Docket No. 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 Amendment NO. 33 to Facility
Operating License No. DPR-56 for the Peach Bottom Atomic Power Station,
Unit No. 3. The amendment consists of changes to the Technical Spec-
ifications and is in response to your request dated November 17, 1976,
and supplement thereto dated February 7, 1977.

The amendment authorizes operation of the Peach Bottom Atomic Power
Station, Unit No. 3 with (1) up to 188 General Electric (GE) 8x8 reload
fuel bundles, (2) one Pressurized Test Assembly (PTA), (3) one Fast
Scram Control Rod Drive (FSCRD), (4) holes drilled in the lower tie
plate of all reload fuel bundles to provide an alternate bypass flow
path, and (5) modifications to the Rod Sequence Control System (RSCS).
The amendment also deletes obsolete paragraphs from the license.

Certain changes and additions to the Technical Specifications proposed
in your application dated November 17, 1977 were necessary. These
changes have been discussed with your staff and they do not object.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are
also enclosed.

                            Sincerely,

                            Original signed by

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

Enclosures and ccs:
See page 2

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE ➤	ORB #3	ORB #3	OELD	ORB #3	AD:DOR KRG	
SURNAME ➤	*CParrish	EVerdery	CUTCHIN	GLear	KRGoller	
DATE ➤	3/ /77	3/21/77	3/21/77	3/23/77	3/23/77	

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

Enclosures:
1. Amendment No.
2. Safety Evaluation
3. Federal Register Notice

cc:
See next page

	OFFICE ►	ORB #3	ORB #3	OELD	AD:DOR	ORB #3	
	SURNAME ►	EVerdery:mj	Parrish		KRGoller	GLear	
	DATE ►	2/ /77	2/25/77	/ /77	/ /77	/ /77	

Form AEC-318 (Rev. 9-53) AECM 0240   ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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

March 23, 1977

Docket No. 50-278

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

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Operating Reactors Branch #3
Division of Operating Reactors

Enclosures and ccs:
See page 2

Philadelphia Electric Company   - 2 -

Enclosures:
1. Amendment No. 33
2. Safety Evaluation
3. FEDERAL REGISTER Notice

cc:

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Martin Memorial Library
159 E. Market Street
York, Pennsylvania 17401

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. 33
License No. 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 November 17, 1976, as supplemented by filing dated February 7,
1977, complies with the standards and requirements of the Atomic
Energy Act of 1954, as amended (the Act), and the Commission's
rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-56 is hereby amended to read as follows, and paragraphs 2.C(3) and 2.C(4) are deleted in their entirety:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 33, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller, Assistant Director
  for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 23, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness. No changes were made on the overleaf pages which are identified below by an asterisk:

REMOVE	INSERT
10	10
11	11
14	14
15	15
15a	15a
15b	15b
17	17
18	18
20	20
33	33
35	35
37	37
38	38*
39	39*
40	40
54	54
73	73
74	74
101	101
102	102
109	109
110	110
111	111
119	119
120	120*
133a	133a
133b	133b
140	140
140a	140a
140b	140b
140c	140c
140d	140d
140e	140e

REMOVE	INSERT
141a	141a
	142e (new)
	142f (new)
157	157
158	158*
241	241
242	242*

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A (cont'd)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified to the more limiting (lower) of the two values determined by the following:

a. $S \leq (0.66 \ W + 54\%) \dfrac{2.66}{MTPF}$ for 7x7 fuel

b. $S \leq (0.66 \ W + 54\%) \dfrac{2.48}{MTPF}$ for 8x8 fuel

MTPF = The value of the existing maximum total peaking factor

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

Design value of A = 2.66 for 7x7 fuel and 2.48 for 8x8 fuel.

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

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

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

PBI.PS

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

SAFETY LIMIT

B. Core Thermal Power Limit
(Reactor Pressure $\leq$ 800 psia)

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

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

LIMITING SAFETY SYSTEM SETTING

B. APRM Rod Block Trip Setting

$$S_{RB} \leq 0.66\ W + 42\%$$

where:

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

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr).

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified to the more limiting of the 2 values determined by the following:

1. $S_{RB} \leq (0.66\ W + 42\%) \dfrac{2.66}{\text{MTPF for 7x7 fuel}}$

2. $S_{RB} \leq (0.66\ W + 42\%) \dfrac{2.48}{\text{MTPF for 8x8 fuel}}$

MTPF = The value of the existing maximum total peaking factor

Design value of A = 2.66 for 7x7 fuel and 2.48 for 8x8 fuel.

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

D. Scram--turbine stop $\leq$10 percent valve closure

E. Scram--turbine control valve fast closure on loss of control oil pressure.

500 $<$ P $<$ 850 psig.

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 Figure 4-2 of Reference 3.

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

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

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

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 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.

- 14 -

The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

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

3. General Electric BWR Generic Reload Licensing Application for 8x8 fuel, Revision 1, Supplement 4, April 1976 (NEDO-20360).

Amendment No. 14, 33

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	8.7
Bypass void effect on TIP	3.58 (core midplane) 4.08 (core exit)
R Factor	1.6
Critical Power	3.6

- 15a -

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	1.080 (7 x 7)
	1.095 (8 x 8)
	1.045 (8 x 8 PTA)

2.1   BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Peach Bottom Atomic Power Station Units have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 3440 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7.1 of the FSAR. In addition, 3293 MWt is the licensed maximum power level of each Peach Bottom Atomic Power Station Unit, and this represents the maximum steady state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in NEDO 10802.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. Active coolant flow is equal to 88% of total core flow. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady state condition.

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

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

In summary:

    i.   The abnormal operational transients were analyzed to a power
         level of 3440 MWt

Amendment No. 18, 33

ii. The licensed maximum power level is 3293 MWt.

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

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

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

A. Neutron Flux Scram

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

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

The scram trip setting must be adjusted to assure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1. when the maximum total peaking factor is greater than the design value of A for each class of fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR $>$ 1.06 when the transient is initiated from a MCPR greater than the operating limit MCPR given in Specification 3.5.K.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate 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

-18-

2.1 BASES (Cont'd.)

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.06. This rod block trip setting, which automatically varies with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the design value of A for each class of fuel, thus preserving the APRM rod block safety margin.

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

The set point for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.06 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 31 in. below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

E. Turbine Control Valve Scram

The turbine control valve fast closure scram anticipates the pressure, neutron flux and heat flux increase that could result from fast closure of the turbine control valves due to a load rejection exceeding the capacity of the bypass valves or a failure in the hydraulic control system which results in a loss of oil pressure. This scram is initiated from pressure switches in the hydraulic control system which sense loss of oil pressure due to the opening of the fast acting solenoid valves or a failure in the hydraulic control

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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 are 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 the Peach Bottom units. 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 1279 psig for Peach Bottom Unit 3 if a neutron flux scram is assumed. This results in a 96 psig margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient (turbine trip with by-pass valve failure to open) assuming a turbine trip scram is presented in Section 6.3 of NEDO-21363 for Peach Bottom Unit 3. This analysis shows that the 11 safety/relief valves limit pressure at the safety valves to 27 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.

-33-

Unit 3

PBAPS

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The designed system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 100 milli-seconds.

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. Daily during reactor power operation, the peak heat flux and peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specification 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds 2.66 for 7x7 fuel or 2.48 for 8x8 fuel.

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

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

Table 3.1.1 (Cont'd.)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

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

T.S. Change #

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable sensor channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place mode switch in the startup position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% of rated.

2. Permissible to bypass, in refuel and shutdown positions of the reactor mode switch.

3. Bypassed when reactor pressure is < 600 psig.

4. Bypassed when turbine first stage pressure is less than 220 psig or less than 30% of rated.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

8. Not required to be operable when primary containment integrity is not required.

9. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

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NOTES FOR TABLE 3.1.1   (cont'd)

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

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

12.   W is the recirculation loop flow in percent of design. W is equal
      to 100 for core flow of 102.5 million pounds/hour or greater.
      Trip level setting is in percent of rated power (3293 MWt).
      A =   2.66 for 7x7 fuel and 2.48 for 8x8 fuel. MTPF is the value
      of the existing maximum total peaking factor.

13.   See Section 2.1.A.1.

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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; e.e., in the period of a month a maximum drift of 0.4% could occur, thus providing for adequate margin.

For the APRM system, 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 maximum total peaking factor is checked once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check is adequate.

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

TABLE 3.2 C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

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

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 is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt). Refer to Limiting Safety Settings for variation with peaking factors. A ± 2.66 for 7x7 fuel and 2.48 for 8x8 fuel. MTPF is the value of the existing maximum total peaking factor.

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

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

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

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

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

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

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Amendment No. 33

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.B (cont'd.)

B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3. a. Whenever the reactor is in the startup or run modes below 30% rated power the Rod Sequence Control System shall be operable, that is no position switches shall be bypassed except as permitted in 3.3.A.2d, except during shut down margin testing.

   b. Whenever the reactor is in the startup or run modes below 25% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

4.3.B (cont'd.)

b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

c. During each refueling outage and after control rod maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3. a. The "sequence mode" of RSCS shall be demonstrated to be operable by attempting to select and move a rod in each of the out-of-sequence groups:

   1. Prior to the start of control rod withdrawal for a reactor start-up.

   2. As soon as the "sequence mode" of RSCS is automatically initiated during rod insertion when reducing power.

   The "group notch" mode of RSCS shall be demonstrated to be operable by attempting to move a control rod more than one notch in the first programmed group:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.B (cont'd.)

    c. (deleted)

    d. If Specifications 3.3.B.3.a through c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 25% rated power, it shall be brought to a shutdown condition immediately.

4.3.B (cont'd.)

    1. After reaching 50% rod density on a reactor start-up.

    2. Prior to attaining 25% of rated power during rod insertion when reducing power.

    b. Prior to the start of control rod withdrawal towards criticality and prior to attaining 25% of rated power during rod insertion at shutdown, the Rod Worth Minimizer (RWM) shall be demonstrated to be operable by the following checks:

        1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.

        2. The RWM computer on line diagnostic test shall be successfully performed.

        3. Prior to the start of control rod withdrawal only, proper annunciation of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified.

        4. The rod block function of the RWM shall be verified by withdrawing the first rod during start-up only as an out-of-sequence control rod no more than to the block point.

    c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified and recorded.

3.3 and 4.3 BASES (cont'd.)

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

3. The Rod Worth Minimizer (RWM) and sequence mode of the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. The group notch mode of the RSCS restricts movement of rods assigned to each group to notch withdrawal and insertion. All patterns associated with these restrictions have the characteristic that, assuming the worst single deviation from the restrictions, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in the peak enthalpy of any pellet exceeding 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Sections 3.6.6, 14.6.2 and 7.16.3.3 of the FSAR, NEDO-10527 and supplements thereto, and NEDO-21363.

In performing the function described above, the RWM and RSCS are not needed to impose any restrictions at core power levels in excess of 20 percent of rated power; however, Technical Specifications require the use of the RWM below 25% rated power, and the RSCS below 30% of rated power. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at a power level greater than 20 percent, regardless of the rod pattern. This is true for all normal and abnormal patterns, including those which maximize individual control rod worth.

Up to 50% rod density (either sequence A or B control rods fully withdrawn and the other sequence fully inserted), the sequence mode of the RSCS restricts the maximum positive reactivity which can be added to the core due to a dropped control rod by control rod selection. Between 50% rod density and 30% of rated power, the group

3.3 and 4.3 BASES (Con't)

notch mode of the RSCS restricts the reactivity worth by requiring movement of control rods such that rods assigned to each group are kept within one notch of each other.

The Rod Worth Minimizer and the sequence mode of the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted and the group notch mode of RSCS requires notch movement of rods; i.e., the systems limit operator deviations from planned control rod movement. They serve as a backup to procedural control of control rod movement, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls. The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At power levels below 20 percent of rated these devices force adherence to acceptable rod patterns. Above 20 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20 percent of rated power are imposed by power distribution requirements as defined in Section 3.5/4.5 of the Technical Specifications.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur beings at or above the initial value of 10^{-8} of rated power used in analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/ or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3.3 and 4.3   BASES   (Cont'd)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.06. Analysis of the limiting power transients shows that the negative reactivity rates resulting from the scram (Ref. NEDO-21363) with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than 1.06.

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

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

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

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

3.4 BASES

STANDBY LIQUID CONTROL SYSTEM

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

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 600 ppm of boron in the reactor core in less than 125 minutes. The 600 ppm concentration in the reactor core will bring the reactor from full power to at least a 3.0% Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak. .

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

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

3.4   BASES (Cont'd.)

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

    The solution is kept at least $10^{\circ}F$ above the saturation
    temperature to guard against boron precipitation. The
    margin is included in Figure 3.4.2.

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

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

    A minimum quantity of 2800 gallons of solution having a
    19.3 percent sodium pentaborate concentration, or the
    equivalent as shown in Figure 3.4.1, is required to
    meet this shutdown requirement. For the minimum re-
    quired pumping rate of 39 gpm, the maximum net storage
    volume of the boron solution is established as 4850
    gallons.

Amendment No. 33

LIMITING CONDITION FOR OPERTION	SURVEILLANCE REQUIREMENT

3.5.I Average Planar LHGR

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

3.5.J Local LHGR

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

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

$LHGR_d$ = Design LHGR
= 18.5 kW/ft for 7x7 fuel
13.4 kW/ft for 8x8 fuel

$(\Delta P/P)_{max}$ = Maximum power spiking penalty
= 0.026 for 7x7 fuel
= 0.022 for 8x8 fuel

LT = Total core length = 12.167 ft.
Unit 3

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 checked daily during reactor operation at $\geq 25\%$ rated thermal power.

4.5.J. Local LHGR

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

LIMITING CONDITION OF OPERATION   SURVEILLANCE REQUIREMENT

3.5.J. Local LHGR (Cont'd)

If at any time during operation
it is determined by normal surveil-
ance 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 return-
ed 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 with-
in 36 hours unless LHGR is return-
ed to within limits during this
period.Surveillance and correspond-
ing action shall continue until
reactor operation is within the
prescribed limits.

3.5.K. Minimum Critical Power Ratio (MCPR)   4.5.K. Minimum Critical Power Ratio (MCPR)

During power operation, MCPR   MCPR shall be checked daily
shall be $\geq$ 1.32 for 7x7 fuel,   during reactor power operation
$\geq$1.40 for 8x8 fuel, and $\geq$1.41   at $\geq$ 25% rated thermal power.
for 8x8 PTA fuel, at rated power
and flow. For core flows other
than rated the MCPR shall be
$\geq$1.32 times k_f for 7x7 fuel,
$\geq$1.40 times k_f for 8x8 fuel, and
$\geq$1.41 times k_f for 8x8 PTA fuel,
where k_f is as shown in Figure
3.5.1-E. If at any time during
operation it is determined by nor-
mal surveillance that the limit-
ing value for MCPR is being
exceeded, action shall be in-
itiated within one (1) hour to
restore MCPR to within prescribed
limits. If the MCPR is not re-
turned 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 with-
in limits during this period.
Surveillance and corresponding
action shall continue until
reactor operation is within
prescribed limits.

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·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 equip-
ment 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-C , D , F and G.

The calculational procedure used to establish the APLHGR shown
on Figures 3.5.1.C, D, F and G is based on a loss-of-coolant
accident analysis. The analysis was performed using General
Electric (GE) calculational models which are consistent with
the requirements of Appendix K to 10 CFR Part 50. A complete
discussion of each code employed in the analysis is presented
in Reference 4. Differences in the Peach Bottom analysis as
compared to previous analyses performed with Reference 4 are:
(1) The analyses assumes a fuel assembly planar power con-
sistent with 102% of the MAPLHGR shown in Figure 3.5.1-C, D,
F and G; (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 5,
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 References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≳ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

Densification analyses for 8x8 fuel are presented in Section 3.3.4.3 and Appendix B of Reference 7.

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 analyses of the abnormal operational transients presented in References 6 & 8. 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 transients which determine the required steady state MCPR limits is turbine trip with failure of the bypass valves. This transient yields the largest △CPR for each class of fuel. When added to the safety limit MCPR of 1.06, the required minimum operating limit MCPR's of specification 3.5.K are obtained.

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

(1) The core is operating at full power in the xenon-free
     condition.

(2) The highest worth control rod is assumed to be fully
     inserted.

(3) The analysis is performed for the most reactive point in the
     cycle.

(4) The control rods are assumed to be the worst possible pattern
     without exceeding thermal limits.

(5) A bundle in the vicinity of the highest worth control rod is
     assumed to be operating at the maximum allowable linear
     heat generation rate.

(6) A bundle in the vicinity of the highest worth control rod is
     assumed to be operating at the minimum allowable critical
     power ratio.

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

The analysis of the rod withdrawal error for Peach Bottom Unit 3
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.

A brief summary of the analytical method used to determine the
nuclear characteristics is given in Section 5.3 of Reference 7.

Analysis of the abnormal operational transients is presented in
Section 6.3 of Reference 6 and in Reference 8. Input data and
operating conditions used in this analysis are shown in Table 6-1
of Reference 6.

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

In the event that the calculated value of APLHGR, LHGR and 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 indicated value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

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.

L. __Reference__

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

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

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

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

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

6. "General Electric Boiling Water Reactor Reload No. 1 License Amendment for Peach Bottom Atomic Power Station Unit 3," NEDO-21363, November 1976.

7. General Electric BWR Generic Reload Application for 8x8 fuel, NEDO-20360, Revision 1, Supplement 4, April 1976.

8. "Pressurized Test Assembly Supplemental Information for Reload 1 Licensing Amendment for Peach Bottom Atomic Power Station Unit 3," NEDO-21363, Supplement 1, November 1976.

TABLE 3.5-1
PEACH BOTTOM Unit 3 (Alternate Flow Path)
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
Reload No. 1	8 x 8	13.4	1.4	1.17

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

*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 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

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 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
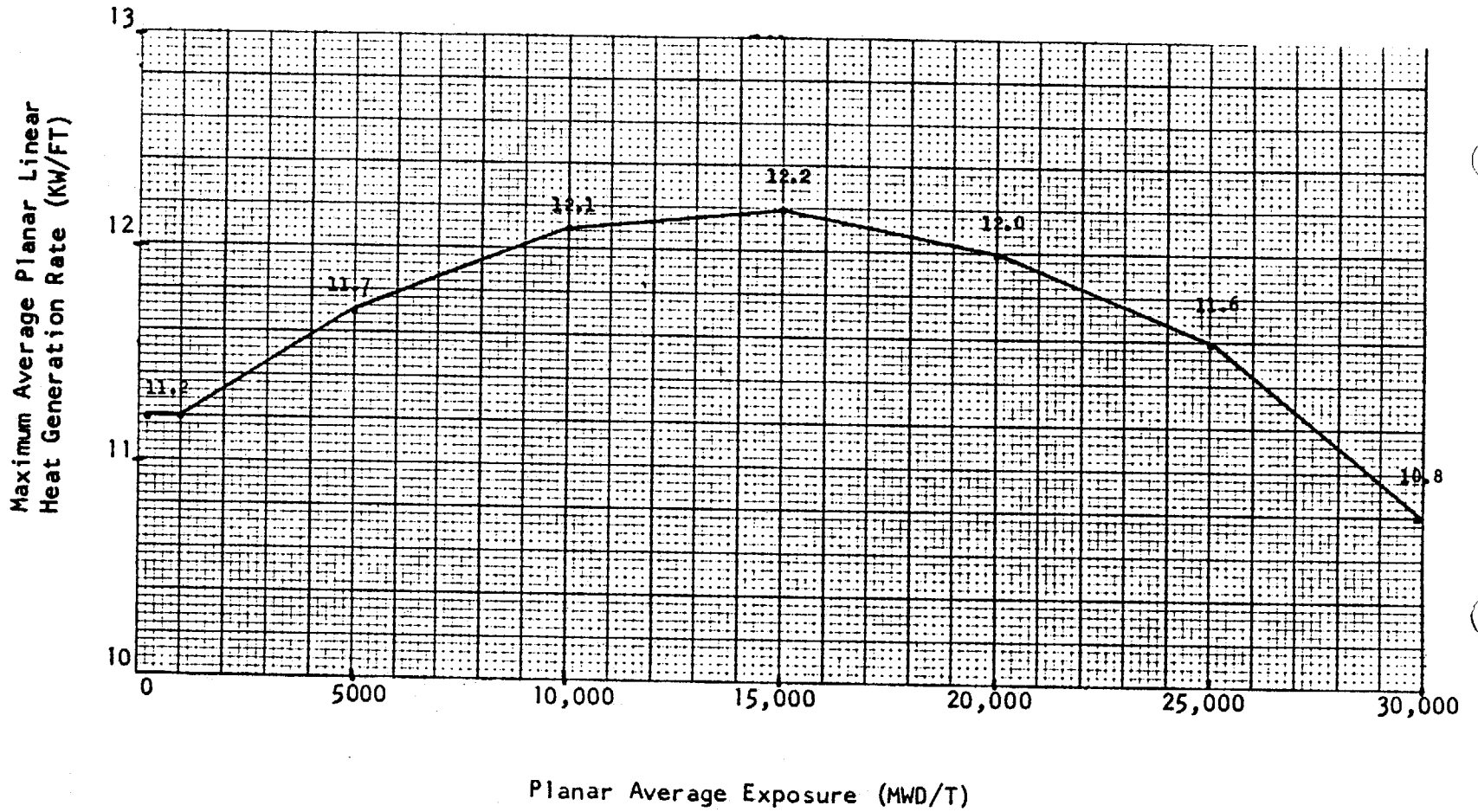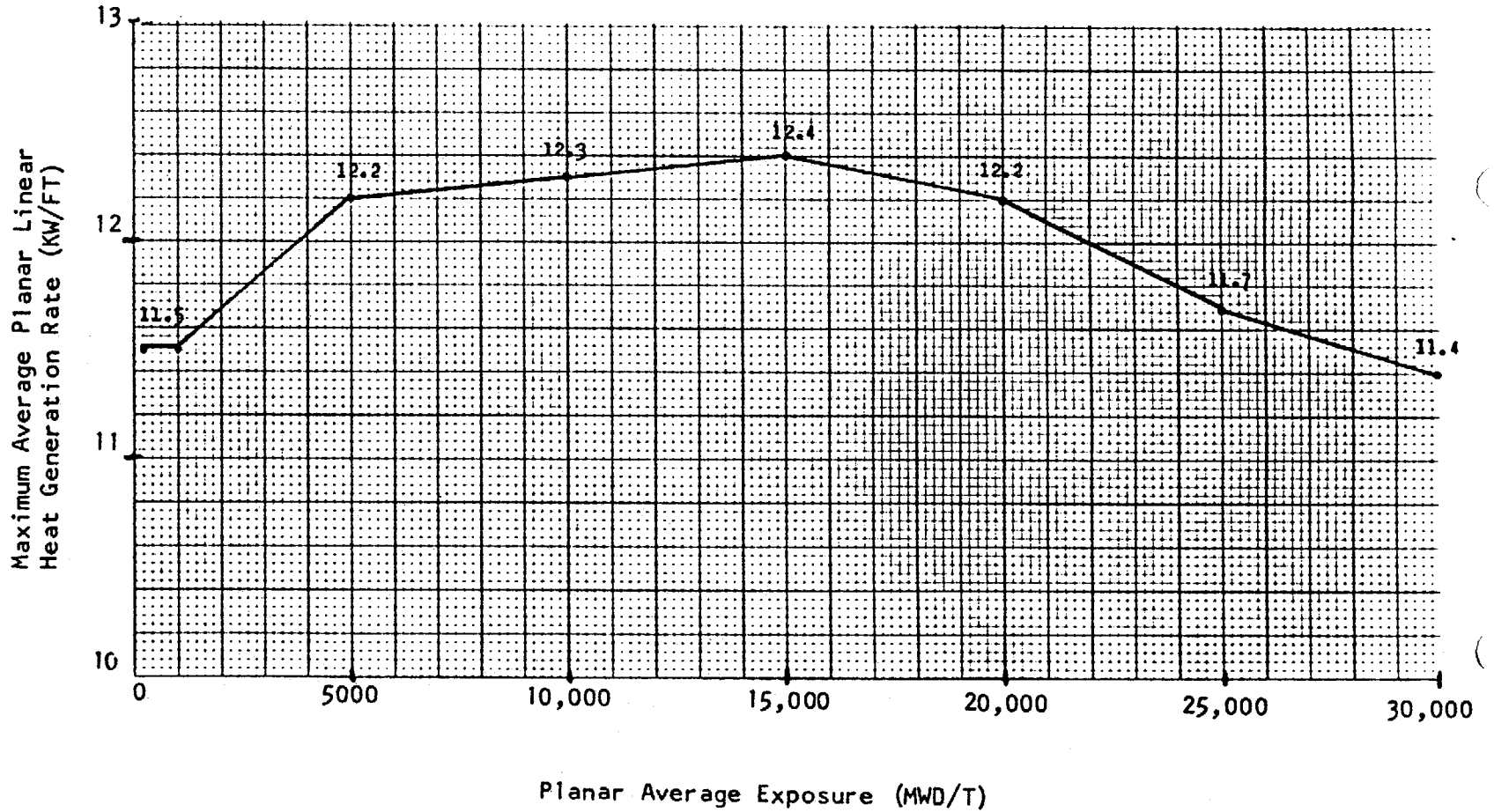
PEACH BOTTOM UNIT 3

RELOAD 1

8x8 Fuel

Planar Average Exposure (MWD/T)

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

PEACH BOTTOM UNIT 3

RELOAD 1

8x8 LTA Fuel and
8x8 PTA Fuel

Figure 3.5.1.G   Maximum Average Planar Linear Heat Generation Versus
Planar Average Exposure

3.6.D & 4.6.D BASES

Safety and Relief Valves

The safety and relief valves are required to be operable
above the pressure (122 psig) at which the core spray system
is not designed to deliver full flow. The pressure relief
system for each unit at the Peach Bottom APS has
been sized to meet two design bases. First, the total safe-
ty/relief valve capacity has been established to meet the
overpressure protection criteria of the ASME code. Se-
cond, the distribution of this required capacity between
safety valves and relief valves has been set to meet de-
sign 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 re-
jections.

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

Eleven safety/relief valves and two safety valves have been
installed on Peach Bottom Unit 3 with a total capacity of
of 78.0% of rated steam flow. The analysis of the worst
overpressure transient, (3 second closure of all main steam
line isolation valves) neglecting the direct scram (valve
position scram) results in a maximum vessel pressure of
1279 psig if a neutron flux scram is assumed. This results
in a 96 psig margin to the code allowable overpressure
limit of 1375 psig.

To meet the power generation design basis, the total safety/
relief capacity of 78.1% has been divided into 64.5% relief
(11 valves) and 13.6% safety (2 valves). The analysis of
the plant isolation transient (Turbine trip with bypass
valve failure to open) assuming a turbine trip scram is
presented in NEDO-21363 for Peach Bottom Unit 3. This
analysis shows that the 11 relief valves limit pressure at
the safety valves to 27 psig below the setting of the safety
valves. Therefore, the safety valves will not open.

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

3.6.D & 4.6.D BASES (Cont'd.)

operating cycle to ensure that their set points are within
the + 1 percent tolerance. Additionally, once per opera-
ting cycle, each relief valve is tested manually with
reactor pressure above 100 psig to demonstrate its ability
to pass steam.

The requirements established above apply when the nuclear
system can be pressurized above ambient conditions. These
requirements are applicable at nuclear system pressures
below normal operating pressures because abnormal opera-
tional transients could possibly start at these conditions
such that eventual overpressure relief would be needed.
However, these transients are much less severe, in terms
of pressure, than those starting at rated conditions. The
valves need not be functional when the vessel head is re-
moved, since the nuclear system cannot be pressurized.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

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

5.2 REACTOR

A. The core shall consist of not more than 764 fuel assemblies. 7 x 7 fuel assemblies shall contain 49 fuel rods and 8 x 8 fuel assemblies shall contain 62 or 63 fuel rods. The core shall consist of not more than 188 8x8 fuel assemblies.

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

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

D. One Fast Scram Control Rod Drive may be utilized for up to two full fuel cycles.

5.3 REACTOR VESSEL

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

5.4 CONTAINMENT

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

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

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

5.5 FUEL STORAGE

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.90.

C. Spent fuel shall only be stored in the spent fuel pool in a vertical orientation in approved storage racks.

5.6 SEISMIC DESIGN

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

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 33 TO FACILITY LICENSE NO. DPR-56

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 3

DOCKET NO. 50-278

1.0 Introduction

By letter dated November 17, 1976 and supplemented by letter dated February 7, 1977, Philadelphia Electric Company (PECO) requested an amendment to Facility Operating License No. DPR-56. The amendment would modify the Technical Specifications for the Peach Bottom Atomic Power Station Unit No. 3 to permit operation of the facility with (1) up to 188 General Electric (GE) 8x8 reload fuel bundles, (2) one Pressurized Test Assembly, (PTA), (3) one Fast Scram Control Rod Drive (FSCRD), (4) holes drilled in the lower tie plate of all reload fuel bundles to provide an alternate bypass flow path, and (5) modifications to the Rod Sequence Control System (RSCS).

2.0 Background

The licensee has proposed to reload the Peach Bottom Unit No. 3 reactor with 188 G.E. 8x8 fuel bundles with 100 mil wall thickness channels and an average enrichment of 2.74 wt% U-235. The lower tie plates in the reload assemblies will be drilled with two 9/32 inch holes in order to provide an alternate bypass flow path. The licensee will include, with this reload, the use of one (1) lead Pressurized Test Assembly (PTA). Also, during this refueling outage, the licensee has proposed to (1) replace one control rod drive with a Fast Scram Control Rod Drive (FSCRD) and (2) modify the RSCS to replace the "group C" mode with a "group notch" mode.

The documentation submitted for our review in support of the proposed reload includes: the G.E. BWR Reload 1 licensing submittal for Peach Bottom for the 8x8 fuel [1], information on the PTA [2] and the FSCRD [3], and other supplemental information [11]. Proposed Technical Specification changes [4] were also submitted.

3.0 EVALUATION

3.1 Nuclear Characteristics

The reload information presented in the licensing submittal closely follows
the guidelines of Appendix A of the Generic Reload Licensing Application,
NEDO-20360[5]. The NRC staff has reviewed the contents of the latest
supplements to this generic topical report and has found them acceptable
for use in connection with the information submitted by the licensee in
support of his application for this core reload. The information contained
in NEDO-20360 through supplement No. 4 does not alter the conclusions and
approvals as stated in Reference 6.

A total of one-hundred-eighty-eight (188) reload fuel bundles with an average
U-235 enrichment of 2.74 wt% will be loaded throughout the core. Sixty-eight
of the reload fuel bundles contain fuel rods having a high gadolinia content
(8D274H) and 120 bundles contain rods having a low gadolinia content (8D274L).
In addition, one pre-pressurized test assembly (PTA) will be loaded in place
of a standard 8x8 reload assembly and is expected to operate for at least four
full reactor cycles pending further NRC review and approval. Twenty-four of
the fuel rods in the PTA have been pre-pressurized with helium to an initial
cold fill pressure of 3 atmospheres. The PTA has a total fueled length of
150 inches, compared to 146 inches for a standard 8x8 bundle. The top six
inches and bottom four inches of the PTA contain fuel pellets with natural
uranium enrichment. The remaining central 140 inches contain pellets of 2.74
wt% enrichment. The core contains a total of 764 bundles. Thus, about 25
percent of the fuel bundles are being replaced for the reload.

The high gadolinia reload bundles are scatter-loaded toward the center of the
core while the low gadolinia reload bundles are loaded in the outer portions
of the core. The PTA has a low gadolinia content and is therefore loaded
in the outer portion of the core.

The information in Reference 1 indicates that the nuclear characteristics
of the Cycle 2 core, consisting of both the reload 8x8 fuel (including the
PTA) and the once burned 7x7 fuel, are very similar to the previous core.
Typical nuclear characteristics of the reloaded core are given in Table 5-1
of Reference 1. The void coefficient of reactivity at a core average void
content of 36.3 percent varies from -14.12×10^{-4} to -13.33×10^{-4} $\Delta k/K/\%V$.
The Doppler coefficient, at a fuel temperature of 650°C, varies from -1.221×10^{-5}
to -1.119×10^{-5} $\Delta k/K/°F$. Thus based on our review of the information presented
in the Peach Bottom Unit No. 3 licensing submittal and the generic 8x8 reload
topical report it is concluded that fuel temperature and void dependent
behavior of the reconstituted core will not differ significantly from that
which has been previously reported for Cycle 1 of the Peach Bottom Unit No. 3
reactor.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.38% ΔK subcritical in the most reactive operating state with the single most reactive control rod fully withdrawn and with all other rods fully inserted. For Cycle 2 the minimum shutdown margin is 0.0137 ΔK (1.37% ΔK subcritical for the most reactive operating state).

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by at least 1.10 ΔK (30.%) at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for Peach Bottom Unit No. 3 is that the effective multiplication factor, K_{eff}, of the fuel as stored in the fuel storage racks is equal to or less than 0.90 for normal storage conditions. This requirement is met if the uncontrolled infinite multiplication factor, K_∞, of a fuel bundle in the reactor core configuration is less than or equal to 1.30. Reload fuel bundle types 8D274H and 8D274L at the peak reactivity point have a maximum K_∞ of 1.216 and 1.238, respectively. Both fuel types, therefore, meet the Technical Specifications fuel storage subcriticality requirements.

The Cycle 2 exposure dependent scram reactivity curves used for the analysis of the abnormal operating transients are shown in Figures 6.6a through c of reference 1. These scram curves include a design conservatism factor of 0.8.

3.2 Mechanical Design

The two types of Reload 1 fuel assemblies have the same mechanical design and fuel bundle enrichments as the 8D274L and 8D274H fuel assemblies described in the 8x8 generic reload topical report [5], except for the channel wall thickness and the drilled bypass flow holes in the fuel bundle lower tie plate. The channel wall thickness for the reload fuel assemblies is nominally 0.100 inches whereas the standard product line fuel channels discussed in Reference 5 have a nomimal 0.080 inch wall thickness. The two holes that are drilled through the lower tie plates of the reload bundles, to provide bypass flow, are 9/32 inch in diameter.

The PTA is similar in outline dimensions to the standard 8x8 reload fuel. The PTA, however, contains two water rods (compared to a single water rod in the standard reload fuel bundle) with each fuel rod containing a 150-inch pellet stack length (compared to 146 inches for the standard reload bundle) of which a total of 10 inches are natural uranium. Twenty-four rods in the PTA have been pre-pressurized with helium to an initial pressure of 3 atmospheres at room temperature. Pre-pressurization with helium is

beneficial in that it has the effect of reducing stored energy via improved pellet-to-clad thermal conductivity.

Sufficient plenum volume has been provided above the fuel stack to assure that the increase in internal pressure caused by fission gas release, when combined with the other mechanical design basis, loads does not cause the stress intensity limits [5] to be exceeded.

The generic reload topical report [5] which is under review, has been found acceptable for use in connection with BWR reactors containing 8x8 reload fuel, when supplemented with information required by the Staff Status Report [6] on our evaluation of the GE generic topical report. The thicker (0.100 inch wall thickness) channels will result in greater margins for withstanding operating loads. On the basis of our review of the generic reload topical report and the reload submittal, it is concluded that the Reload 1 fuel for Peach Bottom Unit No. 3 has an acceptable mechanical design.

3.3 Thermal-Hydraulics

The generic 8x8 reload topical report [5] and the General Electric Thermal Analysis Basis (GETAB)[7] are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of GETAB, based on the Minimum Critical Power Ratio (MCPR) concept, was used to establish the:

(1) fuel cladding integrity safety limit,

(2) limiting condition for operation such that the safety limit is not exceeded for normal operation and abnormal operational transients, and

(3) limiting conditions for operation such that the initial conditions assumed in the accident analyses are satisfied.

The Staff has reviewed [8] the GETAB report and has found it acceptable for use in the above applications for 8x8 and 7x7 fuel assemblies.

The Peach Bottom Unit No. 3, Cycle 2 thermal limits based on the GETAB report and the plant specific information provided by the licensee have been reviewed. The Staff evaluation of these limits is reported herein.

3.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR (SLMCPR) is 1.06 for both
7x7 and 8x8 fuel types. This SLMCPR, based of the GETAB statistical
analysis, assures that 99.9% of the fuel rods in the core are not expected
to experience transition boiling for abnormal operational transients. The
uncertainties in the core operating parameters, plant system operating
parameters and the GEXL corrolation (Reference 1, Table 4-1) when combined
with the design relative bundle power histogram for the core, form the
basis of the GETAB statistical determination of the safety limit MCPR.
The tabulated list of uncertainties for Peach Bottom Unit No. 3 during
Cycle 2 are the same or more conservative than those used in GETAB (revision
to Table IV-1 of NEDO-10958[9]). The Cycle 2 analysis includes an increase
in the "TIP Reading" standard deviation from 6.3 to 8.7 and a decrease in
the "Bypass Void Effect of TIP" standard deviation from 4.08 (core mid-
plane) and 5.21 (core exit) to 3.58 (core midplane) and 4.08 (core exit).
These changes in uncertainties result from plugging of the bypass flow
holes in the core plate. The generic core selected for the GETAB statis-
tical analysis is a typical 251/764 core and therefore applies to Peach
Bottom Unit No. 3. The generic GETAB statistical analysis results are
conservative since the core bundle power histogram used for the GETAB
application has more high power bundles than the most adverse bundle power
distribution expected at any time during the second cycle of operation of
Peach Bottom Unit No. 3. This results in a conservative value of the
safety limit MCPR which satisfies the 99.9% criterion.

It is concluded that the proposed fuel integrity SLMCPR of 1.06 is accept-
able for both the 7x7 and reload 8x8 fuel in the Peach Bottom Unit No. 3
reactor core during Cycle 2.

3.3.2 Operating Limit MCPR

Various transient events will reduce the operating MCPR. To assure that
the fuel cladding safety limit MCPR of 1.06 is not violated during antici-
pated abnormal operational transients, the most limiting transients have
been analyzed to determine which results in the largest reduction in the
critical power ratio (i.e. ΔMCPR). The licensee has submitted [1,2] the
results of analyses of those transients which produce a significant
decrease in MCPR. The types of anticipated abnormal operational transients
evaluated were reactor pressure increase, feedwater temperature decrease,
coolant flow increase, etc. The most limiting abnormal operational tran-
sient from rated conditions in these categories for the 7x7, 8x8 and PTA
was the turbine trip with failure of the bypass valves. The licensee
analyzed this transient at three Cycle 2 exposures, corresponding to end
of Cycle 2 (EOC-2), EOC-2 Minus 750 MWd/t, and EOC-2 Minus 1500 MWd/t, in
order to conservatively determine the largest ΔMCPR for this transient

during the fuel cycle. The analysis was performed at these burnups near and at the EOC-2 since all of the nuclear parameters which significantly affect the results of this limiting transient (i.e. void coefficient, Doppler coefficient, and scram reactivity function) do not coincidently have their most limiting values at one burnup. The maximum ΔMCPR's for 7x7, 8x8, and the PTA which resulted from this transient analysis (assuming at least 102% of rated core power and 100% of rated core flow) was 0.26, 0.34 and 0.35 respectively.

Addition of these ΔMCPR's to the safety limit MCPR (1.06) gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the operating limit MCPR's are 1.32 for 7x7 fuel, 1.40 for 8x8 fuel, and 1.41 for the PTA at rated core flow conditions.

The transient analyses were evaluated with scram reactivity functions that included a design conservatism factor of 0.80. The analyses also included a design conservatism factor of 1.25 for the void coefficients. These design cconservatism factors are acceptable as are the initial conditions used for the worst operational transients. The initial MCPR's assumed in the transient analyses were equal to or conservatively greater than the established operating limit MCPR's.

The turbine trip without bypass is also the anticipated operational transient which causes the most severe reactor isolation. Fast closure of the turbine stop valves therefore produces the largest pressure increase in the reactor. The peak transient pressure is limited by opening of the safety/relief valves. The results of the transient analysis show that the peak steam line pressure is limited by the safety/relief valves to 1203 psig, which is 27 psi below the 1230 psig set point of the first spring loaded safety valve. The Staff finds this to be acceptable.

A GE study [7] has shown that the required operating MCPR varies with the axial and local (pinwise) power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. The axial power peaking as assumed in the analysis is representative of beginning-of-cycle conditions located at the core midplane, with an axial peak-to-average of 1.40.

The bundle R-factors, which are a function of the local power peaking distribution, assumed in the GETAB analysis are also representative of a beginning-of-cycle condition. The R-factor values used were 1.080 for 7x7 fuel, 1.095 for 8x8 fuel and 1.045 for the PTA. During the cycle the local peaking, and therefore the R-factor, is reduced while the peak in the axial shape moves toward the bottom of the core. The amount by which

the R-factor decreases from beginning to end-of-cycle would, by itself, increase the required operating limit MCPR by approximately 1 percent. This adverse effect on the MCPR is offset, however, by a beneficial relocation of the axial peak to below the core midplane. Overall conservatism was applied in the determination of the required operating limit MCPR, since the assumed axial and local peaking were representative of the beginning of cycle, which provides the most adverse consistent set of axial and local peaking conditions.

Conservative analyses have shown, therefore, that operating limit MCPR's of 1.32 for 7x7 fuel, 1.40 for 8x8 fuel, and 1.41 for the PTA, at rated flow, will assure that the fuel cladding integrity safety limit MCPR will not be violated during any anticipated abnormal operational transient which may occur during Cycle 2. It is concluded, therefore, that the above operating limit MCPR's are acceptable for Peach Bottom Unit No. 3 during Cycle 2.

3.3.3 Rod Withdrawal Error

The rod withdrawal error transient (RWE) is discussed in References 1 and 2 for worst case conditions. The event description and analysis assumptions for the RWE are given in Reference 5. The information in these references indicates that the local power range monitor subsystem (LPRM's) will detect and alarm a high local power condition. However, if the reactor operator ignores the LPRM alarm, the rod block monitor (RBM) subsystem (set at 107% of full rated power at 100% core flow) will terminate the RWE transient in time to limit the maximum change in the critical power ratio to 0.20 for 7x7 fuel, 0.12 for 8x8 fuel and is less than 0.12 for the PTA. A RBM rod block occurring at 107% power and full core flow results in a peak linear heat generation rate (LHGR) of 21.5 Kw/ft and 17.5 Kw/ft for 7x7 and 8x8 fuels, respectively. These calculated LHGR's are below the safety limit LHGR's for 7x7 and 8x8 fuels respectively and are acceptable.

The rod withdrawal error analysis is based on the most reactive reactor state and conservatively assumes no xenon, which maximizes the amount of excess reactivity inserted upon withdrawal of the maximum worth control rod from the core. The analysis also allows for the most severe rod block monitor detector failure configuration allowed by the Technical Specifications.

Comparing the RWE ΔMCPR for each fuel type with the ΔMCPR's for the turbine trip without bypass transient shows that the latter transient is limiting for all fuel types (i.e 7x7, 8x8 and PTA). Operating limit MCPR's, based on the previously discussed turbine trip without bypass transient, will therefore also preclude the localized RWE transient from violating the safety limit MCPR of 1.06. It is concluded therefore that the analysis performed for the RWE transient and the predicted consequences are acceptable.

3.3.4 Operating MCPR Limits for Less than Rated Flow

To assure that the safety limit MCPR is not violated for the limiting flow increase transient (recirculation pump speed control failure) start-ing from less than rated flow conditions, the licensee will operate Peach Bottom Unit No. 3 in conformance with the limiting conditions for operation as stated in paragraph 3.5.K of the Technical Specifications. This requires that for core flow rates less than full rated flow, the licensee shall maintain the MCPR above the minimum operating values. The minimum MCPR values for less than full rated flow are equal to the MCPR for full rated flow (1.32 for 7x7, 1.40 for 8x8 and 1.41 for the PTA) multiplied by the respective K_f factor values appearing in Figure 3.5.1-E of the Technical Specifications. The K_f factor curves were generically derived and assure that for the most limiting flow increase transients, occurring from less than rated core flow, the actual MCPR will not violate the safety limit MCPR of 1.06.

It is concluded that application of the above stated K_f factors for reduced flow conditions results in calculated consequences for the limiting antic-ipated flow increase transients which do not exceed the thermal limits of the reactor coolant boundary.

3.4 Accident Analysis

3.4.1 ECCS Appendix K Analysis

a. Re-evaluation and Proposed Changes in Technical Specifications

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors". One of the requirements of the Order was that prior to any license amendment authorizing any core reloading "... the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46" The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amend-ments as may be necessary to implement the evaluation results and assumptions.

In Reference 10 the licensee submitted a re-evaluation of the Peach Bottom Unit No. 3 ECCS performance. The cases analyzed assumed that all of the core plate bypass flow holes had been plugged and the LPCI logic modification had been implemented. The results of the analysis are applicable to 7x7 fuel during Cycle 2 operation.

The loss-of-coolant accident was reanalyzed for the Reload 1 8x8 fuel with 100 mil channels and also for the PTA, with the results presented in References 1 and 2. The analysis indicated compliance with the Section 50.46 criteria and Appendix K to 10 CFR Part 50. The sub-mitted Technical Specification changes [4] included new MAPLHGR curves for the 8x8 fuel (Figure 3.5.1F) and the PTA (Figure 3.5.1.G).

b. Generic Errors and Their Resolution

During the review of this reload application, the General Electric Company informed the NRC staff that they had discovered generic errors in various input assumptions used in the "Appendix K" ECCS performance evaluations, which they had performed for several licensees. The Peach Bottom Unit No. 3 ECCS performance evaluations [1,2,10] con-tained several of the identified errors[11]. Some of the input errors were determined by GE and the Staff to be non-conservative. The Staff is currently reviewing this issue both generically and on a plant-by-plant basis. Concurrently, the Staff has also been review-ing several changes to the approved General Electric ECCS Evaluation Model which have been proposed by GE. The proposed model changes, if applied, would have the effect of lowering the post-LOCA calculated Peak Clad Temperature (PCT). Thus, application of these model changes would partially or wholly offset the negative effects of the identified non-conservative input assumptions.

The operation of the facility would be technically in non-conformance with the requirements of §50.46 in that specific computer runs for Peach Bottom Unit No. 3 employing the revised model with the errors corrected will not be complete for some time. However, the limita-tions on MAPLHGR set forth in the Technical Specifications approved herein will assure that the ECCS system will conform to the performance criteria of §50.46 during the interim. Therefore, the Commission has granted an exemption [13] to the requirements of §50.46 until a revised ECCS analysis can be completed.

3.4.2 Steamline Break Accident

The spectrum of steamline break accidents which are postulated to occur inside containment are covered by the ECCS analysis discussed in Section 2.4.1. The analysis results and conclusions of steamline break accidents occurring outside containment, as presented by the licensee, are accept-able on the generic staff review of NEDO-20360[6].

3.4.3 Fuel Loading Error

Fuel loading errors are discussed in References 1 and 2 respectively for 8x8 and PTA fuel bundles placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in References 1 and 2 indicates that a fuel loading error results in a peak linear heat generator rate (LHGR) of 16.8 Kw/ft for both the 8x8 fuel and the PTA in the misloaded fuel bundle. The minimum critical power ratio (MCPR) is 1.09 for the 8x8 fuel and 1.14 for the PTA in the misloaded fuel bundle during steady-state full power operation. The calculated peak LHGR is also less than that required to exceed the 1% plastic strain fuel design limit.

Thus, fuel damage criteria are not exceeded during this postulated event. Fuel bundles adjacent to a misloaded bundle will be negligibly affected. Thus it is concluded that the consequences of a fuel loading error are acceptable.

3.4.4 Control Rod Drop Accident

The Cycle 2 control rod drop accident for Peach Bottom Unit No. 3 is not within the generic bounding analysis presented in Reference 5. That is, although the actual Cycle 2 Doppler coefficient and accident reactivity shape functions for the cold and hot startup conditions conservatively fall within the values assumed in the bounding analysis, the scram reactivity shape functions for both hot and cold startup conditions do not. Therefore, the licensee has performed a plant specific control rod drop accident for Peach Bottom Unit No. 3 for Cycle 2.

The plant specific analysis was performed using actual hot and cold Doppler coefficients of reactivity corresponding to the beginning of the cycle (BOC), which is most limiting for this accidents since the Doppler coefficient is least negative at BOC.

The results of the analysis show that the reactivity insertion rate of the dropped rod is compensated sufficiently by Doppler feedback and scram reactivity effects to limit the energy deposition in the fuel to a maximum of 91.7 cal/gm for cold startup and 239.7 cal/gm for hot startup conditions. The results presented in Reference 2 also apply to the pressurized test assembly.

Thus, it is concluded that the results of a control rod drop accident from any in-sequence control rod movement will be below the design limit of 280 cal/gm and therefore are acceptable.

3.4.5 Fuel Handling Accident

The licensee notes in References 1 and 2 that fuel handling accident description, analysis and results provided in the FSAR and discussed in the generic reload topical report [5] are applicable to the 8x8 reload fuel and the PTA. That is, the total activity released to the environment and the resulting radiological exosures for the reload fuel will be less than those values presented in the FSAR for the 7x7 core. As identified in the FSAR, the radiological exposures for this accident with 7x7 fuel are well below the guidelines set forth in 10 CFR Part 100. Therefore, it is concluded that the consequences of this accident for the 8x8 and PTA fuel will also be well below the 10 CFR Part 100 guidelines.

3.5 Overpressure Analysis

In Reference 1, the licensee presented the results of an overpressure analysis to demonstrate that an adequate margin exists to the ASME code allowable vessel pressure, which is 110% of the vessel design pressure. The transient analyzed was the fast closure of all main steamline isolation valves with the conservative assumption that a reactor scram would occur on the second (high neutron flux) scram signal rather than the first (10% valve closure position switches). The analysis was performed for 105% reactor power with an end of cycle scram reactivity insertion rate curve, a void reactivity applicable for this reload, no credit for the relief function of the safety/relief valves and with all safety valves operative as necessary. The results of this analysis show that the peak pressure at the bottom of the vessel would be limited to 1279 psig. Furthermore, the results of this analysis indicate that neither of the two safety valves open during the event and thus the peak pressure will be limited to the aforementioned value by opening of the safety/relief valves only. Thus, the assumed failure of a safety valve would not alter the results of the analysis. Hence, the peak pressure at the bottom of the reactor vessel caused by fast closure of all MSIV's with indirect flux scram, no relief function of the safety/relief valves and one failed safety valve results in a 96 psi margin to the ASME vessel code limit of 1375 psig (110% of 1250 psig). This result is acceptable to the Staff.

3.6 Thermal-Hydraulic Stability Analysis

A thermal-hydraulic stability using the analytical methods discussed in Reference 5 and approved by Reference 6 was presented by the licensee for Peach Bottom Unit No. 3.

The results of the Cycle 2 analysis show that the 7x7 and 8x8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is well within the operational design guide in terms of decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios at both conditions are well within the operational design guide decay ratio. These results are acceptable to the NRC staff.

The NRC staff has expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve. The staff concerns relate to both the consequences of operating at an ultimate decay ratio and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing the staff concerns through meetings, topical reports and a proposed test program.

PECO has applied for a license amendment for Peach Bottom Unit No. 2 that would authorize a reactor core stability test program to be performed at the End of Cycle 2. The proposed test program is expected to be a significant aid in resolving the staff concerns on stability. The testing is scheduled in April 1977. The data obtained from the testing will be provided to the NRC staff by PECO and General Electric Company.

In the interim the staff has imposed a requirement on Peach Bottom Unit No. 3 which will restrict planned operations in the natural circulation flow mode. The licensee has agreed to this Technical Specification limitation. This restriction will provide a significant increase in the reactor core stability margins at Peach Bottom Unit No. 3 during Cycle 2. On the basis of the foregoing, the NRC staff considers the Thermal-Hydraulic stability of Peach Bottom Unit No. 3 to be acceptable.

3.7 Core Modification (Partial Drilling)

The NRC staff has previously approved the proposed modification to eliminate significant in-core vibration for facilities employing holes drilled in all fuel bundle lower tie plates (Reference 12). Concurrently with the evaluation for a "fully drilled" core, Reference 12 also considered and approved the mechanical and hydraulic affects of operating with only some of the fuel bundle lower tie plates drilled. This same

modification was previously approved and implemented on the Peach Bottom Unit No. 2 reactor. On the basis of the foregoing, the NRC Staff concludes that the "Partial Drilling" of fuel bundle lower tie plates for the Peach Bottom Unit No. 3 reactor is also acceptable.

3.8 Modifications to the Rod Sequence Control System (RSCS)

We have reviewed the licensee's submittal regarding the proposed modifications to the RSCS (Reference 4). The proposed modifications will upgrade the RSCS at Peach Bottom Unit No. 3 to the RSCS/BWR-4 design (Group Notch Control for plants with 25 wire probes) as reviewed on the Browns Ferry Dockets Nos. 50-259, 50-260 and 50-296).

The originally installed RSCS consists of two modes of operation: (1) between 100% rod density and 50% rod density, the sequence mode restricts rod movement such that all rods contained in the preselected group must be fully withdrawn before any other rods are withdrawn, and (2) between 50% rod density and 30% of rated power, the group "C" mode prevents any movement of certain designated rods ("C"group rods).

The proposed modification will delete the group "C" mode. Instead, the "group notch mode" will be provided between 50% rod density and 30% of rated power. The group notch mode of the RSCS will restrict the movement of rods assigned to a particular group so that no rod within a group can be moved more than one notch from any other rod in the group.

The control rod reactivity worths possible for a rod-drop accident are unchanged by the electrical-mechanical changes involved in the replacement of the simple notch control by the group notch control RSCS. Both systems are intended to maintain the same type of rod group patterns during zero and low power operation. These patterns were developed to assure that control rod worths of a magnitude sufficiently large to exceed the NRC Staff's criterion of 280 cal/gm maximum energy deposition, as a result of a rod drop accident, will not occur. The evaluation presented in Section 2.4.4 above considered the modified RSCS.

We conclude, based on our review, that the proposed RSCS design modifications will upgrade the Peach Bottom Unit No. 3 design so that it is equal to the design accepted on the Browns Ferry facility and that it will meet the requirements specified in our Safety Evaluation Report for the Peach Bottom Atomic Power Station Units Nos. 2 and 3, Supplement 1, dated December 11, 1972.

4.0 Physics Startup Testing

As part of our review of Reload 1 of Peach Bottom Unit No. 3, the licensee was requested to provide a description of the Cycle 2 physics startup test

program. In response to that request, the physics startup test program
was provided by the licensee in Reference 14. The combined physics
startup tests proposed by the licensee along with the tests required
to assure compliance with the Technical Specifications, provide an
acceptable physics startup test program.

5.0   Pressurized Test Assembly

Based on our review of References 2 and 11, we conclude that the
Pressurized Test Assembly (PTA) meets the design acceptance criteria
for a standard 8x8 reload bundle as stated in Reference 5. Therefore,
we approve the use of one lead PTA in Peach Bottom Unit No. 3. This
approval does not permit expanded use of like assemblies in this
reactor or in other reactors without further NRC Staff review of their
specific application.

6.0   Fast Scram Control Rod Drive

The licensee has proposed [4] to replace one standard control rod drive
(CRD) in the Peach Bottom Unit No. 3 control rod drive system with one
Fast Scram Control Rod Drive [3] (FSCRD) supplied by GE. The mechanical
design of the new replacement drive includes interface features which
are identical to the conventional CRD. The internal design of the FSCRD
incorporates the use of higher strength materials to withstand the increased
internal pressures and impact loads associated with the faster scram. The
FSCRD design has also significantly reduced the high stresses associated
with the cross-sectional change of the standard CRD.

The FSCRD also features a change in the buffer design, which includes a
new stop piston, piston tube, buffer shaft, buffer piston and spring.
The change in buffer design permits an increase in the buffer pressure
associated with the FSCRD without increasing the duty on the index tube
and buffer seals.

For an actual scram situation, during Cycle 2, the scram time of the
FSCRD will be less than that of a standard CRD since the nitrogen pre-
charge pressure for the associated Hydraulic Control Unit will be
increased during Cycle 2. Thus, the Technical Specification CRD minimum
scram time requirements will also apply to the FSCRD.

The internal changes to the drive will not increase the possibility that the control rod can disengage from its drive, nor does it increase the possibility that the rod could become stuck out of the core due to mechanical interference. Moreover, the FSCRD, prior to shipment to Peach Bottom Unit No. 3, will have experienced the equivalent of 5 years of normal operation in a simulated reactor environment.

Only a limited review of the FSCRD has been performed because (1) there will be only one in the Peach Bottom Unit No. 3 CRD system and (2) it will be installed for only one cycle (but may be replaced with a second FSCRD for one additional cycle).

Based on our limited review of References 3 and 11, we approve the use of one Fast Scram Control Rod Drive in Peach Bottom Unit No. 3 during Cycle 2. The approval does not allow expanded use of similar FSCRD's in this reactor or in other reactors without further Staff review of their specific application.

In order to facilitate future reviews in which FSCRD's are to be more extensively used for the Control Rod Drive System, the licensee should first report the results of his findings concerning the performance of the FSCRD at Peach Bottom Unit No. 3 during Cycle 2.

7.0 Technical Specification Changes

The proposed Technical Specification changes [4], based on GETAB, for Peach Bottom Unit No. 3 incorporate the fuel cladding Integrity Safety Limit MCPR and Operating Limit MCPR requirements for 7x7, 8x8 and the PTA as identified in References 1 and 2. The licensee has proposed: (a) MAPLHGR curves for the standard reload 8x8 fuel and the PTA, (b) to incorporate densification power spiking effects on the maximum LHGR for the reload 8x8 by using a correction equation. The licensee has agreed to Technical Specifications which restrict all planned reactor operation in natural circulation.

We find the Technical Specification changes, proposed to date, acceptable and consistent with the information in the Peach Bottom Unit No. 3, Reload 1 licensing submittal.

8.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves

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

9.0 Conclusion

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

Dated: March 23, 1977

REFERENCES

1. "General Electric Boiling Water Reactor Reload-1 License Amendment for Peach Bottom Atomic Power Station Unit No. 3." NEDO-21363, Class 1, November, 1976. Appendix A to NEDO-20360, Revision 1, Supplement 4, April, 1976 (NEDO-20360).

2. "Pressurized Test Assembly Supplemental Information for Reload-1 Licensing Amendment for Peach Bottom Atomic Power Station Unit No. 3." NEDO-21363-1, Supplement 1, Class 1, November, 1976.

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

4. Philadelphia Electric Company Application for Amendment of Facility Operating License DPR-56, November, 1976.

5. General Electric Generic Reload Licensing Application for 8x8 Fuel," Revision 1, Supplement 4, April, 1976, NEDO-20360.

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

7. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application", General Electric Company, BWR Systems Department, November, 1973 (NEDO-10958).

8. "Topical Report Evaluation of General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application", by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commision, September, 1974.

9. General Electric Letter (John A. Hinds) to U. S. Atomic Energy Commission (Walter Butler) "Responses to the Third set of AEC Questions on the General Electric Licensing Topical Reports, NEDO-10958 and NEDE-10958, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application." July, 1974.

10. "Peach Bottom Atomic Power Station Unit No. 3 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged." NEDO-21140, 75 NED72, Class 1, December, 1975.

11. Philadelphia Electric Company Letter (M. J. Cooney) to United
    Stated Nuclear Regulatory Commission (G. Lear), February 7, 1977.

12. "Safety Evaluation Report on the Reactor Modification to Eliminate
    Significant In-Core Vibration in Operating Reactors with 1-Inch
    Bypass Holes in the Core Support Plant," by Office of Nuclear
    Reactor Regulation, U. S. Nuclear Regulatory Commission, February, 1976.

13. Exemption from 10 CFR §50.46 for Peach Bottom Atomic Power Station
    Unit No. 3, March 11, 1977.

14. Philadelphia Electric Company letter (M. J. Cooney) to United
    States Nuclear Regulatory Commission (G. Lear), March 7, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-278

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 to Facility Operating License No. DPR-56, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised the license and its appended Technical Specifications for operation on the Peach Bottom Atomic Power Station, Unit No. 3 (the facility), located in Peach Bottom, York County, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment consists of changes in the Technical Specifications to authorize operation of the facility with (1) up to 188 General Electric (GE) 8x8 Reload Fuel Bundles, (2) one Pressurized Test Assembly (PTA), (3) one Fast Scram Control Rod Drive (FSCRD), (4) holes drilled in the lower tie plate of all reload fuel bundles to provide an alternate bypass flow path, and (5) modifications to the Rod Sequence Control System (RSCS). The amendment also deleted obsolete paragraphs from the license.

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. Notice

of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on December 30, 1976 (41 F.R. 56894). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.
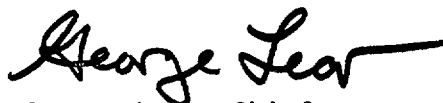
The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 17, 1976, and supplement thereto dated February 7, 1977, (2) Amendment No. 33 to License No. 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 single 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 23 day of March, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

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