter (E. Fuller) to NRC (O. Parr), dated June 8, 1978.
13. "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Report NEDO-21506, dated January 1977.
14. NRC letter (O. Parr) to General Electric (G. Sherwood) dated November 21, 1978.
15. Loss-Of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Station Plant (Lead Plant), NEDO-21662, July 1977.

References (Continued)

16. Loss-Of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit No. 3, NEDO-24082, December 1977.
17. General Electric letter (R. Engle) to USNRC (D. Eisenhut), "Fuel Assembly Loading Error," dated June 1, 1977.
18. General Electric letter (R. Engle) to USNRC (D. Eisenhut), dated November 30, 1977.
19. USNRC letter (D. Eisenhut) to General Electric (R. Engle), dated November 30, 1977.
20. USNRC letter (D. Eisenhut) to General Electric (R. Gridely), dated June 9, 1978.

Dated: June 13, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-277PHILADELPHIA ELECTRIC COMPANY, ET AL.NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 70 to Facility Operating License No. DPR-44, 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, Unit No. 2 (the facility) located in York County, Pennsylvania. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications for Cycle 5 operation and involves: (1) modification of the average power range monitor and rod-block monitor 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) increase in the Standby Liquid Control System capacity, (5) use of two control rods containing hafnium control pins, and (6) extension of exposure times on the Lead Test Assemblies.

The application for the amendment complies 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

8006270 368

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 3, 1980, as supplemented April 28, 1980, (2) Amendment No. 70 to License No. DPR-44, 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 Licensing.

Dated at Bethesda, Maryland, this 13th day of June 1980.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing