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and 50-278

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 16 and 15 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. These amendments consist of changes to the Technical Specifications and are in response to your request dated December 23, 1975.

These amendments will modify the Technical Specifications to correct several editorial errors. In addition to the amendments, the Technical Specification bases for Peach Bottom Unit 2 have been revised.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendments Nos. 16 and 15
2. Safety Evaluation
3. Federal Register Notice

cc:
See next page

SEE PREVIOUS YELLOW FOR CONCURRENCE

OFFICE ➤	ORB#3	ORB#3	OELD	ORB#3		
SURNAME ➤	CParrish	TVerdery:acr		GLear		
DATE ➤	3/ /76	3/ /76	3/ /76	3/ /76		

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and 50-278

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These amendments will modify the Technical Specifications to correct several editorial errors and revise the Technical Specification bases for Peach Bottom Unit 2. Similar revision of the bases in Technical Specifications for Peach Bottom Unit 3 was accomplished by Amendment No. 14 of the license for Unit 3.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendments Nos. 16 and 15
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See next page

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3		
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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166 cc w/enclosures:

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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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No.d DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, (the licensees) dated December 23, 1975, 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: April 1, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 16

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace pages 5, 13, 14, 15, 33, 36, 91, 92, 123, 131, 133a, 140, 140a, 142, 142a, 148, 149, 184, 209 and 227 with the attached revised pages. No change has been made on pages 124, 132, 147, 183, 210 and 228. Add pages 15a, 15b, 15c, 15d, 140b, 140c, 141a and 141b.

1.0 DEFINITIONS (Cont'd.)

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.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. 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).

2. Maximum Total Peaking Factor - The Maximum Total Peaking Factor (MTPF) is the lowest Total Peaking Factor which limits a fuel type to a Linear Heat Generation Rate (LHGR) corresponding to the operating limit at 100% power.

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

4. Total Peaking Factor - The ratio of the maximum fuel rod surface heat flux in an assembly to the average surface heat flux of the core.

5. 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.

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. 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.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure ≥ 800 psia and Core Flow ≥10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB(1), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power.
The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia	
Mass flux:	0.1 to 1.25 x 10^6 lb/hr.-ft.^{2}	
Inlet Subcooling:	0 to 100 Btu/lb.	
Local Peaking:	1.61 at a corner rod to	
	1.47 at an interior rod	
Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array:	16, 64 Rods in an 8 x 8 array	
	49 Rods in a 7 x 7 array	

The required input to the statistical model are the uncertainties listed on Table 1.1-1, the nominal values of the core parameters listed in Table 1.1-2, and the relative assembly power distribution shown in Table 1.1-3. Table 1.1-4 shows the R-factor distributions that are input to the statistical model which is used to establish the safety limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The basis for the uncertainties in the core parameters are given in NEDO-20340[2] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Peach Bottom Atomic Power Station Unit 2 during any fuel cycle would not be as severe as the distribution used in the analysis.

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

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

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

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1A or 1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. 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 1.1C will be relied on 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 Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, November 1973 (NEDO-10958).

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

Table 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	6.3
Bypass void effect on TIP	4.08 (core midplane) 5.21 (core exit)
R Factor	1.6
Critical Power	3.6

Table 1.1-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

Core Thermal Power	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1078 ft^2
R-Factor	1.098 (High Enriched Bundle)
	1.154 (Low Enriched Bundle)

Table 1.1-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
1.275 to 1.325	16.8
1.225 to 1.275	8.2
1.175 to 1.225	7.2
1.125 to 1.175	5.0
1.075 to 1.125	12.0
1.025 to 1.075	4.6
0.975 to 1.025	7.0
0.875 to 0.975	4.0
0.875 to 0.925	2.0
0.825 to 0.875	4.4
0.775 to 0.825	3.0
0.675 to 0.775	2.0
0.625 to 0.675	5.0
0.575 to 0.625	4.2
0.275 to 0.575	14.6
	Sum =100

Table 1.1-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

7x7 Rod Array

R-Factor	Rod Sequence No.
1.098	1
1.083	2
1.075	3
1.062	4
1.052	5
1.042	6
1.042	7
$\leq$ 1.027	8 thru 49

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 safety/relief valve capacity 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 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements is presented in subsection 4.4 of the PSAR 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, (3-second closure of all main steamline isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1292 for Peach Bottom Unit 2 if a neutron flux scram is assumed. This results in an 83 psig margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in Section 7.2 and Figures 7-2 and 7-3 of NEDO-21104 for Peach Bottom 2. These analyses show that the 11 relief valves limit pressure at the safety valves to 49 psig below the setting of the safety valves. Therefore, the safety valves will not open.

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

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

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
	C. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels. The trip system may be in the untripped position for no more than eight hours per functional trip period for this testing.

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" direct current control bus and two (2) by "B" direct current 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 uncovery 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 uncovery 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 1.06. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four 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 1.06. |

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in a 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 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 rapid 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 or one has an upscale

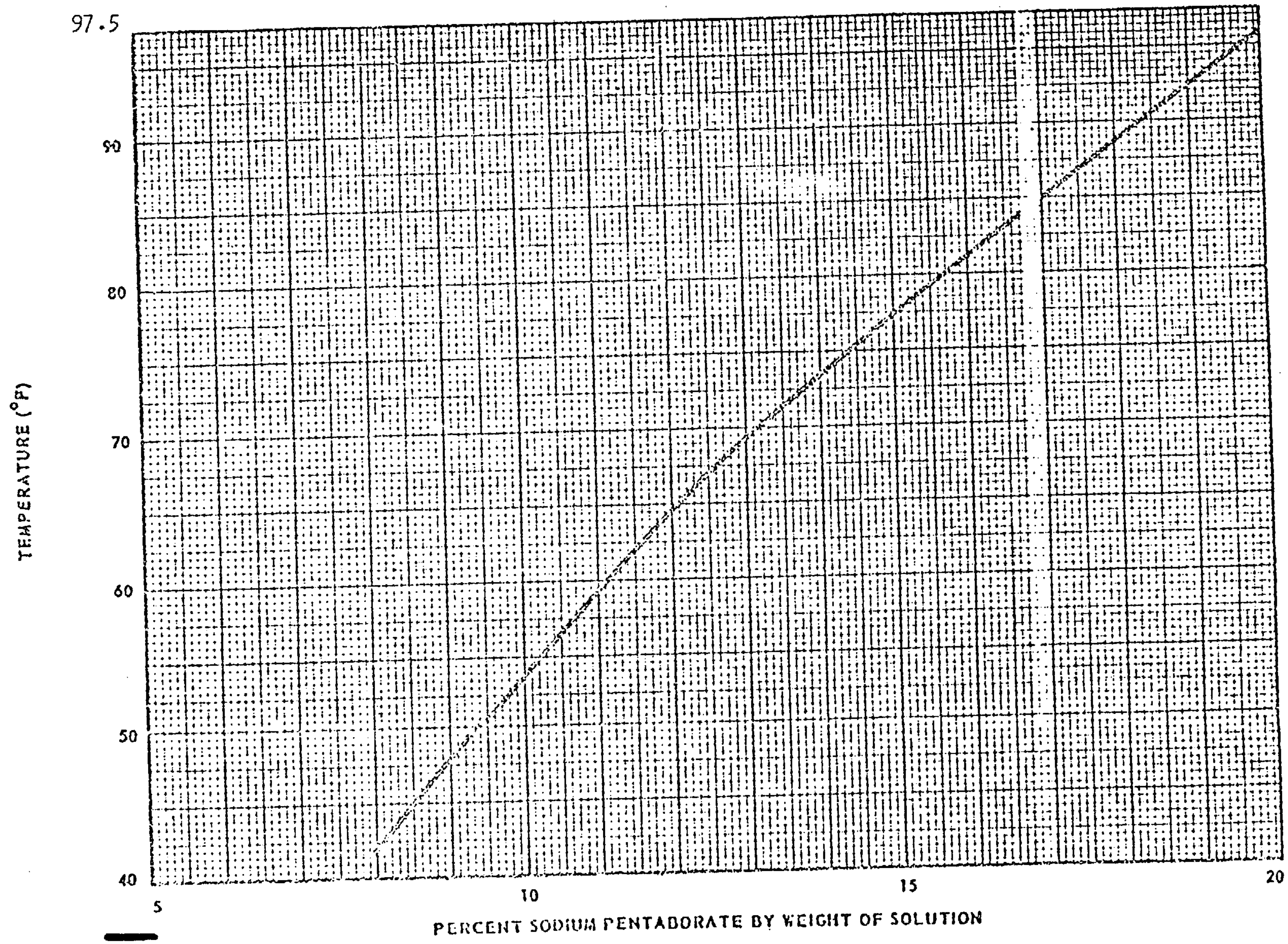
97.5
90
80
70
60
50
40
TEMPERATURE (°F)
5
10
15
20
PERCENT SODIUM PENTABORATE BY WEIGHT OF SOLUTION

Figure 3.4.2

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the core and suppression pool cooling subsystems.

Objective:

To assure the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification:

A. Core Spray and LPCI Subsystems

1. Both core spray subsystems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.2 and 3.5.F.3 below.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the Surveillance Requirements of the core and suppression pool cooling subsystems which are required when the corresponding Limiting Condition for operation is in effect.

Objective:

To verify the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification:

A. Core Spray and LPCI Subsystems

1. Core Spray Subsystem Testing.

	Item	Frequency
(a)	Simulated Automatic Actuation test.	Once/Operating Cycle
(b)	Pump Operability	Once/month
(c)	Motor Operated Valve Operability	Once/month

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.E Automatic Depressurization System (ADS)	4.5.E Automatic Depressurization System (ADS)
1. The Automatic Depressurization Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 105 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.	1. During each operating cycle the following tests shall be performed on the ADS: A simulated automatic actuation test shall be performed prior to startup after each refueling outage.
2. From and after the date that one valve in the automatic depressurization subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days the HPCI subsystem is operable.	2. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the other ADS valves and the HPCI subsystem shall be demonstrated to be operable immediately and at least weekly thereafter.
3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 105 psig within 24 hours.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.F <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>	4.5.F <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>
1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generators shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.	1. When it is determined that one diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generators shall be demonstrated to be operable immediately and daily thereafter.
2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.	
3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.	
4. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of thirty days.	

3.5.I. Average Planar LHGR

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.5.1-A or 3.5.1-B, as applicable. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

3.5.J. Local LHGR

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

$$LHGR \leq LHGR_d \left[1 - (\Delta P/P)_{max}(L/LT)\right]$$

$LHGR_d$ = Design LHGR = 18.5 kW/ft

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = 0.026

LT = Total core length = 12 ft
Unit 2

L = Axial position above bottom of core

4.5.I. Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power. This daily requirement is relaxed provided there has been no significant changes in power level or distribution as determined by the reactor engineer.

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. This daily requirement is relaxed provided there has been no significant change in power level or distribution as determined by the reactor engineer.

3.5 BASES (Cont'd)

H. Engineered 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 Linear Heat Generation Rate (APLHGR)

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 corrected for densification. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figure 3.5.1-A and B.

The calculational procedure used to establish the APLHGR shown on Figures 3.5.1 A and B 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 1. Differences in this analysis as compared to previous analyses performed with Reference 1 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.5.1-A and B; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input 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 rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

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 1.06, and an analysis of abnormal operational transients presented in Reference 1. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

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

The limiting transient which determines the required steady state MCPR limit is the rod withdrawal error transient. This transient yields the largest Δ MCPR. When added to the safety limit MCPR of 1.06 the required minimum operating limit MCPR 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 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. This rod pattern is shown in Figure 7-6 of NEDO-21104.[1/]

A brief summary of the analytical method used to determine the nuclear characteristics is given in Section 5.3 of NEDO-20360.[2/]

L. References

1. "Peach Bottom Atomic Power Station Unit 2 Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21104, November 1975.

2. General Electric BWR Generic Reload Application for 8x8 fuel, NEDO-20360, Revision 1, November 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

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

TABLE 3.5-1
PEACH BOTTOM 2 (PLUGGED)
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	14.049×10^6 lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Design Basis Recirculation Line Break Area	4.28* and 1.0
Recirculation Line Break Area for Small Breaks	1.0 and 0.07

FUEL PARAMETERS:

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (kW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio
Initial Core	7 x 7	18.5	1.5	1.17

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

*The DBA area includes: the area of the recirculation suction line (3.66 ft^2); plus the throat area of ten jet pumps (0.54 ft.) and the reactor water cleanup system line (0.08 ft.^{2}).

4.5.K <u>Minimum Critical Power Ratio (MCPR) - Surveillance Requirement</u>

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

4.5.L <u>MCPR Limits for Core Flows Other than Rated</u>

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of 1.25 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.5.1-E were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow rate (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.5.1-E, are acceptable for Peach Bottom Unit 2 operation because the operating limit MCPR is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

4.5.M <u>References</u>

1. "Peach Bottom Atomic Power Station Unit 2 Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21104, November 1975.

2. General Electric BWR Generic Reload Application for 8 x 8 fuel, NEDO-20360, Revision 1, November 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

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

Peach Bottom Unit 2

Fuel Types 1 and 3

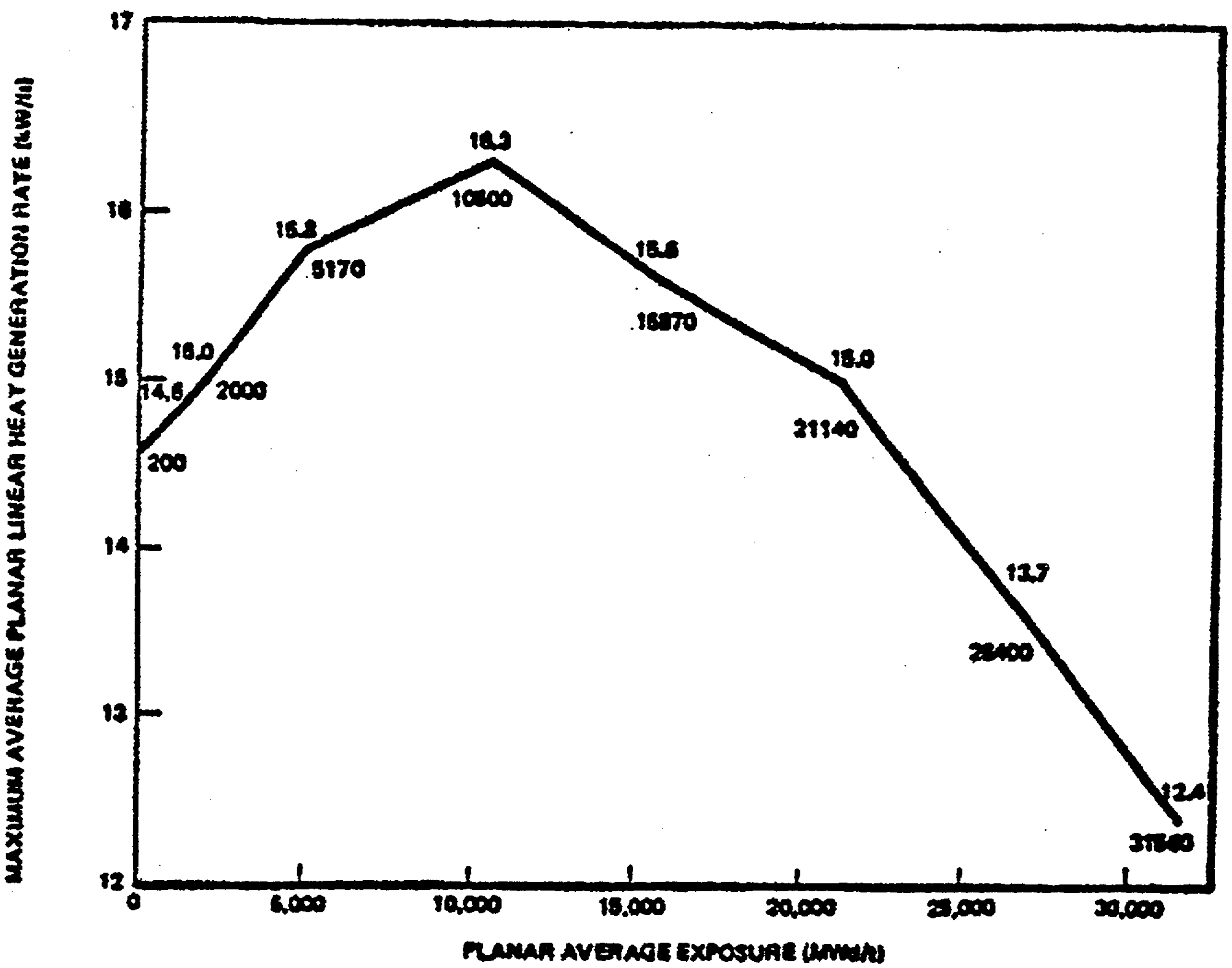

Figure 3.5.1.A **Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure**

Peach Bottom Unit 2

Fuel Type 2

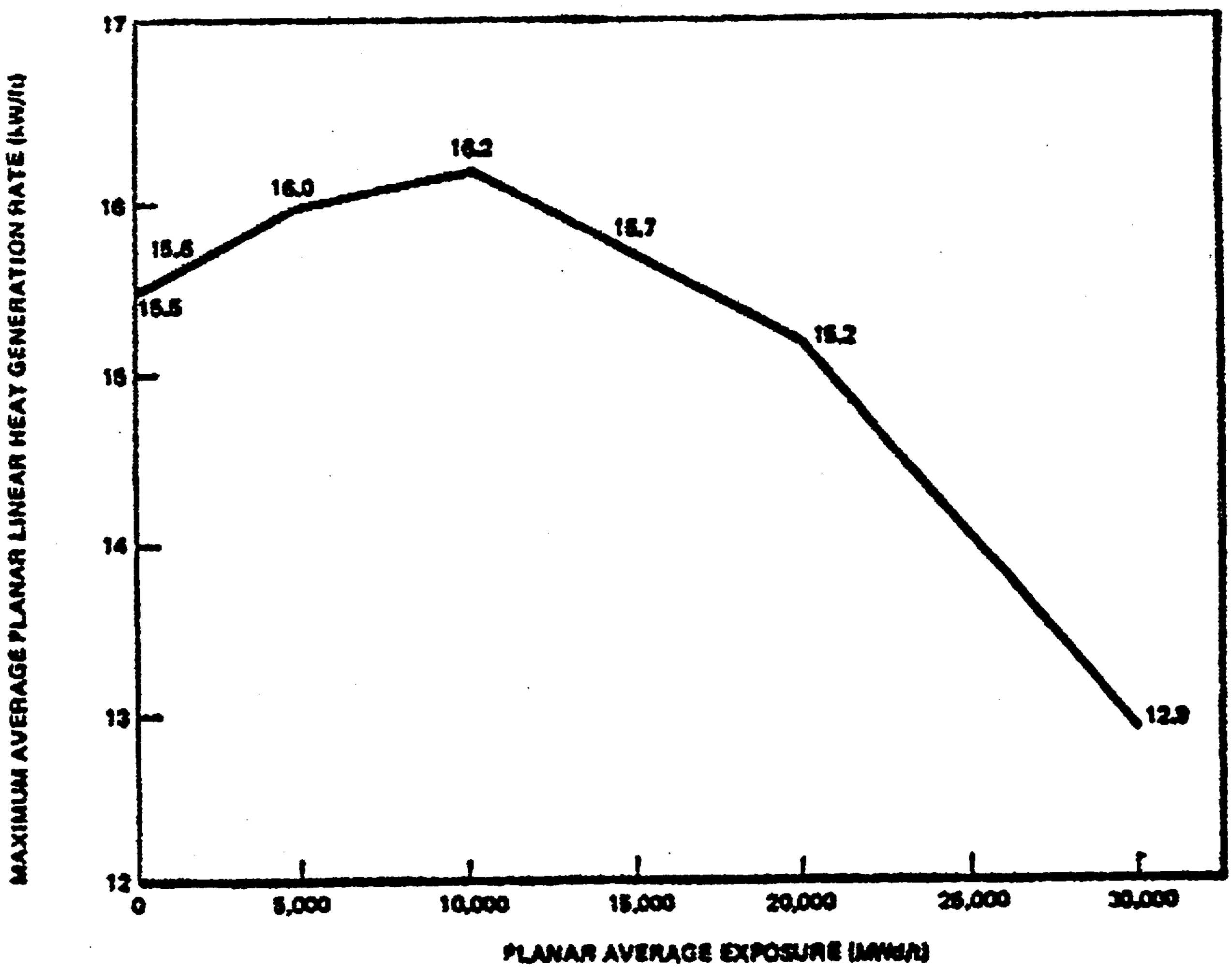

Figure 3.5.1.B **Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure**

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.D Safety and Relief Valves	4.6.D Safety and Relief Valves
1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, both safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.	1. At least one safety valve and 5 relief valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles. The set point of the safety valves shall be as specified in Specification 2.2.
2. (a) From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable. (b) From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.	2. At least one of the relief valves shall be disassembled and inspected each refueling outage.
3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to atmospheric within 24 hours.	3. The integrity of the relief safety valve bellows shall be continuously monitored. The switches shall be calibrated once per operating cycle. The accumulators and air piping shall be inspected for leakage using leak test fluid once per operating cycle.
	4. With the reactor pressure $\geq$100 psig, each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve once per operating cycle.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.6.E Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

2. Flow indications from each of the 20 jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the 20 jet pumps. If flow indication failure occurs for two or more jet pumps immediate corrective action shall be taken. If flow indication for all but 1 jet pump cannot be obtained within 12 hours on orderly shutdown shall be initiated, and the reactor shall be in a cold shutdown condition within 24 hours.

4.6.E Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

(a) The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

(b) The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

(c) The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

2. Additionally when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established pattern.

3. The baseline data required to evaluate the conditions in Specification 4.6.E.1 and 4.6.E.2 will be obtained each operating cycle.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.F (cont'd)	
1. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.	
2. The reactor shall not be operated for a period in excess of 24 hours with one recirculation loop out of service.	
3.6.G Structural Integrity The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. The reactor shall be maintained in a Cold Shutdown condition until each indication of a defect has been investigated and evaluated.	4.6.G Structural Integrity The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

1. Reactor vessel low water level.
2. High drywell pressure.
3. Reactor building ventilation exhuast high radiation.
4. Refuel floor ventilation exhaust high radiation.

GROUP 4 : The valves in Group 4 are actuated by any one of the following conditions:

1. HPCI steam line high flow.
2. HPCI steam line space high temperature.
3. HPCI steam line low pressure.

GROUP 4A: The valves in Group 4A are actuated by either of the following conditions:

1. Reactor vessel low-low water level.
2. High drywell pressure.

GROUP 5 : The valves in Group 5 are actuated by any one of the following conditions:

1. RCIC steam line high flow.
2. RCIC steam line space high temperature.
3. RCIC steam line low pressure.

GROUP 5A: The valves in Group 5A are actuated by the following condition:

1. Reactor vessel low low water level.

TABLE 3.7.2

TESTABLE PENETRATIONS WITH DOUBLE O-RINGS SEALS

Pen No.		Notes	Pen No.		Notes
N-1	Equipment Access Hatch	(1)(2)(4)(6)	N-35-A through N-35-G	TIP System	(1)(2)(4)(6)
N-2	Equipment Access and Personnel Lock	(1)(4)(7)(8)	N-200A&B	Suppression Chamber Access Hatch	(1)(2)(4)(6)
N-4	Drywell Head Access Hatch	(1)(2)(4)(6)	N-213A&B	Construction Drain	(1)(2)(4)(6)
N-6	CRD Removal Hatch	(1)(2)(4)(6)			

TABLE 3.7.3

TESTABLE PENETRATIONS WITH TESTABLE BELLOWS

Pen No.		Notes	Pen No.		Notes
N-7A	Primary Steamline 'A'	(1)(2)(4)(6)	N-13A	RHR Pump Discharge	(1)(2)(4)(6)
N-7B	Primary Steamline 'B'	(1)(2)(4)(6)	N-13B	RHR Pump Discharge	(1)(2)(4)(6)
N-7C	Primary Steamline 'C'	(1)(2)(4)(6)	N-14	Reactor Water Cleanup Line	(1)(2)(4)(6)
N-7D	Primary Steamline 'D'	(1)(2)(4)(6)	N-16A	Core Spray Pump Discharge	(1)(2)(4)(6)
N-9A	Feedwater Line 'A'	(1)(2)(4)(6)	N-16B	Core Spray Pump Discharge	(1)(2)(4)(6)
N-9B	Feedwater Line 'B'	(1)(2)(4)(6)	N-17	RPV Head Spray	(1)(2)(4)(6)
N-11	Steam Line to HPCI Turbine	(1)(2)(4)(6)	N-201A through N-201H	Suppression Chamber to Drywell Vent Line	(1)(2)(4)(6)
N-12	RHRS Shutdown Pump Supply	(1)(2)(4)(6)			

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.C (Cont'd.)	
9. The containment shall not be purged except through the standby gas treatment system.	
10. a. Except as specified in 3.8.C.10b below, two monitors downstream of the recombiners shall be operable during power operation.	
b. If the above specified required hydrogen monitors are not operable, an orderly reduction of power shall be initiated to bring the hydrogen production rate to less than 4% of the off-gas flow rate.	
3.8.D <u>Mechanical Vacuum Pump</u>	4.8.D <u>Mechanical Vacuum Pump</u>
1. The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity in the steam lines whenever the main steam isolation valves are open.	At least once during each operating cycle verify automatic securing and isolation of the mechanical vacuum pump.
2. If the limits of 3.8.D.1 are not met the vacuum pump shall be isolated.	

TABLE 4.8.1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS

Sample Type	Sample Frequency	Sample Analysis	Sample Detectable Limit (5) (2)
Waste Tank to be released	Each Batch	Gamma Scan (3)	5×10^{-7} uCi/ml
		Gross beta	1×10^{-7} uCi/ml
Proportional Composite of Batches	Monthly	Tritium	1×10^{-5} µCi/ml
		Gross alpha	1×10^{-7} µCi/ml (4)
Proportional Composite of Batches	Monthly	Sr^{89}	5×10^{-8} uCi/ml
	Quarterly	Sr^{90}	5×10^{-8} uci/ml
One Batch	Monthly	dissolved noble gases	1×10^{-4} uCi/ml

NOTES:

1. Certain mixtures of radionuclides may cause interference in the measurement of individual radionuclides at their detectable limit especially if other radionuclides are at much higher concentrations. Under these circumstances use of known ratios of radionuclides will be appropriate to calculate the levels of such radionuclides.

2. The above sample detectable limits are applicable to grab samples used to determine liquid waste release levels. Reported data shall reflect any improvment in detectable limits as such improvments are achieved.

3. Significant radionuclides are to be identified and where possible, quantitative values obtained.

4. Self absorption will result in a higher detectable limit for alpha counting.

5. Sample detectable limits are subject to revision. The values listed are believed to be attainable.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A.5.b (Cont'd.) directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.	4.10.A
c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.	
d. An appropriate number of SRM's are available as defined in specification 3.10.B.	
6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:	
a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.	
B. Core Monitoring	B. Core Monitoring
During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:	Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B (Cont'd.)	4.10.B (Cont'd.)
1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)	
2. The SRM shall have a minimum of 3 cps with all rods fully inserted in the core.	
C. Spent Fuel Pool Water Level	C. Spent Fuel Pool Water Level
Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8 1/2' above the top of the fuel.	Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

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

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

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No.d DPR- 56

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, (the licensees) dated December 23, 1975, 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: April 1, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 15

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace pages 92, 123, 184, 209 and 227 with the attached revised pages.

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 1.06.

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in a 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 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 rapid 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 or one has an upscale

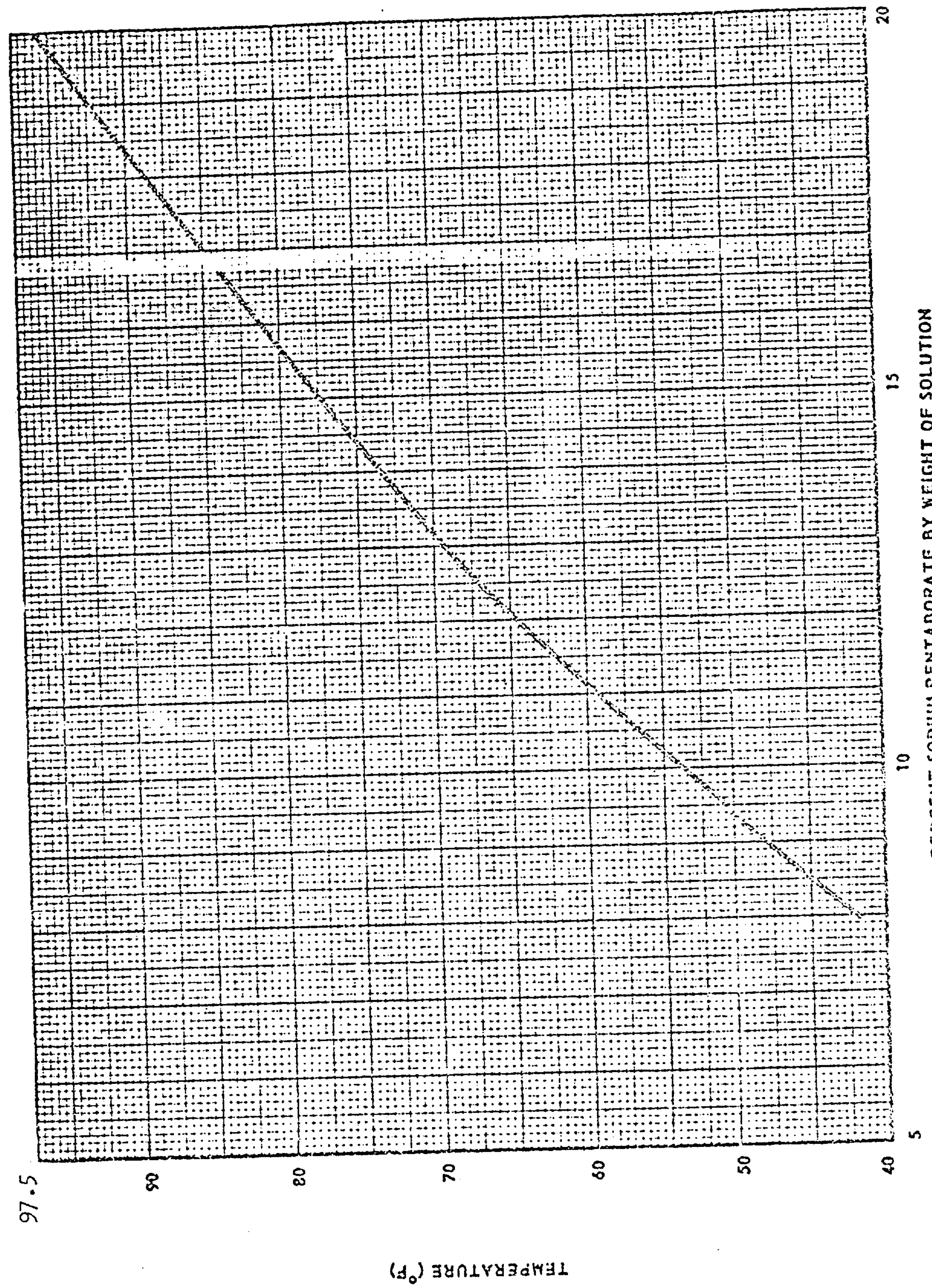
PERCENT SODIUM PENTABORATE BY WEIGHT OF SOLUTION
5
10
15
20
TEMPERATURE (°F)
40
50
60
70
80
90
97.5

Figure 3.4.2

TABLE 3.7.2

TESTABLE PENETRATIONS WITH DOUBLE O-RINGS SEALS

Pen No.		Notes	Pen No.		Notes
N-1	Equipment Access Hatch	(1)(2)(4) (6)	N-35-A through N-35-G	TIP System	(1)(2)(4) (6)
N-2	Equipment Access and Personnel Lock	(1)(4)(7) (8)	N-200A&B	Suppression Chamber Access Hatch	(1)(2)(4) (6)
N-4	Drywell Head Access Hatch	(1)(2)(4) (6)	N-213A&B	Construction Drain	(1)(2)(4) (6)
N-6	CRD Removal Hatch	(1)(2)(4) (6)			

TABLE 3.7.3

TESTABLE PENETRATIONS WITH TESTABLE BELLOWS

Pen No.		Notes	Pen No.		Notes
N-7A	Primary Steamline 'A'	(1)(2)(4) (6)	N-13A	RHR Pump Discharge	(1)(2)(4) (6)
N-7B	Primary Steamline 'B'	(1)(2)(4) (6)	N-13B	RHR Pump Discharge	(1)(2)(4) (6)
N-7C	Primary Steamline 'C'	(1)(2)(4) (6)	N-14	Reactor Water Cleanup Line	(1)(2)(4) (6)
N-7D	Primary Steamline 'D'	(1)(2)(4) (6)	N-16A	Core Spray Pump Discharge	(1)(2)(4) (6)
N-9A	Feedwater Line 'A'	(1)(2)(4) (6)	N-16B	Core Spray Pump Discharge	(1)(2)(4) (6)
N-9B	Feedwater Line 'B'	(1)(2)(4) (6)	N-17	RPV Head Spray	(1)(2)(4) (6)
N-11	Steam Line to HPCI Turbine	(1)(2)(4) (6)	N-201A through N-201H	Suppression Chamber to Drywell Vent Line	(1)(2)(4) (6)
N-12	RHRS Shutdown Pump Supply	(1)(2)(4) (6)			

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.C (Cont'd.)	
9. The containment shall not be purged except through the standby gas treatment system.	
10. a. Except as specified in 3.8.C.10b below, two monitors downstream of the recombiners shall be operable during power operation.	
b. If the above specified required hydrogen monitors are not operable, an orderly reduction of power shall be initiated to bring the hydrogen production rate to less than 4% of the off-gas flow rate.	
3.8.D <u>Mechanical Vacuum Pump</u>	4.8.D <u>Mechanical Vacuum Pump</u>
1. The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity in the steam lines whenever the main steam isolation valves are open.	At least once during each operating cycle verify automatic securing and isolation of the mechanical vacuum pump.
2. If the limits of 3.8.D.1 are not met the vacuum pump shall be isolated.	

LIMITING CONDITIONS FOR OPERATION

3.10.A.5.b (Cont'd.)

directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are available as defined in specification 3.10.B.

6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:

a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

SURVEILLANCE REQUIREMENTS

4.10.A

B. Core Monitoring

Prior to making any alterati[illegible] to the core, the SRM's shall functionally tested and chec[illegible] for neutron response. There-after, while required to be operable, the SRM's will be checked daily for response.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY LICENSE NO. DPR-44 AND

AMENDMENT NO. 15 TO FACILITY LICENSE NO. DPR-56

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

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

Introduction

By letter dated December 23, 1975, Philadelphia Electric Company proposed changes to the Technical Specifications appended to Facility Operating Licenses Nos DPR-44 and DPR-56, for Peach Bottom Atomic Power Station, Units 2 and 3. The proposed changes involve correction of several editorial errors in various sections of the Technical Specifications. Additionally, the Commission is incorporating revised Technical Specification bases for Peach Bottom Unit 2 which are similar to the revised bases previously issued to Peach Bottom Unit 3 with Amendment No 14.

Evaluation

The changes proposed by the licensee involve purely editorial corrections and will not affect the operation of the facilities. The revised bases in Technical Specification for Peach Bottom Unit 2 are being issued to provide a better justification for the revised operating limits, using the General Electric Thermal Analysis Basis (GETAB), which were authorized by Amendment Nos. 15 of the Unit 2 license. Similar bases have previously been issued to Peach Bottom Unit 3 with Amendment No. 14. The revised bases are purely informational and will not affect the operation of the facility.

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 insiginificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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.

Date: April 1, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

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

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 16 and 15 to Facility Operating Licenses Nos. DPR-44 and DPR-56, respectively, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station, Units 2 and 3, located in Peach Bottom, York County, Pennsylvania. The amendment is effective as of its date of issuance.

The amendments will modify the Technical Specifications to correct several editorial errors, and will not affect the operation of either facility.

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 is not required since the amendment does not involve a significant hazards consideration.

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 statement, negative declaration or 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 December 23, 1975, (2) Amendments Nos 16 and 15 to Licenses Nos. DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street N.W., Washington, D. C. and at the Martin Memorial Library, 159 E. Market Street, York, Pennsylvania 17401.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 1 day of April 1976

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors