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Docket No. 50-277

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

Dear Mr. Bauer:

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station Unit No. 2. The amendment consists of changes to the Technical Specifications and is in response to your application dated July 28, 1978 as supplemented by letters dated September 5, 26, and October 4, 1978.

The changes permit operation of Peach Bottom Unit No. 2 with a new type reload fuel (8x8R) for Cycle 4.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 48
2. Safety Evaluation
3. Notice

cc w/enclosures:
see next page

subject to amendment not being signed or issued prior to 16Oct78

Contract 1

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ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et al (the licensee), dated July 28, 1978, as supplemented September 5, 26, and October 4, 1978, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48, 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

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 16, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

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

Remove	Replace
iv	iv
v	v
vi	vi
9	9
10	10
11	11
12*	12*
14	14
15	15
15a	deleted
15b	deleted
18	18
19	19
20	20
21	21
33	33
35	35
40	40
73	73
74	74
91	91
92	92
103	103
108	108
111	111
133a	133a
133b	133b
-	133c (added)
140	140
140a	140a
140b	140b
140c	140c
140d	140d
140e	140e
141b	141b
-	142f (added)
157	157

*No change on this page.

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objectives:

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

Specification:

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

The existence of a minimum critical power ratio MCPR less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

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

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objectives:

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

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

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

$$S \le 0.66W + 54\%$$

where:

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

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

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 3 values determined by the following:

a. $S \leq (0.66W+54\%) \dfrac{2.62}{\text{MTPF for 7x7 fuel}}$

b. $S \leq (0.66W+54\%) \dfrac{2.44}{\text{MTPF for 8x8 fuel}}$

c. $S \leq (0.66W+54\%) \dfrac{2.51}{\text{MTPF for 8x8R fuel}}$

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 = 2.62 for 7x7 fuel, 2.44 for 8x8 fuel, and 2.51 for 8x8R 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.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia) When the reactor pressure is ≤ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.	B. APRM Rod Block Trip Setting SRB ≤ 0.66W + 42% where: SRB= Rod block setting in percent of rated thermal power (3293 MWt) W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 x 10^6 lb/hr). In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified to the more limiting (lower) of the 3 values determined by the following: 1. $SRB \le (0.66W+42\%) \dfrac{2.62}{\text{MTPF for 7x7 fuel}}$ 2. $SRB \le (0.66W+42\%) \dfrac{2.44}{\text{MTPF for 8x8 fuel}}$ 3. $SRB \le (0.66W+42\%) \dfrac{2.51}{\text{MTPF for 8x8R fuel}}$ MTPF = The value of the existing maximum total peaking factor Design value of A = 2.62 for 7x7 fuel, 2.44 for 8x8 fuel, and 2.51 for 8x8R fuel.
C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.1 in. above the top of the normal active fuel zone.	C. Scram and isolation reactor low water level -- ≥538 in. above vessel zero (0" on level instruments)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING		
	2.1 (Cont'd)		
	D.	Scram-- turbine stop	≤10 percent valve closure
	E.	Scram-- turbine control fast closure on loss of control oil pressure. 500<P<850 psig.	
	F.	Scram--low condenser vacuum	≥23 inches Hg vaccum
	G.	Scram--main steam line isolation	≤10% valve closure
	H.	Main steam isolation valve closure--nuclear system low pressure	≥850 psig
	I.	Core Spray & LPCI actuation--reactor low water level	≥378 in. above vessel zero (-159.5 in. indicated level)
	J.	HPCI & RCIC actuation--reactor low water level	≥490 in. above vessel zero (-49.5 in. indicated level)
	K.	Main steam isolation valve closure--reactor low water level	≥490 in above vessel zero (-49.5 in. indicated level)

1.1.A BASES (Cont'd)

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

The basis for the uncertainties in the core parameters is 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 2 during any fuel cycle would not be as severe as the distribution used in the analysis.

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

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

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factor this 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 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage.

1.1.C BASES (Cont'd.)

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

The computer provided with Peach Bottom Unit 2 has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied upon to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

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

E. References

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

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

3. "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDE-24011-P-3, March 1978.

2.1 BASES (Cont'd.)

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

Steady state operation without forced recirculation will not be permitted, except during startup testing. 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.

ii. The licensed maximum power level is 3293 MWt.

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

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

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

A. Neutron Flux Scram

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

2.1.A BASES (Cont'd.)

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

The scram trip setting must be adjusted to assure that the LHGR transient peak is not increased for any combination of 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 greater than 1.07 when the transient is initiated from MCPR greater than the operating limit given in Specification 3.5.K.

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

2.1.A BASES (Cont'd.)

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

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

B. APRM Rod Block Trip Setting

The APRM system provides a control rod block to avoid conditions which would result in an APRM scram trip if allowed to proceed. The APRM rod block trip setting, like the APRM scram trip setting, is automatically varied with recirculation loop flow rate. The flow variable APRM rod block trip setting provides margin to the APRM scram trip setting over the entire recirculation flow range. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the Maximum Total Peaking Factor (MTPF) exceeds the design value A for each fuel type.

2.1 BASES (Cont'd.)

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

The set point for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in 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 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 31 in. below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of less than or equal to 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.07 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

E. Turbine Control Valve Scram

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

2.2 BASES: REACTOR COOLANT SYSTEM INTEGRITY

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

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

Eleven safety/relief valves and two safety valves have been installed on 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 1295 psig for Peach Bottom Unit 2 if a neutron flux scram is assumed. This results in a 80 psig margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient (load rejection with bypass valve failure to open and Recirculation Pump Drive Motor Trip) assuming a turbine trip scram is presented in NEDO-24132, Revision 1 for Peach Bottom Unit 2. This analysis shows that the 11 safety/relief valves limit pressure at the safety valves to 25 psig below the setting of the safety valves. Therefore, the safety valves will not open.

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

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

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

LIMITING CONDITIONS FOR OPERATION

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

SURVEILLANCE REQUIREMENTS

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2 respectively.

B. Daily during reactor power operation, the 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 2.62 for 7x7 fuel, 2.44 for 8x8 fuel, or 2.51 for 8x8R fuel.

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 = 2.62 for 7x7 fuel, 2.44 for 8x8 fuel, and 2.51 for 8x8R fuel. MTPF is the value of the existing maximum total peaking factor.

13. See Section 2.1.A.1.

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)	$\leq (0.66W+42) \times \frac{A}{MTPF}$ (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)	$\leq (0.66W+41) \times \frac{A}{MTPF}$ (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)

Amendment No. 23, 34, 38, 42, 48

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 = 2.62 for 7x7 fuel, 2.44 for 8x8 fuel, and 2.51 for 8x8R 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 ≥ 100 cps.

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

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

7. The trip is bypassed when the reactor power is ≤ 30%.

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

3.2 BASES (Cont'd)

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

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

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

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

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

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1 out of n: e.g., any trip on one of 6 APRM's, 8 IRM's, or 4 SRM's will result in a rod block.

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

3.2 BASES (Cont'd)

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection: i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

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

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

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

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

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

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.B Control Rods (Cont'd)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

   a. Both RMB channels shall be operable, or
   b. Control rod withdrawal shall be blocked, or
   c. The operating power level shall be limited so that the MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of a single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

Above 950 psig

% Inserted from Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	5.0

4.3.B Control Rods (Cont'd)

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

C. Scram Insertion Times

1. After each refueling outage all operable fully withdrawn insequence rods shall be scram time tested during operational hydrostatic testing or during startup from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to synchronizing the main turbine generator initially following restart of the plant.

3.3 and 4.3 BASES (Cont'd.)

B. Control Rods

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

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

3. The Rod Worth Minimizer (RWM) and sequence mode of the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. The group notch mode of the RSCS restricts movement of rods assigned to each notch group to notch withdrawal and insertion. All patterns associated with these restrictions have the characteristic that, assuming the worst single deviation from the restrictions, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in the peak enthalpy of any pellet exceeding 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Sections 3.6.6, 14.6.2 and 7.16.3.3 of the FSAR, NEDO-10527 and supplements thereto, and NEDO-24132, Revision 1.

3.3 and 4.3 BASES (Cont'd.)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. Analysis of the limiting power transients shows that the negative reactivity rates resulting from the scram (Ref. NEDO-24132, Revision 1) with the average response of all drives as given in the above Specification, provide the required protection, and the MCPR remains greater than 1.07.

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.I Average Planar LHGR	4.5.I Average Planar LHGR
During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.5.1.A, B, C, D, F, & G, 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.	The APLHGR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at ≥25% rated thermal power.
3.5.J Local LHGR	4.5.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:	The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.
$LHGR \leq LHGRd\ [1-(\Delta P/P)max\ (L/LT)]$	
LHGRd = Design LHGR = 18.5 kW/ft for 7x7 fuel 13.4 kW/ft for 8x8, 8x8R, and 8x8 LTA fuel	
$(\Delta P/P)max$ = Maximum power spiking penalty = 0.026 for 7x7 fuel = 0.022 for 8x8, 8x8R, and 8x8 LTA fuel	
LT = Total core length = 12 ft for 7x7 & 8x8 fuel = 12.5 ft for 8x8R & 8x8 LTA fuel	
L = Axial position above bottom of core	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.J Local LHGR (Cont'd) If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	
3.5.K Minimum Critical Power Ratio (MCPR) During power operation, the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Table 3.5-2 times kf, where kf is as shown in Figure 3.5.1.E. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within one (1) hour to restore MCPR to within prescribed limits. If the MCPR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless MCPR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	4.5.K Minimum Critical Power Ratio (MCPR) MCPR shall be checked daily during reactor power operation at ≥25% rated thermal power.

Table 3.5-2

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

Fuel Type	MCPR Operating Limit For Incremental Cycle 3 Core Average Exposure	
	BOC to 1000 MWD/t Before EOC	1000 MWD/t before EOC To EOC
7x7	1.31 (RWE)	1.31 (RWE)
8x8	1.26 (RWE)	1.28 (LR)
8x8R/LTA	1.25 (LR)	1.28 (LR)

RWE - Rod Withdrawal Error
LR - Load Rejection with failure of bypass valves to open

3.5 BASES (Cont'd.)

H. Engineering Safeguards Compartments Cooling and Ventilation

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

I. Average Planar LHGR

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

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

The calculational procedure used to establish the APLHGR shown on Figures 3.5.1.A, B, 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. Input and model changes in the Peach Bottom loss-of-coolant analysis which are different from the previous analyses performed with Reference 4 are described in detail in Reference 8. These changes to the analysis include: (1) consideration of the counter current flow limiting (CCFL) effect, (2) corrected code inputs, and (3) the effect of drilling alternate flow paths in the bundle lower tie plate.

3.5.I BASES (Cont'd.)

A list of the significant plant 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 or greater 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 5.2.3 of Reference 7.

K. Minimum Critical Power Ratio (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 1.07, and analyses of the abnormal operational transients presented in References 6 and 7. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

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

3.5.K BASES (Cont'd.)

The limiting transients which determine the required steady state MCPR limits are given in Table 3.5-2. These transients yield the largest ΔCPR for each class of fuel. When added to the safety limit MCPR of 1.07, the required minimum operating limit MCPR's of specification 3.5.K are obtained.

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

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

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

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

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

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

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

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

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

3.5.K BASES (Cont'd.)

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

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

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

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

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

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

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

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

M. <u>References</u>

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

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

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

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

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

6. Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 2 Reload No. 2, NEDO-24132, Revision 1, September 1978.

7. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-3, March 1978.

8. Loss-of-Coolant Accident Analysis For Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977.

TABLE 3.5-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power	3440 MWt which corresponds to 105% of rated steam flow
Vessel Steam Output	14.05×10^6 1bm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area For Large Breaks -	
Discharge	1.9 ft² (DBA)
Suction	4.1 ft²
Assumed Number of Drilled Bundles	360

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
7x7, Type 2	7 x 7	18.5	1.5	1.2
7x7, Type 3	7 x 7	18.5	1.5	1.2
8x8, Type H	8 x 8	13.4	1.4	1.2
8x8, Type L	8 x 8	13.4	1.4	1.2
8x8R/LTA	8 x 8	13.4	1.4	1.2

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

4.5.L BASES (Cont'd)

adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the Kf.

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

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

PEACH BOTTOM UNIT 2

8x8R Fuel

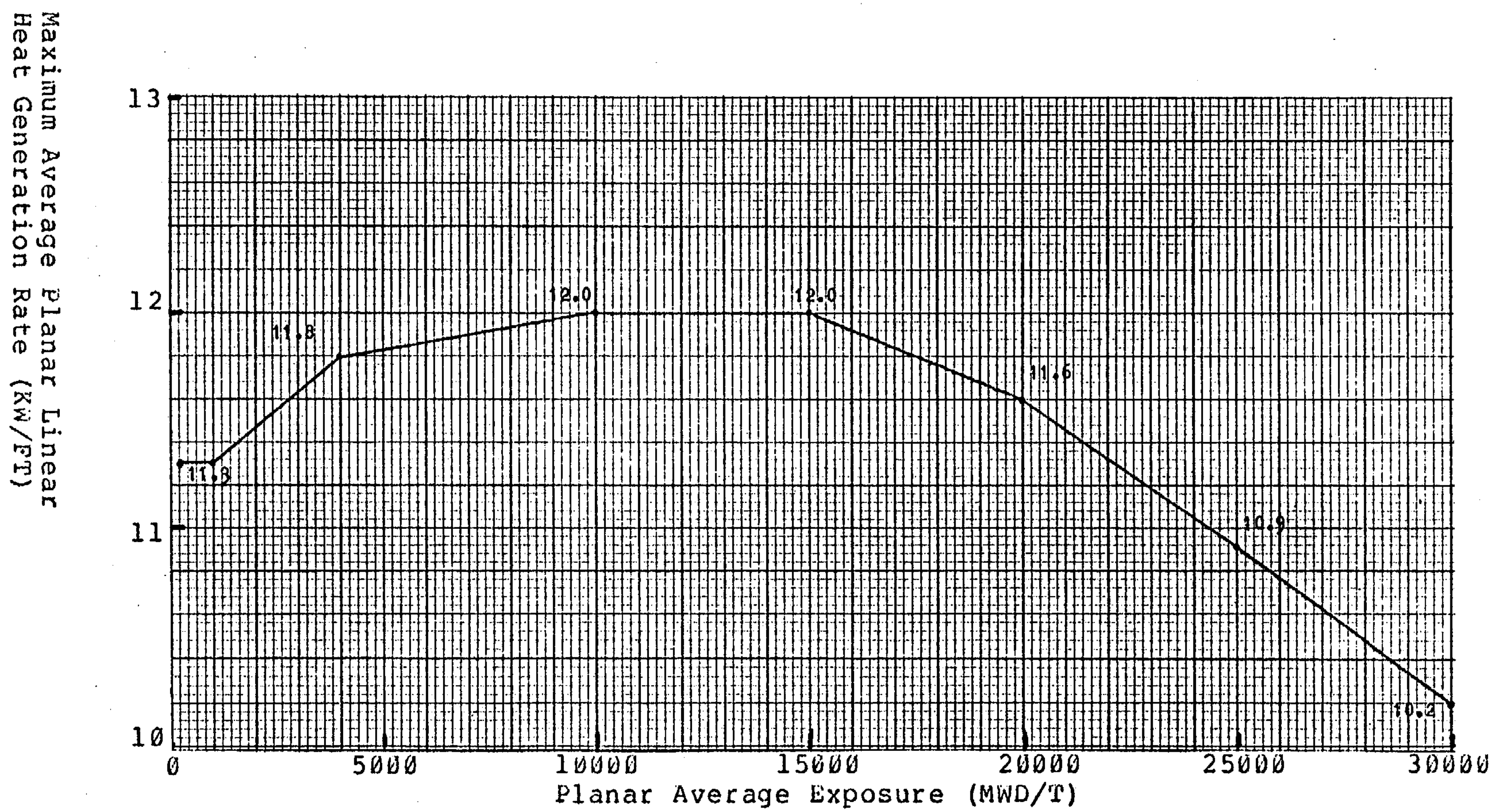

Figure 3.5.1.G Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

3.6.D & 4.6.D BASES: SAFETY AND RELIEF VALVES

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

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

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit 2 with a total capacity of 79.51% of rated steam flow. The analysis of the worst overpressure transient, (3 second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1295 psig if a neutron flux scram is assumed. This results in a 80 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the power generation design basis, the total pressure relief system capacity of 79.51% has been divided into 65.96% safety/relief (11 valves) and 13.55% safety (2 valves). The analysis of the plant isolation transient (load rejection with bypass valve failure to open and Recirculation Pump Drive Motor Trip) assuming a turbine trip scram is presented in NEDO-24132, Revision 1. This analysis shows that the 11 safety/relief valves limit pressure at the safety valves to 25 psig below the setting of the safety valves. Therefore, the safety valves will not open.

Experience in safety/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 safety/relief and safety valves are benchtested every second

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 48 TO FACILITY LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 2

DOCKET NO. 50-277

1.0 Introduction

By letter[1] dated July 28, 1978 and supplemented by letters[2,3] dated September 5, 26 and October 4, 1978, the Philadelphia Electric Company (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-44 for Peach Bottom Atomic Power Station Unit No. 2 (PB-2). The proposed changes relate to the third refueling of PB-2, involving the replacement of 260 exposed 7x7 fuel assemblies with a like number of fresh, two water rod, retrofit 8x8 fuel assemblies designed and fabricated by the General Electric Company, together with the reconstitution and reloading of an exposed lead retrofit 8x8 assembly, previously irradiated during Cycles 2 and 3. The proposed amendment was noticed in the FEDERAL REGISTER on September 7, 1978 (43FR39869). In support of this reload application for PB-2, the licensee has submitted a supplemental reload licensing document [4] prepared by the General Electric Company (GE), proposed Technical Specification changes[1], information relating to the reconstitution of a lead test fuel assembly[2] and responses[3] to our request[5] for additional information on the reload application.

This reload (Reload 3) is the first for PB-2 to incorporate GE's retrofit 8x8R fuel design on a batch basis. Previously, for Reload 1, four lead retrofit test assemblies (LTAs) were loaded into the PB-2 core. These assemblies have operated satisfactorily for two cycles.

The description of the nuclear and mechanical design of the Reload 3 8x8R fuel and the exposed standard 8x8 fuel design used for Reloads 1 and 2 is contained in GE's generic licensing topical report for BWR reloads[6]. Reference 6 also contains a complete set of references to GE's topical reports which describe GE's BWR reload analysis methods for the nuclear, mechanical, thermal-hydraulic, transient and accident calculations, together with information addressing the applicability of these methods to cores containing a mixture of 7x7, 8x8 and 8x8R fuel. Portions of the plant-specific data, such as operating conditions and design parameters which are used in transient and accident calculations, have also been included in the topical report.

Our safety evaluation(7) of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic and transient and accident calculations, as applied to mixed cores containing 7x7, 8x8, and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was expressed in the staff's evaluation(8) of the information in Reference 9. References 7 and 8 are incorporated in this safety evaluation report by reference.

As part of our evaluation(7) of Reference 6 we found the cycle-independent input data for the reload transient and accident analyses for PB-2 to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 4, which follows the format and content of Appendix A of Reference 6.

As a result of the staff's generic evaluation(7) of a substantial number of safety considerations related to use of 8x8R fuel in mixed core loadings with 8x8 and 7x7 fuel, only a limited number of additional review items are included in this evaluation. These include the plant and cycle-specific input data and results presented in References 3 and 4, the LOCA-ECCS analysis results for the reload fuel design, and those items identified in Reference 7 as requiring special attention during reload reviews.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 4, 260 fresh 8x8R fuel bundles, with a bundle average enrichment of 2.84 wt/% U-235 will be loaded into the core, replacing a like number of exposed 7x7 assemblies. The remainder of the 764 fuel assembly reload core will consist of the irradiated 7x7, 8x8 and lead 8x8R fuel assemblies exposed during the first three fuel cycles. The reference core loading for Cycle 4 will result in eighth core symmetry, which is consistent with previous cycles.

The information provided in Section 6 of Reference 4 indicates that the fuel temperature and void dependant behavior of the reconstituted core is not significantly different from previous cycles of PB-2. Additionally, scram effectiveness, as shown in Figures 2a and 2b of Reference 4, is also similar to earlier cycles. The 1.1%Δk/k calculated shutdown margin for the reconstituted core meets the Technical Specification requirement that the core be subcritical by at least 0.38%Δk/k in the most reactive operating state with the single most reactive control rod fully withdrawn and all other rods fully inserted. Finally, Reference 4 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by 3.3%Δk at 20^{o}C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System.

2.2 Thermal-Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 7, for BWR cores which reload with GE's retrofit 8x8R fuel, the allowable minimum critical power ratio (MCPR), resulting from either core-wide or localized abnormal operational transients, is equal to 1.07. With this MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee for Cycle 4 represents a .01 increase from the 1.06 SLMCPR applicable during Cycle 3. The basis for the revised safety limit is addressed in Reference 6, while our generic approval of the new limit is given in Reference 7.

2.2.2 Operating Limit MCPR

Various transient events will reduce the MCPR from its normal operating value. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed by the licensee to determine which event results in the largest reduction in the minimum critical power ratio. Each of the events has been analyzed for each of the several fuel types (i.e., 7x7, 8x8, 8x8R), for exposure intervals corresponding to BOC4 to EOC4-1000Mwd/t and EOC4-1000 Mwd/t to EOC4.

The methods used for these calculations, including cycle-independent initial conditions and transient input parameters are described in Reference 6. Our acceptance of the values used and related transient analysis methods appear in Reference 7. Supplementary cycle-dependent initial conditions and transient input parameters used in the analysis appear in the ~~Table~~ table in Sections 6 and 7 of Reference 4. Our evaluation of the methods used to develop these supplementary transient input values have already been addressed and appear in Reference 7. The overall transient methodology, including cycle-independent transient analysis inputs, provides an adequately conservative basis[7] for the determination of transient ΔMCPRs. The transient events analyzed were load rejection without bypass, turbine trip without bypass, feedwater controller failure, loss of 100°F feedwater heating and control rod withdrawal error.

All of the transients, except for the load rejection without bypass (LR w/o BP), were analyzed by the licensee using the generic methods and assumptions described in the generic reload topical report. For Cycle 4, the licensee analyzed the LR w/o BP modelling a plant-unique load shedding recirculation pump trip which is currently installed at both Peach Bottom Unit No. 2 and Peach Bottom Unit No. 3. The pump trip results in a substantial reduction in the calculated transient ΔMCPR when compared to the LR w/o BP without recirculation pump trip (RPT). This stems from the substantial negative reactivity addition which occurs when the core void fraction rapidly increases as a result of the core flow coastdown.

The LR w/o BP, with the load shedding RPT included in the analysis, was not calculated to be a limiting event for any fuel type or exposure interval. However, in view of the uncertain MCPR benefits of the subject RPT feature as well as its undocumented reliability of the RPT system the licensee was requested to either (a) perform a reanalysis of the LW w/o BP without taking credit for the load shedding RPT or (b) document the high reliability of the pump trip system based on design, testing and related technical specification requirements. The licensee elected to reanalyze the event without taking credit for the RPT feature.

The reanalysis[3] showed that, depending on fuel types, the LR w/o BP is the most limiting transient during certain exposure intervals of Cycle 4.

Based on our composite review of the original[4] and revised analyses[3], for the 7x7 fuel types, the most limiting abnormal operational transient is the control rod withdrawal error regardless of cycle exposure. For the standard 8x8 fuel type, the most limiting event from BOC4 to EOC4-1000 MWd/t is the control rod withdrawal error, while from EOC4-1000 MWd/t to EOC4 the load rejection without bypass is most limiting. Finally, for the reload 8x8R and lead test assemblies the load rejection without bypass is limiting throughout Cycle 4. A summary of the most severe ΔMCPRs is as follows:

	ΔMCPR	
Fuel Type	BOC4 to EOC4-1000 Mwd/t	EOC4-1000 Mwd/t to EOC4
7x7	0.24	0.24
8x8	0.19	0.21
8x8R/LTA	0.18	0.21

Addition of the above ΔMCPRs to the 1.07 safety limit MCPR gives the required operating limit MCPR for each fuel type and exposure interval. Accordingly, based on the original and revised analyses, the licensee has proposed the following operating limit MCPRs for PB-2 during Cycle 4:

	OPERATING LIMIT MCPR	
Fuel Type	BOC4 to EOC4-1000 Mwd/t	EOC4-1000 Mwd/t to EOC4
7x7	1.31	1.31
8x8	1.26	1.28
8x8R/LTA	1.25	1.28

The licensee has also considered the effect of a possible fuel loading error on bundle CPR. An analysis of the most severe misoriented fuel loading error using GEs new methodology[10,11], which, as modified, has been approved[12] by the staff, shows that rotating a fresh 8x8R fuel bundle will not cause a violation of the 1.07 safety limit MCPR. Additionally, an analysis of the most severe mislocated fuel bundle using GE's standard analysis procedure, shows that mislocating a fresh 8x8R will not violate the MCPR safety limit. Thus, when PB-2 is operated in accordance with the above operating limit MCPRs the 1.07 SLMCPR will not be violated in the event of the most severe abnormal operational transients or fuel loading errors. This is acceptable to the staff.

2.2.3 Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawl error and fuel loading error events were also analyzed by the licensee using methods acceptable to the Staff to determine the maximum linear heat generation rates (LHGR). The results for PB-2, Cycle 4 show that the fuel type and exposure dependent safety limit LHGRs, given in Table 2-3 of Reference 6 will not be violated should these events occur.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

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 amendments as may be necessary to implement the evaluation assumptions.

For Cycle 4 the licensee has reevaluated the adequacy of PB-2 ECCS performance in connection with the new reload fuel design, using methods previously approved by the staff. The results of these plant-specific analyses are given in Reference 4.

We have reviewed the information submitted by the licensee and conclude that PB-2 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when operated in accordance with the MAPLHGR versus Average Planar Exposure values given in Section 6 of Reference 4.

2.3.2 Control Rod Drop Accident

For the worst case control rod drop accident (CRDA) during hot startup conditions, the key plant-specific nuclear characteristics are within those used in the bounding CRDA analysis given in Reference 6. Since the bounding analysis showed that the peak fuel enthalpy does not exceed the 280 cal/gm fuel enthalpy design limit, the peak fuel enthalpy associated with a CRDA from hot startup condition for PB-2 during Cycle 4 will also be within the 280 cal/gm design limit.

Because the characteristic accident analysis input parameters for the worst case CRDA starting from cold startup conditions did not satisfy all of the assumptions of the bounding analysis the licensee reanalyzed this event on a plant-specific basis. The results showed the peak fuel enthalpy to be less than the 280 cal/gm limit which is acceptable.

2.4 Overpressure Analysis

The licensee has reanalyzed the limiting pressurization transient to demonstrate that the ASME Boiler and Pressure Vessel Code requirements are met during Cycle 4. The methods used for this analysis, when modified to account for one failed safety valve, have been previously approved by the staff. The acceptance criteria for this event is that the calculated peak transient pressure not exceed 110% of design pressure, i.e., 1375 psig. The reanalysis shows that the peak pressure at the bottom of the reactor vessel is equal to 1315 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is acceptable to the staff.

2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for PB-2, Cycle 4 using the methods described in Reference 6. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state, (corresponding to the intersection of the natural circulation curve and 105% rod line on the power-flow map) are below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. Although a final test report has not as yet been received by the staff for review, it is expected that the test results will aid considerably in resolving the staff concerns.

For Cycle 3, the staff, as an interim measure, added a requirement to the PB-2 Technical Specifications which restricted planned operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins during Cycle 4 so that the decay ratio is <1.0 in all operating modes. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of PB-2 during Cycle 4 to be acceptable.

4.0 Peach Bottom Unit No. 2 Test Programs

4.1 Lead Test Assemblies and Developmental Channels

As part of the first reload of PB-2, twelve developmental channels and four lead test assemblies (LTAs) of the retrofit 8x8 fuel type were inserted in the core for qualification irradiation testing. Examinations (13,14) of these channels and LTAs have shown that they are performing acceptably. For Cycle 4, one of the LTAs will be reconstituted by replacing two exposed fuel rods with two fresh 3.00 wt/% fuel rods, in order to permit destructive examination of the irradiated test rods. Analyses which have been performed by the licensee(2) demonstrate that the planned reconstitution will not adversely impact fuel bundle performance during normal, abnormal operational transient and postulated accident conditions. Based on our review, we approve the continued use of the four LTAs and twelve developmental channels at Peach Bottom Unit No. 2 during Cycle 4.

4.2 Physics Startup Testing

Several of the key reload safety analysis inputs and results can be assured via preoperational testing. In order to provide this assurance the licensee will perform a series of physics startup tests, which are described in Reference 15. Based on our review this program is acceptable. A written report, describing the results of the physics startup tests, will also be provided by the licensee within 90 days of startup which is also acceptable.

5.0 Technical Specification Changes

The proposed technical specification changes(1) nclude a revised fuel cladding integrity safety limit MCPR, a revised exposure-dependent operating limit minimum critical power ratios (MCPR) for each fuel type, addition of a MAPLHGR vs average planar exposure curve and addition of a design maximum total peaking factor for the reload 8x8R fuel assemblies.

The revised 1.07 safety limit MCPR results in a .01 increase from the 1.06 safety limit MCPR (SLMCPR) used during Cycle 3. Based on our generic review(7), we find the use of a 1.07 SLMCPR for PB-2 during Cycle 4 to be acceptable. Also, based on the discussions appearing in Section 2.2.2 herein, the staff finds the proposed operating limit MCPRs, as modified(3) to reflect the reanalysis of the load rejection without bypass transient to be consistent with and adequately supported by the Reload 3 safety analyses.

The proposed 8x8R design maximum total peaking factor of 2.51, used in connection with the APRM Flux Scram and APRM Rod Block Trip Settings has been reviewed and found to be acceptable. Finally, we find that the proposed MAPLHGR vs average planar exposure curve is adequate to assure conformance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 for the reload 8x8R fuel assemblies.

6.0 Environmental Considerations

We have determined that the amendment does not involve 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 Section 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.

7.0 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: October 16, 1978

References:

1. Philadelphia Electric Company letter (Bradley) to USNRC (Denton) dated July 28, 1978.

2. Philadelphia Electric Company letter (Bradley) to USNRC (Denton) dated September 5, 1978

3. Philadelphia Electric Company letters (Bradley) to USNRC (Denton) dated September 26, 1978 and October 4, 1978 transmitting NEDO-24132, Rev. 1.

4. "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit No. 2, Reload No. 3" NEDO-24132, July 1978.

5. USNRC letter (Ippolito) to Philadelphia Electric Company (Bauer) dated September 12, 1978.

6. "Generic Reload Fuel Application", General Electric Report, NEDE-24011-P-3, dated March 1978.

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

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

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

10. GE letter (Engle) to NRC (Eisenhut), "Fuel Assembly Loading Error" dated June 1, 1977.

11. GE letter (Engle) to NRC (Eisenhut) dated November 30, 1977.

12. NRC letter (Eisenhut) to GE (Engle) dated May 8, 1978.

13. Philadelphia Electric Company letter (Coonay) to NRC (Lear) received July 5, 1977.

14. "Boiling Water Reactor Fuel Rod Performance Evaluation Program" NEDC-23719, October 1977.

15. Philadelphia Electric Company letter (Harkins) to NRC (Lear), dated June 12, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-277

PHILADELPHIA ELECTRIC COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 48 to Facility Operating License No. DPR-44 issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station, Unit No. 2. The amendment is effective as of its date of issuance.

The changes permit operation of Peach Bottom Unit No. 2 with a new type reload fuel (8x8R) for Cycle 4.

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 September 7, 1978 (43 FR 39869). 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 Section 51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 28, 1978, as supplemented September 5, 26, and October 4, 1978, (2) Amendment No. 48 to License No. DPR-44, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16 day of October 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
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