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

September 14, 1993

Docket No. 50-278

Mr. George A. Hunger, Jr.
Director-Licensing, MC 52A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: EXPANDED OPERATING DOMAIN (ARTS/MELLLA) TECHNICAL SPECIFICATIONS, PEACH BOTTOM ATOMIC POWER STATION, UNIT 3 (TAC NO. M86133)

The Commission has issued the enclosed Amendment No. 184 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit 3. The amendment consists of changes to the Technical Specifications in response to your application dated April 1, 1993, as supplemented by letters dated April 7, July 16, and August 20, 1993.

These amendments implement an expanded power-to-flow operating domain supported by the Average Power Range Monitor, Rod Block Monitor, Technical Specifications Improvement/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) (NEDC-32162P, Revision 1, February 1993) submitted in your April 1, 1993, application. In addition, the NRC staff has also reviewed the SAFER/GESTR Loss-of-Coolant-Accident Analysis submitted by your letter dated April 7, 1993.

The amendment is effective upon startup from Refueling Outage 3R09. You are requested to inform the staff when you have implemented the provisions of this amendment. In your application, you proposed that the ARTS/MELLLA amendments apply to both Units 2 and 3, however, you noted that the ARTS/MELLLA modifications would not be made on Unit 2 until refueling outage 2R010, which is currently scheduled for fall 1994. In order to preclude confusion between the effective date for the Unit 2 ARTS/MELLLA amendment and any subsequent amendment requests that might affect the same TS pages, the staff will issue the ARTS/MELLLA amendment for Unit 2 just prior to 2R010.

220005

NRC FILE CENTER COPY

9309230202 930914
PDR ADOCK 05000278
P PDR

DF01 1/1

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 184 to DPR-56
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	MO'Brien(2)	CGrimes, 11E21	CAnderson, RGN-I
NRC & Local PDRs	JShea	RJones	JDonoghue
PDI-2 Reading	OGC	ACRS(10)	
SVarga	DHagan, 3206	OPA	
JCalvo	GHill(4), P1-22	OC/LFMB	
MBoyle	RJones, 8E23	EWenzinger, RGN-I	

*Previously Concurred

OFC	:PDI-2/LA	:PDI-2/PM	:SRXB/BC*	:OGC	:PDI-2/D For	
NAME	:MO'Brien	:JShea	:RJones	: U Young	:MBoyle	:
DATE	:9/1/93	:9/13/93	:08/16/93	: 9/3/93	:9/14/93	:

OFFICIAL RECORD COPY
FILENAME: PB86132.AMD

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

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

Enclosures:
1. Amendment No. 184 to DPR-56
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

Philadelphia Electric Company
ATTN: Mr. D. B. Miller, Vice President
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

Philadelphia Electric Company
ATTN: Regulatory Engineer, A1-2S
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

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

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

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

Carl D. Schaefer
External Operations - Nuclear
Delmarva Power & Light Company
P.O. Box 231
Wilmington, DE 19899

Mr. William P. Dornsife, Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
P. O. Box 8469
Harrisburg, Pennsylvania 17105-8469

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

Public Service Commission of Maryland
Engineering Division
ATTN: Chief Engineer
231 E. Baltimore Street
Baltimore, MD 21202-3486

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

Mr. John Doering, Chairman
Nuclear Review Board
Philadelphia Electric Company
955 Chesterbrook Boulevard
Mail Code 52C-1
Wayne, Pennsylvania 19087

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

PHILADELPHIA ELECTRIC COMPANY

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et. al. (the licensee) dated April 1, 1993, and supplemented by letters dated April 7, July 16, and August 20, 1993, 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 or safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-56 is hereby amended to read as follows:

9309240058 930914
PDR ADOCK 05000278
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 184, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective following startup from Refueling Outage 3R09.

FOR THE NUCLEAR REGULATORY COMMISSION

James C. Stone

for Michael L. Boyle, Acting Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 14, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 184

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove	Insert
1	1
2	2
3	3
4	4
9	9
9a	9a
10	10
11	11
11a	11a
15	15
16	16
19	19
20	20
35	35
37	37
40	40
54	54
73	73
74	74
74a	74a
133a	133a

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove	Insert
133b	133b
140	140
140b	140b
140c	140c
141a	141a
141b	141b
256	256

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 with the vessel head removed and fuel in the vessel.

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.

Average Planar Linear Heat Generation Rate (APLHGR) - The APLHGR shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod, for all the fuel rods in the specific bundle at the specific height, divided by the number of fuel rods in the fuel bundle at that height.

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.

Core Operating Limits Report (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current Operating Cycle. These cycle-specific core operating limits shall be determined for each Operating Cycle in accordance with specification 6.9.1.e. Plant operation within these limits is addressed in individual Specifications.

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).

Dose Equivalent I-131 - That concentration of I-131 (Ci/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.

Downscale Trip Set Point (DTSP) - The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

1.0 DEFINITIONS (Cont'd)

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 core spray pump to verify that it runs and that it pumps the required volume of water).

Gaseous Radwaste Treatment System - Any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

High (power) Trip Set Point (HPTS) - The high power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable above 85% reactor thermal power.

Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212 F.

Hot Standby Condition - Hot Standby Condition means operation with coolant temperature greater than 212 F, system pressure less than 1055 psig, and the mode switch in the Startup/Hot Standby position. The main steam isolation valves may be opened to provide steam to the reactor feed pumps.

Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

1.0 DEFINITIONS (Cont'd)

Instrument or Channel Calibration - An instrument or channel calibration means the adjustment of an instrument or channel signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument or channel monitors. The known value of the parameter shall be injected into the channel or instrument as close to the primary sensor as practicable.

Instrument or Channel Check - An instrument or channel check is a qualitative determination of acceptable operability by observation of instrument or channel behavior during operation. This determination shall include, where possible, comparison of the instrument or channel with other independent instruments measuring the same variable.

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

Intermediate (power) Trip Set Point (ITSP) - The intermediate power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 65% and 85% reactor thermal power.

Limiting Conditions for Operations (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.

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

1.0 DEFINITIONS (Cont'd)

(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 shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements etc., of a logic circuit, from sensor through and including the actuated device, to verify Operability. The Logic System Functional Test may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

Low (power) Trip Set Point (LTSP) - The low power trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting applicable between 30% and 65% reactor thermal power.

MAPFAC(F) (MAPLHGR Flow Factor) - A core flow dependent multiplication factor used to flow bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MAPFAC(P) (Power Dependent MAPLHGR Multiplier) - A core power dependent multiplication factor used to power bias the standard Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit.

MEMBERS OF THE PUBLIC - Members of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core. Associated with the minimum critical power ratio is a core flow dependent (MCPR(F)) and core power dependent (MCPR(P)) minimum critical power ratio.

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.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation, or 1.08 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66W + 71\% - 0.66\ \Delta W$
(Clamp @ 120%)

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculating flow rate in percent of design.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

ΔW = Difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.66 ΔW) is accomplished by correcting the flow input of the flow biased scram to preserve the original (two loop) relationship between APRM scram setpoint and recirculation drive flow or by adjusting the APRM flux trip setting. ΔW = 0 for two loop operation. The APRM flux scram trip setting shall not exceed 120% of rated thermal power.

SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING

2.1.A (Cont'd)

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.

Amendment No. ~~14~~, ~~33~~, ~~41~~, ~~62~~, 77, ~~79~~, ~~107~~, ~~130~~, ~~133~~, 184

SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia)

When the reactor pressure is ≤ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

B. APRM Rod Block Trip Setting

$S_{RB} \leq (0.66\ W + 59\% - 0.66\ \Delta W)$
(Clamp @ 108%)

where:

S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of design.

ΔW = Difference between two loop and single loop effective recirculation drive flow at the same core flow. During single loop operation, the reduction in trip setting ($-0.66\ \Delta W$) is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between APRM Rod block setpoint and recirculation drive flow or by adjusting the APRM Rod block trip setting. $\Delta W = 0$ for two loop operation.

The APRM rod block trip setting shall not exceed 108% of rated thermal power.

Amendment No. ~~14~~, ~~33~~, ~~41~~, ~~62~~, 77, ~~150~~, 184

SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia)

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than minus 160 inches indicated level (378 inches above vessel zero).

B. APRM Rod Block Trip Setting

C. Scram and isolation reactor low water level — ≥ 538 in. above vessel zero (0" on level instruments)

Amendment No. 77, 79, 113, 130, 133, 184

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 Units 2 and 3 has a sequence annunciation program which will indicate the sequence in which events such as a 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 minus 160 inches indicated level (378 inches above vessel zero) provides adequate margin to assure sufficient cooling during shutdown conditions. This level will be continuously monitored.

E. References

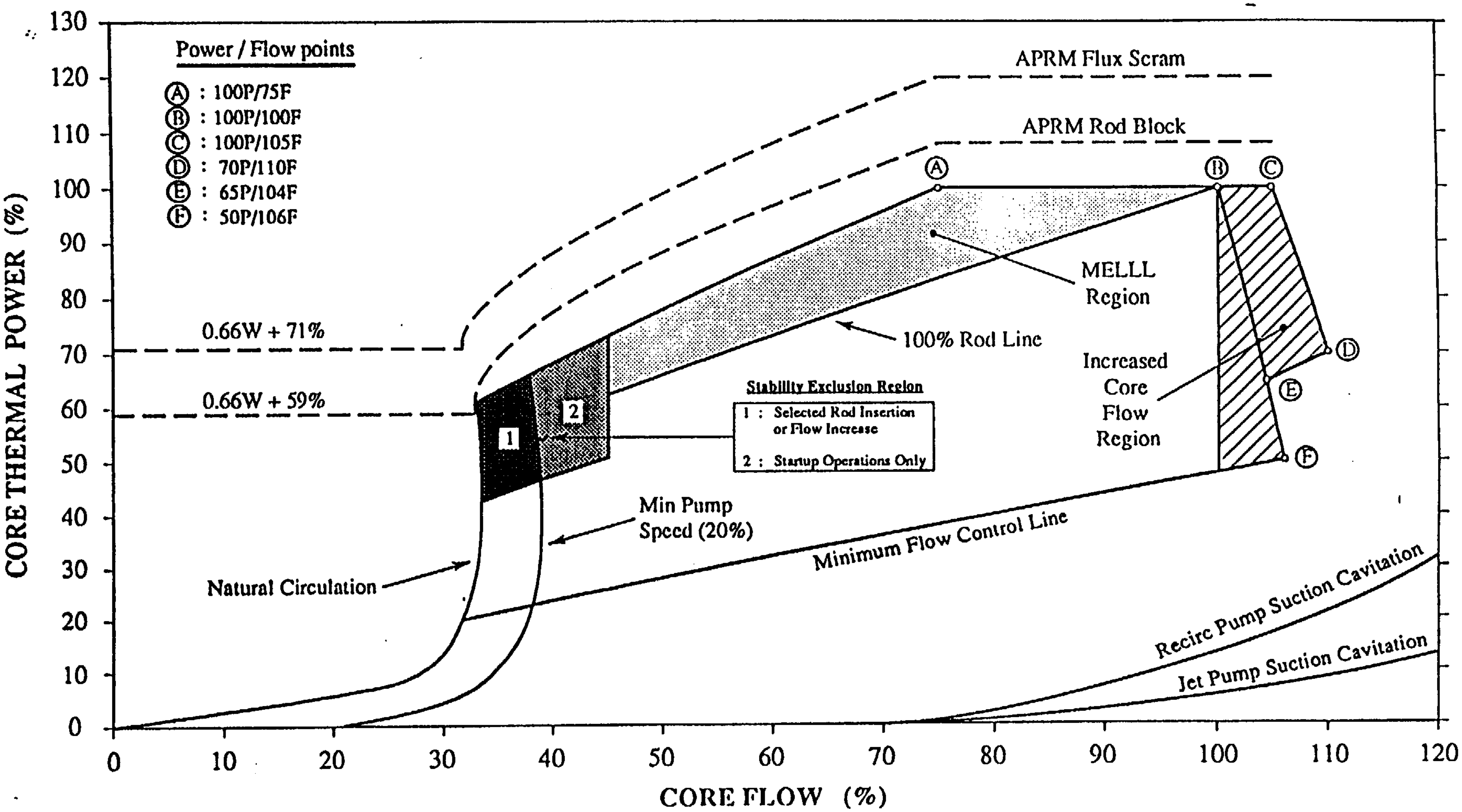
1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, January 1977 (NEDO-10958-A).

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

3. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

4. "Peach Bottom Atomic Power Station Units 2 and 3 Single-Loop Operation", NEDO-24229-1, May 1980.

5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993.



APRM Flow Bias Scram
Relationship to Normal Operating Conditions

Figure 1.1-1

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

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

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, adjusted for power and flow as specified in the COLR.

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. 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 dos 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.

Amendment No. ~~14~~, ~~41~~, ~~62~~, ~~79~~, ~~153~~, 184

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 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

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

LIMITING CONDITIONS FOR OPERATION

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:

A. 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.

B. 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 50 milliseconds. Otherwise, the affected trip system shall be placed in the tripped condition, or the action listed in Table 3.1.1 for the specific trip function shall be taken.

SURVEILLANCE REQUIREMENTS

4.1 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. DELETED

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Items	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	1	Mode Switch In Shutdown		x	x	x	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		x	x	x	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	x	x	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		x	x	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(0.66W+71%-0.66ΔW) (Clamp @ 120%) (12) (13)			x	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	x	x	x	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	x	x		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	x(9)	x	x	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	x(8)	x(8)	x	4 Instrument Channels	A
11	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. W = Loop Recirculation flow in percent of design.

    Delta W = The difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.66 delta W) is accomplished by correcting the flow input of the flow biased High Flux trip setting to preserve the original (two loop) relationship between APRM High Flux setpoint and recirculation drive flow or by adjusting the APRM Flux trip setting. Delta W equals zero for two loop operation.

    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 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
4 (2)	APRM Upscale (Flow Biased)	(0.66W+59%-0.66ΔW) (Clamp at 108% max)	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	≤12%	6 Inst. Channels	(10)
4	APRM Downscale	≥2.5 indicated on scale	6 Inst. Channels	(10)
1 (7)(11)	Rod Block Monitor (Power Biased)	(RTP ≥85%), S_{RB} ≤HTSP (65% ≤RTP <85%), S_{RB} ≤ITSP (30% ≤RTP <65%), S_{RB} ≤LTSP	2 Inst. Channels	(1)
1 (7)(11)	Rod Block Monitor Downscale	≥DTSP	2 Inst. Channels	(1)
6	IRM Downscale (3)	≥2.5 indicated on scale	8 Inst. Channels	(10)
6	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(10)
6	IRM Upscale	≤108 indicated on scale	8 Inst. Channels	(10)
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)
1	Scram Discharge Instrument Volume High Level	≤25 gallons	1 Inst. Channel	(9)

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. W = Loop Recirculation flow in percent of design.

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

   ΔW is the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between the rod block setpoint and recirculation drive flow. $\Delta W = 0$ for two loop operation.

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 ≤ 30%.

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

Amendment No. ~~33~~, ~~41~~, ~~62~~, 77, ~~79~~, ~~150~~, ~~155~~, 184

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.

10. For the Startup (for IRM rod block) and the Run (for APRM rod block) positions of the Reactor Mode Selector Switch and with the number of OPERABLE channels:

    a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

    b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

11. The values for HTSP, ITSP, LTSP, and DTSP are specified in the CORE OPERATING LIMITS REPORT.

LIMITING CONDITIONS FOR OPERATION

3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure and reactor power/flow multipliers (provided in the CORE OPERATING LIMITS REPORT) shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limit for the most limiting lattice (excluding natural uranium) specified in the CORE OPERATING LIMITS REPORT during two recirculation loop operations. 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 design LHGR.

LHGR ≤ LHGRd

LHGRd = Design LHGR
The values for Design LHGR for each fuel type are specified in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.5.I Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure and reactor power/flow multipliers (provided in the CORE OPERATING LIMITS REPORT) shall be checked daily during reactor operation at ≥25% rated thermal power.

4.5.J Local LHGR

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

LIMITING CONDITIONS FOR OPERATION

3.5.J Local LHGR (Cont'd)

If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR 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 LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

1. During power operation the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Specification 3.5.K.2 or 3.5.K.3, or MCPR(F), or the MCPR operating limit as determined by application of MCPR(P), whichever is greater. MCPR(F) and MCPR(P) are provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within one (1) hour to restore MCPR to within prescribed limits. If the MCPR 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 MCPR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at ≥25% rated thermal power.

2. Except as provided in Specification 3.5.K.3, the verification of the applicability of 3.5.K.2.a Operating Limit MCPR Values shall be performed every 120 operating days by scram time testing 19 or more control rods on a rotation basis and performing the following:

a. The average scram time to the 20% insertion position shall be:

$$\tau\ ave \le \tau\ B$$

b. The average scram time to the 20% insertion position is determined as follows:

$$\tau\ ave = \frac{\sum_{i=1}^{n} Ni\ \tau\ i}{\sum_{i=1}^{n} Ni}$$

where: n = number of surveillance tests performed to date in the cycle.

Amendment No. ~~150~~, ~~155~~, 184

3.5 BASES (Continued)

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 indicated 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 heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is the LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the applicable figure for each fuel type in the CORE OPERATING LIMITS REPORT, and must be adjusted for power and flow by application of MAPFAC(P) and MAPFAC(F). MAPFAC(P) and MAPFAC(F) are provided in the CORE OPERATING LIMITS REPORT.

Only the most limiting APLHGR operating limits are shown in the figures for the multiple lattice fuel types. Compliance with the lattice-specific APLHGR limits is ensured by using the process computer. When an alternate method to the process computer is required (i.e. hand calculations and/or alternate computer simulation), the most limiting lattice APLHGR limit for each fuel type shall be applied to every lattice of that fuel type.

The calculational procedure used to establish the APLHGR is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (G.E.) 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. The plant specific results using the Reference 4 methodology are presented in Reference 8.

3.5.K. BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Analysis of the abnormal operational transients is presented in References 7, 10 and 11. Input data and operating conditions used in this analysis are shown in References 7, 10 and 11 in the Supplemental Reload Licensing Analysis.

3.5.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 in-core 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 indication value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominately occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate including MELLL operation with implementation of ARTS restrictions (Ref. 11).

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. Letter, C. O. Thomas (NRC) to J. F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), 'The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident'," June 1, 1984.

5. DELETED.

6. DELETED.

7. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

8. "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analyses," NEDC-32163P, January, 1993.

9. DELETED.

10. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

11. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993.

4.5.K Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

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 MCPR Limits for Core Flows Other Than Rated

A flow dependent MCPR limit, MCPR(F), is necessary to assure that the safety limit MCPR is not violated during recirculation flow increase events. The design basis flow increase event is a slow-power increase event which is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Flow runout events are analyzed along a constant xenon flow control line assuming a quasi steady state heat balance.

The flow dependent MCPR limit, MCPR(F), is provided in the CORE OPERATING LIMITS REPORT. The MCPR(F) is independent of the rated flow limit provided in Specification 3.5.K.2 and 3.5.K.3. To verify applicability of this curve to PBAPS, recirculation flow runout events were analyzed with a PBAPS specific model at a typical mid cycle exposure condition. These flow runout events were simulated along the Maximum Extended Load Line Limit rod line to the maximum core flow runout value of 105%. The results of the analyses indicated that application of the MCPR(F) curve will preclude a violation of the MCPR safety limit in the event of a recirculation flow runout. The MCPR(F) curve is cycle independent.

"LEFT BLANK INTENTIONALLY"

6.9.1 Routine Reports (cont'd)

c. Annual Safety/Relief Valve Report

Describe all challenges to the primary coolant system safety and relief valves. Challenges are defined as the automatic opening of the primary coolant safety or relief valves in response to high reactor pressure.

d. Monthly Operating Report

Routine reports of operating statistics and shutdown experience and a narrative summary of the operating experience shall be submitted on a monthly basis. Each report shall be submitted no later than the 15th of the month following the calendar month covered by the report.

e. Core Operating Limits Report

(1) Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each Operating Cycle, or prior to any remaining portion of an Operating Cycle, for the following:

a. The APLHGR for Specification 3.5.I,

b. The MCPR for Specification 3.5.K,

c. The core flow and power adjustment factors for Specification 3.5.K and 3.5.I,

d. The LHGR for Specification 3.5.J,

e. The upscale power biased Rod Block Monitor setpoints and corresponding power levels.

(2) The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents as amended and approved:

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version)

b. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," NEDC-32162P, Revision 1, February, 1993

c. Philadelphia Electric Company Methodologies as described in:

(1) PECo-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184 TO FACILITY OPERATING

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, UNIT 3

DOCKET NO. 50-278

1.0 INTRODUCTION

By letter dated April 1, 1993 (Reference 1), as supplemented by letters dated April 7, July 16 (Reference 2), and August 20, 1993 (Reference 11), the Philadelphia Electric Company, (PECo, the licensee) submitted a request for changes to the Peach Bottom Atomic Power Station (PBAPS), Unit Nos. 2 and 3, Technical Specifications (TSs). The requested changes implement an expanded power-to-flow operating domain supported by the Average Power Range Monitor (APRM), Rod Block Monitor (RBM), Technical Specifications Improvements/Maximum Extended Load Line Limit (ARTS/MELLLA) program and analyses. The April 7, July 16, and August 20, 1993, letters provided additional clarifying information that did not change the initial no significant hazards consideration determination.

The request proposed three fundamental changes: (1) Deletion of the flow-biased APRM scram and rod block trip setpoint setdown requirements; (2) Revision of flow-biased APRM scram and rod block trip equations to expand the power-to-flow operating domain; and, (3) modifications to RBM trip set-points. The changes involve hardware modifications, procedure changes and associated TS changes.

The first change, eliminating APRM setpoint setdown, involves several thermal-hydraulics associated updates made to ensure that the safety limit minimum critical power ratio (MCPR) and fuel thermal-mechanical design bases are not violated. These are:

a. Elimination of reference to k_f, the MCPR flow adjustment factor,
b. Introduction of power and flow dependent adjustments to the maximum average planar linear heat generation rate (MAPLHGR) and MCPR limits,
c. Revision of Core Operating Limits Report (COLR) documentation requirements to include parameters used to determine thermal operating limits, and
d. Removal of the fraction of rated power (FRP) and the maximum fraction of limiting power density (MFLPD) definitions since they are used only in relation to the APRM setpoint setdown.

9309240059 30914
PDR ADOCK 05000278
P PDR

The changes to the APRM scram and rod block trip equations require modification to the APRM rod block electronics. The RBM trip setpoint changes include alterations to the RBM trip logic.

In support of its request, the licensee has submitted the proposed TS changes, a brief explanation of the changes, and a General Electric (GE) topical report (Reference 3) describing in detail the ARTS/MELLLA program for PBAPS.

In order to further support the proposed ARTS/MELLLA changes, the design basis loss-of-coolant accident (LOCA) for PBAPS was analyzed using SAFER/GESTR methodology. The licensee had previously expressed its intention to implement SAFER/GESTR (Reference 4). The GE LOCA analysis topical report (Reference 5) was submitted by letter dated April 7, 1993, and supplemental information was furnished by the licensee in Reference 2.

2.0 EVALUATION

These proposed changes for PBAPS are common for GE boiling water reactors. They have become part of standard operating flexibility options as described in the GE standard application for reactor fuel (Reference 6). The NRC staff has previously reviewed and approved the ARTS/MELLLA changes for several boiling water reactors (BWRs). The methodologies used for the safety analyses justifying the changes and establishment of new operating limits have been previously reviewed and approved by the staff and are documented in Safety Evaluations for Hatch (Reference 9) and Monticello (Reference 10). The new operating region and the APRM and RBM changes proposed for PBAPS are similar to equivalent changes approved previously by the staff in Reference 9 and Reference 10.

Since the submittal for PBAPS includes changes which have become standard and have been well considered for other plants, only a brief description of them is included here. More detailed information is available in Reference 9 and Reference 10. Aspects of the changes or analyses specific to PBAPS are discussed in more depth, although all of the analyses considered previously were reexamined for this review.

2.1 ARTS/MELLLA

2.1.1 Program Description

The MELLLA mode of operation and the ARTS program include the following changes:

a. The operating power to flow map is expanded to allow operation above the rated rod line.

b. A power dependent minimum critical power ratio (MCPR) is implemented to complement the updated RBM system.

c. Power and flow dependent thermal limits are introduced to replace the APRM trip setdown requirement. These are power and flow dependent MAPLHGR and MCPR multiplication factors: MAPLHGR(P), MAPLHGR(F), MCRP(P), and MCPR(F).

d. Power dependent RBM trips replace flow biased trips. RBM inputs are reassigned to improve system characteristics and operability.

e. An updated rod withdrawal error analysis is presented to account for system changes and more closely reflect plant conditions.

f. RBM electronics are updated to produce a trip signal which is a function of the percentage increase from the initial signal.

Fuel performance transient analyses, mechanical evaluation of the reactor internals, structural vibration, LOCA analyses, containment loads evaluations and rod withdrawal error analyses are all required to justify the above ARTS/MELLLA changes. The thermal limits introduced under ARTS program are specified to protect fuel during anticipated operational occurrences (AOOs). The plant thermal limits used in the PBAPS analyses are intended to remain applicable to future reload cycles, including GE fuel designs through GE11 type fuel. Future changes in fuel designs, analytical methods or plant configurations may require confirmatory verification. Plant-specific portions of the ARTS limits for PBAPS were developed based on the Unit 2 Cycle 10 core configuration. Similarity of fuel types and plant configuration makes these ARTS plant-specific limits applicable to both PBAPS Units 2 and 3.

2.1.2 MELLLA Analyses

PBAPS is currently licensed to operate in the extended load line limit (ELLL) region above the rated rod line along the 108% APRM rod block line to the 100% power/87% flow (100P/87F) point on the power-to-flow map. The MELLLA analysis expands the operating domain along the 121% rod line to 100P/75F, allowing rated power operation at any flow between 75% and 100%. This expansion extends the analyzed operating domain to the 121% rod line. The clamped values of the flow biased APRM flux scram and APRM rod block trips will be inserted at 75% flow.

To justify operation of PBAPS in the MELLLA domain, core-wide anticipated operational occurrences (AOO) were analyzed in Reference 3 to determine the limiting MCPR requirement, peak vessel pressure, and stability effects. The events chosen as potentially limiting and re-evaluated in detail are the same events analyzed for previous ARTS/MELLLA submittals reviewed by the staff in Reference 9 and Reference 10. These include; generator load rejection without bypass, turbine trip without bypass, feedwater controller failure, inadvertent high-pressure coolant injection (HPCI), and loss of feedwater heating.

Inputs for analyses corresponding to the 100P/75F condition were developed from the PBAPS Unit 2 Cycle 10 information. The methods used were consistent with the bases of the Cycle 10 reload submittal. The analyses indicate that the generator load rejection event was most limiting for MELLLA conditions.

Further, the results show that for the events examined, the operating limit MCPRs for rated conditions (100P/100F) bound those for MELLLA conditions. Subsequent reload licensing reviews will include examination of cycle-specific data in the MELLLA region.

Vessel overpressure protection was demonstrated by analysis of the main steam line isolation valve (MSIV) closure with flux scram. Initiation of this event from the MELLLA region yielded results that comply with the American Society of Mechanical Engineers Pressure Vessel Code.

As is the case for other BWRs operating under ARTS/MELLLA, PBAPS will maintain compliance with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors." The added operating region does not alter compliance with the stability requirements.

Although minor differences from equivalent MELLLA analyses are included for sensitivity study and future consideration, the analyses presented for PBAPS operation in the MELLLA region yield acceptable results and conform to those previously evaluated by the staff and are acceptable.

2.1.3 ARTS Analyses

To justify operating PBAPS under the ARTS program, analyses of AOOs done in support of MELLLA were used to determine the off-rated power and flow dependent MCPR and MAPLHGR functions. Flow run-out events were also analyzed to assure that the flow dependent MCPR limit is sufficient to prevent violation of the safety limit MCPR (SLMCPR) during recirculation flow increase events. Rod withdrawal error (RWE) analysis was performed to determine setpoints for the updated RBM system. A generic statistical RWE examination was validated for GE11 fuel designs for PBAPS. A LOCA analysis, discussed in Section 2.2 of this evaluation, was performed to verify the flow dependent MAPLHGR limits.

PBAPS specific analyses were performed to confirm the applicability of generic power and flow dependent MCPR and MAPLHGR limits taken from the ARTS data base. These plant limits were selected to remain valid through future reloads using GE11 fuel and currently approved analysis methods. The ARTS analyses used current Cycle 10 inputs along with bounding values for core power, maximum core flow, and reduced feedwater temperature (for the feedwater controller failure analysis).

Overall, the ARTS analyses and the proposed changes to the APRM and RBM systems parallel ARTS submittals for other BWRs which were accepted by the staff (Reference 9 and Reference 10). An important exception was the SAFER/GESTR LOCA analysis, which required additional study, as discussed in Section 2.2.

The ARTS hardware updates proposed for PBAPS are the same as those previously reviewed by the staff. However, the GE report states that the adjustable trip time delay option t_{d2} for the RBM will not be used for PBAPS. Although the option is included with the hardware, sufficient RWE analysis was not performed to allow its use. The suggestion made by the GE report that the t_{d2} setting could be used to bypass the RBM system is counter to previous staff findings (Reference 9) and is not permitted. Any future use of this time delay setting will require the evaluation of further analysis, as discussed in the GE report.

Based on the review of the Peach Bottom specific ARTS analyses and changes described above and comparison to the generic ARTS analyses and changes evaluated in Reference 9 and Reference 10, the staff finds the proposed implementation of system changes associated with the ARTS updates, including the hardware modifications and proposed analytical limits, with the exception of the RBM adjustable trip time delay option described in the previous paragraph, to be acceptable.

2.2 SAFER/GESTR LOCA Analysis

To ensure that the 10 CFR 50.46 LOCA criteria were met by the flow dependent MAPLHGR multipliers, a LOCA analysis was performed using the GE SAFER/GESTR LOCA methodology. Application of SAFER/GESTR to PBAPS was detailed in a GE report (Reference 5) and was evaluated as part of the ARTS/MELLLA application.

Requirements for the use of SAFER/GESTR were established in the Topical Report Evaluation contained in Reference 7. The evaluation includes the stipulation that the plant-specific peak cladding temperature (PCT) versus break size curve match the trend of the generically determined curve. The nominal PCT (PCT_{NOM}) curve is determined using best-estimate values of plant response. This curve establishes the limiting break (normally the large break LOCA) which is used for subsequent calculations. Licensing basis PCT ($PCT_{App\ K}$) is determined for the limiting case. Upper bound PCT (PCT_{UB}) is then determined to confirm the conservatism of the ($PCT_{App\ K}$). The analysis presented in the generic report uses assumptions arising from conditions based on the large break event. The requirements of the Topical Report Evaluation ensure that plant LOCA response does not significantly diverge from the generic LOCA response and possibly invalidate application of SAFER/GESTR LOCA analysis.

Results of break calculations presented in the PBAPS PCT vs. break size plot in Figure 5-1 of Reference 5 are noticeably different from the generic BWR 4 break spectrum (Figure 3.3 of Reference 8). The PBAPS nominal PCT (PCT_{NOM}) for a small break (0.08 ft^2) LOCA in the discharge line is greater than that for the normally limiting large break. The PBAPS report attributes the difference to a lower ADS capacity relative to vessel volume for PBAPS as well as relaxed ECCS parameters used for this particular analysis.

Additional analysis submitted by the licensee (Reference 2) describes a determination of the PCT_{UB} for the small break to validate the $PCT_{App\ K}$ value determined in the original report and ensure that the large break LOCA is the limiting event. The process applied is based on a propagation of errors procedure described in the generic report (Reference 8) and indicates that a margin of 35°F exists between the PCT_{UB} and $PCT_{App\ K}$ for the small break. The analysis, largely based on the generic SAFER/GESTR evaluation for BWR 4 plants, is considered satisfactory and yielded adequate margin to validate the licensing basis $PCT_{App\ K}$.

Thus, the application of SAFER/GESTR to PBAPS is considered acceptable. However, changes to plant operating conditions which could affect LOCA analyses should consider possible impacts on the small break PCT_{UB} calculation to ensure that adequate margin is maintained to the $PCT_{App\ K}$.

As discussed in Reference 3, a determination of containment response under revised assumptions introduced by MELLLA operation was conducted coincident with the LOCA analysis. Short term containment response was examined for MELLLA thermal-hydraulic conditions, including current rated power and feedwater temperature. The results indicated that the maximum drywell airspace temperature would exceed the design value of 281°F for about 10 seconds at the beginning of the event. The peak pressure, however, would remain below the design limit of 56 psig. The PBAPS UFSAR specifies that the maximum drywell temperature is limiting coincident with the maximum internal pressure limit. Since the high temperature is expected to be of short duration and the pressure limit is not approached, the staff agrees that drywell structural integrity is not threatened by MELLLA operation. However, changes in the parameters associated with this analysis, especially core power or feedwater temperature, may necessitate re-evaluation of the containment response to ensure that containment integrity is not threatened.

With the qualifications discussed above, the application of SAFER/GESTR LOCA methodology to PBAPS Units 2 and 3 is acceptable.

2.3 TECHNICAL SPECIFICATIONS

Changes to PBAPS limits and operability requirements in the TS are necessary to implement ARTS/MELLLA. The proposed TS changes follow:

a. Definitions are added to Section 1.0 for down-scale trip setpoint (DTSP), high power trip setpoint (HPTS), intermediate power trip setpoint (ITSP), low power trip setpoint (LTSP), MAPLHGR flow factor, and power dependent MAPLHGR multiplier. The definition for MCPR is revised to include MCPR(F) and MCPR(P).

b. Limiting Safety System Setting Section 2.1 is changed to revise the APRM flux scram and APRM rod block trip setting equations. Numerical values for core flow are removed, and maximum values for the scram and rod block trips are added (120 and 108 percent of rated power, respectively). The setpoint setdown requirements, along with the fraction of rated thermal power (FRP) and maximum fraction of limiting power density (MFLPD) definitions are removed.

c. The Safety Limit Bases Section 1.1 References are updated to include the GE ARTS/MELLLA analysis report (Reference 3).

d. Figure 1.1-1, entitled, "APRM Flow Bias Scram Relationship to Normal Operating Conditions," (the power-to-flow operating map) is revised to include the MELLLA region and the updated APRM limits.

e. Explanations for setpoint setdown are removed from Bases Section 2.1.A, and a reference is included to power and flow dependent MCPR factors located in the COLR.

f. Surveillance Requirement (SR) 4.1.B, to determine MFLPD and employ setpoint setdown specifications, is deleted, as is its explanation in Bases Section 4.1.B.

g. Table 3.1.1, Reactor Protection System (SCRAM) Instrumentation Requirement, incorporates the revised APRM high flux scram equation.

h. Table 3.2.C, Instrumentation That Initiates Control Rod Blocks, is changed to include new APRM upscale, rod block monitor and rod block monitor downscale trip values, and remove mention of setpoint setdown.

i. Note 11 is added to Table 3.2.C to indicate that the values of HTSP, ITSP, LTSP, and DTSP are included in the Core Operating Limits Report (COLR).

j. Surveillance Requirement 4.5.I includes a reference to the COLR for the MAPLHGR(F) and MAPLHGR(P) multipliers.

k. Limiting Condition for Operation (LCO) 3.5.K includes the MCPR(F) and MCPR(P) multipliers and refers to the COLR for their values.

l. Bases Section 3.5.I refers to the COLR for APLHGR values and indicates that the values for MAPFAC(F) and MAPFAC(P) adjustment factors are in the COLR.

m. Bases Sections 3.5.K, 3.5.L and 3.5.M are updated to include the GE ARTS/MELLLA analysis report (Reference 3).

n. Bases Section 4.5.L is rewritten to eliminate the K_f factor and describe the power and flow dependent MCPR limits, MCPR(F) and MCPR(P).

o. Routine Reports Section 6.9.1(e), detailing the contents of the COLR, is updated to list core flow and power adjustment factors and the upscale power biased RBM setpoints. K_f is eliminated, and Reference 3 is included in a list of analytical methods used for core operating limits determination.

Based upon the acceptance of the methods and results of the ARTS/MELLLA for PBAPS as discussed in Section 2 of this evaluation, these TS changes are acceptable.

By letter dated August 20, 1993, the licensee submitted revisions to the Unit 2 TS Pages 4, 20, and 140b, and the Unit 3 TS Pages 4 and 9. The revisions corrected typographical errors contained in the TS pages contained in the April 1, 1993, submittal. The revisions also updated the affected pages to reflect Unit 2 TS Amendment 178 and Unit 3 TS amendments 181 and 183, which were issued subsequent to the licensee's April 1, 1993, application. The revisions are editorial in nature and ensure that the amendment described in this SE accurately reflects previous TS amendments and, therefore, are acceptable.

The licensee's application proposed the ARTS/MELLLA TS changes for both Units 2 and 3. The application requested that the amendments be effective upon completion of the ARTS/MELLA modifications. The licensee plans to implement the ARTS/MELLLA modifications on Unit 3 during Refueling Outage 3R09 scheduled for the fall of 1993 and on Unit 2 during Refueling Outage 2R010 scheduled for the fall of 1994. To prevent confusion between the effective date for the Unit 2 ARTS/MELLLA amendment and the effective date of subsequent amendments that may affect the same TS pages. The staff is issuing the Unit 3 ARTS/MELLLA amendment at this time and will issue the Unit 2 ARTS/MELLLA amendment just prior to Refueling Outage 2R010.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of this amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 39058). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Donoghue

Date: September 14, 1993

REFERENCES

1. Letter from G. A. Hunger, Jr., PECo, to NRC, dated April 1, 1993, Peach Bottom Atomic Power Station, Units 2 and 3, Technical Specification Change Request 93-01.

2. Letter from G. A. Hunger, Jr., PECo, to NRC, dated July 16, 1993, Peach Bottom Atomic Power Station, Units 2 and 3, Response to Request for Additional Information on SAFER/GESTR LOCA Methodology.

3. NEDC-32162P, Revision 1, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," February 1993, (General Electric proprietary information).

4. Letter from G. A. Hunger, Jr., PECo, to NRC, dated March 18, 1993, Peach Bottom Atomic Power Station, Units 2 and 3, Adoption of SAFER/GESTR LOCA Methodology.

5. NEDC-32163P, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993, (General Electric proprietary information).

6. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel," April 1991 (General Electric proprietary information).

7. Letter from C. O. Thomas, NRC, to J. F. Quirk, GE, dated June 1, 1984, Accepting GE Topical Report NEDE-23785 Revision 1, Volume III(P), "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident."

8. NEDE-23785-1-PA, The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, Revision 1, October 1984, (General Electric proprietary information).

9. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 39 to Facility Operating License No. NPF-5, Edwin I. Hatch Nuclear Plant, Unit No. 2, Docket No. 50-366, dated July 13, 1984

10. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 29 to Facility Operating License No. DPR-22, Monticello Nuclear Generating Plant, Docket No. 50-263, dated November 16, 1984.

11. Letter from G. A. Hunger, Jr., PECo, to NRC, dated August 20, 1993, Peach Bottom Atomic Power Station, Units 2 and 3, Revised Technical Specification Change Request.