September 16, 1981

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Docket No. 50-278

RECEIVED SEP 22 1981 U.S. NUCLEAR REGULATORY COMMISSION

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

Dear Mr. Bauer:

The Commission has issued the enclosed Amendment No. 79 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit No. 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your applications dated September 30, 1980 and March 30, 1981, as supplemented April 24, June 30 and July 15, 1981.

The changes to the TSs permit reactor operation of Peach Bottom Unit No. 3 with the Reload Number 4 core (Cycle 5). In addition, the maximum average planar linear heat generation rates are extended to 40,000 megawatt days per short ton of uranium for all the Cycle 5 fuel in Unit No. 3.

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

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"
John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 79 to DPR-56
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

No legal objection to
Notice & action et
amendment. SER
not reviewed.

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SURNAME	RIngram	MFairtile	JStolz	TNovak	Cutchin		
DATE	9/4/81	9/8/81:cb	9/14/81	9/ /81	9/16/81		

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

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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. 79
License No. DPR-56

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

   A. The applications for amendment by Philadelphia Electric Company, et al. (the licensee) dated September 30, 1980, and March 30, 1981, as supplemented April 24, June 30 and July 15, 1981, comply 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 applications, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.79 , are hereby incorporated in the license. PECo shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile for

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 16, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 79

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.

Remove Pages	Insert Pages
iv	iv
vi	vi
4	4
9	9
10	10
11a	11a
13	13
14	14
17	17
19	19
20	20
21	21
24	24
31	31
33	33
40	40
74	74
91	91
92	92
103	103
104	104
111	111
133a thru 133d	133a thru 133d
--	133e
140	140
140a thru 140e	140a thru 140d
142	142
142a thru 142c	142a thru 142c
142e thru 142g	142e thru 142g
--	142h
--	142i
152a	152a
157	157
241	241

PBAPS
Unit 3

LIST OF FIGURES

LIST OF TABLES

1.0 DEFINITIONS (Cont'd)

component, or device to perform its function are also capable of performing their related support function.

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

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

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

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

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

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

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

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objectives:

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

Specification:

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

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

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objectives:

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

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

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

$$S \leq 0.66W + 54\% - 0.66\ \Delta W$$

where:

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

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

SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING

2.1.A (Cont'd)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows.

$$S \leq (0.66\ W + 54\% - 0.66\ \Delta W)\left(\frac{FRP}{MFLPD}\right)$$

where,

FRP = fraction of rated thermal power (3293 MWt)

MFLPD = maximum fraction of limiting power density where the limiting Power density is 13.4 KW/ft for all 8x8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

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)	B. APRM Rod Block Trip Setting $SRB \leq (0.66\ W + 42\% - 0.66\ \Delta W) \frac{(FRP)}{MFLPD}$ where: FRP = fraction of rated thermal power (3293 MWt). MFLPD = maximum fraction of limiting power density where the limiting Power density is 13.4 KW/ft for all 8x8 fuel. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.
C. 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)

1.1 BASES: FUEL CLADDING INTEGRITY

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

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

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis described in references 1 and 3.

1.1.A BASES (Cont'd)

B. Core Thermal Power Limit (Reactor Pressure < 800 psia or 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 in 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 factors this 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.

2.1 BASES: 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. 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 analysis of fast pressurization transients as described in reference 3. Conservatism is incorporated in all other 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 models. These transient models evolved over many years, have 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 and results are summarized in Reference 1 for cold water events, and in Reference 2 for pressurization events.

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 rod. 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.

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

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

The scram trip setting must be adjusted to assure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the MFLPD is greater than the fraction of rated power (FRP).

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than the fuel cladding integrity safety limit when the transient is initiated from MCPR greater than the operating limit given in Specification 3.5.K.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the Safety Limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the Rod Worth Minimizer and Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of 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 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

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

B. APRM Rod Block Trip Setting

The APRM system provides a control rod block to avoid conditions which would result in an APRM scram trip if allowed to proceed. The APRM rod block trip setting, like the APRM scram trip setting, is automatically varied with recirculation loop flow rate. The flow variable APRM rod block trip setting provides margin to the APRM scram trip setting over the entire recirculation flow range. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

2.1 BASES (Cont'd.)

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

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

D. Turbine Stop Valve Closure Scram

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

E. Turbine Control Valve Scram

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

2.1 BASES (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973.

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO 24154 and NEDE 24154-P, Volumes I, II, and III.

3. "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors NEDO-24154 and NEDE 24154-P, Volumes I, II, and III.

1.2 BASES

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

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

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

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

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

The details of the analysis which 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 Peach Bottom Unit 3. The analysis of the worst overpressure transient, is provided in the Supplemental Reload Licensing Safety Evaluation and demonstrates margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient is provided in the Supplemental Reload Licensing Safety Evaluation and demonstrates that the safety valves will not open.

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

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

NOTES FOR TABLE 3.1.1 (Cont'd)

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

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

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), where:

    FRP = fraction of rated thermal power (3293 MWt).
    MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for all 8x8 fuel.

    The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

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

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

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

13. See Section 2.1.A.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. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP) where:

   FRP = fraction of rated thermal power (3293 MWt)

   MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for all 8x8 fuel.

   The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

   W = Loop Recirculation flow in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.
   Trip level setting is in percent of rated power (3293 MWt).

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

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

4. This function is bypassed when the count rate is ≥ 100 cps.

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

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

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

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

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 the fuel cladding integrity safety limit. The trip logic for this function is 1 out of n: e.g., any trip on one of 6 APRM's, 8 IRM's, or 4 SRM's will result in a rod block.

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

3.2 BASES (Cont'd)

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

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

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

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

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

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

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3.B <u>Control Rods</u> (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 the fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of a single operable control rod.

C. <u>Scram Insertion Times</u>

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:

% Inserted from Fully Withdrawn	Avg.Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.5

4.3.B <u>Control Rods</u> (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. <u>Scram Insertion Times</u>

1. After each refueling outage, and prior to synchronizing the main turbine generator initially following restart of the plant, 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.

LIMITING CONDITIONS FOR OPERATION

3.3.C (Cont'd)

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.398
20	0.954
50	2.120
90	3.8

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

SURVEILLANCE REQUIREMENTS

4.3.C (Cont'd)

After exceeding 30 percent power all previously untested operable control rods shall be tested as described above prior to exceeding 40 percent power.

2. Whenever such scram time measurements are made (such as when a scram occurs and the scram insertion time recorders are operable) an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained

3.3 and 4.3 <u>BASES</u> (Cont'd)

C. <u>Scram Insertion Times</u>

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the fuel cladding integrity safety limit. Analysis of the limiting power transients shows that the negative reactivity rates resulting from the scram with the average response to all drives as given in the above Specification, provide the required protection.

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

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

In the analytical treatment of the transients, which are assumed to scram on high neutron flux, 290 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 typical time delay of about 210 milliseconds estimated from scram test results. The 290 milliseconds used in the analyses consists of 90 milliseconds for sensor and circuit delay and 200 milliseconds to start of control rod motion. 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

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 the applicable figures during two recirculation loop operation. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multipled by the following reduction factors: 0.71 for 7X7 fuel; 0.83 for 8X8 fuel; 0.81 for PTA, 8X8R and P8X8R fuel. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.J Local LHGR

During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed design LHGR.

LHGR≤LHGRd

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

SURVEILLANCE REQUIREMENTS

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 ≥25% rated thermal power

4.5.J Local LHGR

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

LIMITING CONDITIONS FOR OPERATION

3.5.J Local LHGR (Cont'd)

If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K.1 Minimum Critical Power Ratio (MCPR)

1. During power operation, the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Specification 3.5.K.2 or 3.5.K.3 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.

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at ≥25% rated thermal power.
2. Except as provided in Specification 3.5.K.3, the verification of the applicability of 3.5.K.2.a Operating Limit MCPR Values shall be performed every 120 operating days by scram time testing 19 or more control rods on a rotating basis and performing the following:

   a. The average scram time to the 20% insertion position shall be:
   $\tau_{ave} \leq \tau_B$

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

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

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

LIMITING CONDITIONS FOR OPERATION

3.5.K. Minimum Critical Power Ratio (MCPR) (Cont'd)

2. Except as specified in 3.5.K.3, the Operating Limit MCPR Values are as follows:

   a. If requirement 4.5.K.2.a is met:
      The Operating Limit MCPR values are as given in Table 3.5.K.2

   b. If requirement 4.5.K.2.a is not met:
      The Operating Limit MCPR values as a function of τ are as given in Figures 3.5.K.1 and 3.5.K.2.

   Where:

$$\tau = \frac{\tau_{ave} - \tau_B}{0.90 - \tau_B}$$

3. The Operating Limit MCPR values shall be as given in Table 3.5.K.3 if the Surveillance Requirement of Section 4.5.K.2 to scram time test control rods is not performed

SURVEILLANCE REQUIREMENTS

4.5.K. Minimum Critical Power Ratio (MCPR) (Cont'd)

N_i = number of active control rods measured in the ith surveillance test.

τ_i = average scram time to the 20% insertion position of all rods measured in the ith surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{1/2} \sigma$$

Where:

μ = mean of the distribution for average scram insertion time to the 20% position = 0.710 sec

N_1 = total number of active control rods measured in specification 4.3.C.1

σ = standard deviation of the distribution for average scram insertion time to the 20% position = 0.053.

Table 3.5.K.2

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

Fuel Type	MCPR Operating Limit** For Incremental Cycle Core Average Exposure	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
8x8	1.24	1.27
PTA &P 8X8R	1.25	1.30
8x8R	1.24	1.27

* If requirement 4.5.K.2.a is met.

** These valves shall be increased by 0.01 for single loop operation.

Table 3.5.K.3

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

Fuel Type	MCPR Operating Limit ** For Incremental Cycle Core Average Exposure	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
8x8	1.33	1.39
PTA &P 8X8R	1.36	1.42
8x8R	1.33	1.39

* If surveillance requirement 4.5.K.2 is not performed.

** These valves shall be increased by 0.01 for single loop operation.

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 the LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is as shown in the applicable figures for each fuel type.

The calculational procedure used to establish the APLHGR for each fuel type 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.K <u>BASES</u> (Cont'd.)

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

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.

3.5.K BASES(Cont'd.)

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

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

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

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

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

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

3.5.L BASES(Cont'd.)

Operating experience has demonstrated that a calculated value of 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.

3.5.M. References

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

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

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

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

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

6. DELETED

7. General Electric Boiling Water Reactor Generic Reload Fuel Application. NEDO-24011-P-A.

8. Loss-of-Coolant Accident Analysis For Peach Bottom Atomic Power Station Unit 3, NEDO-24082, December 1977.

Figure 3.5.K.1 MCPR Operating Limit Vs. τ
8X8 & 8X8R Fuel

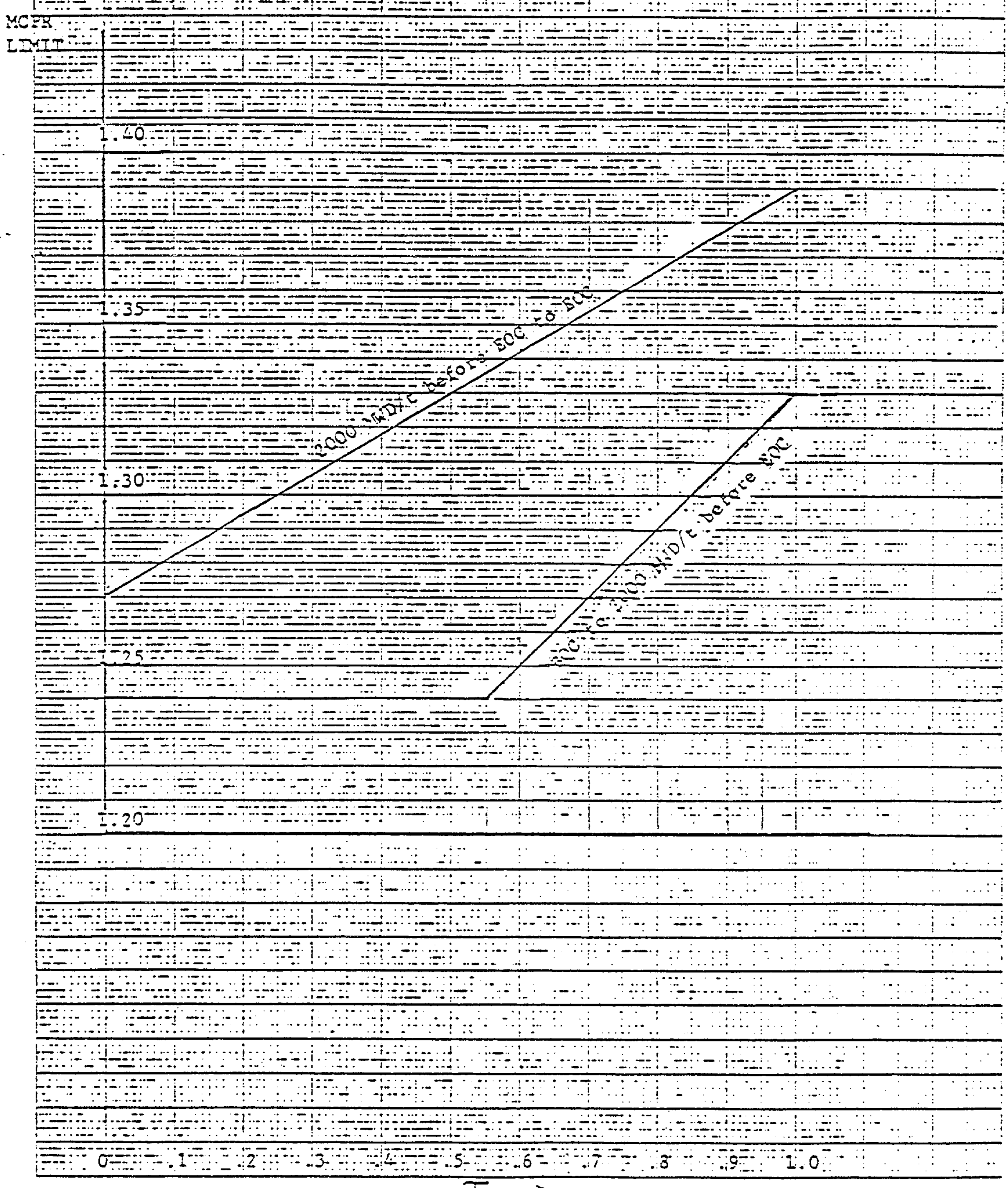
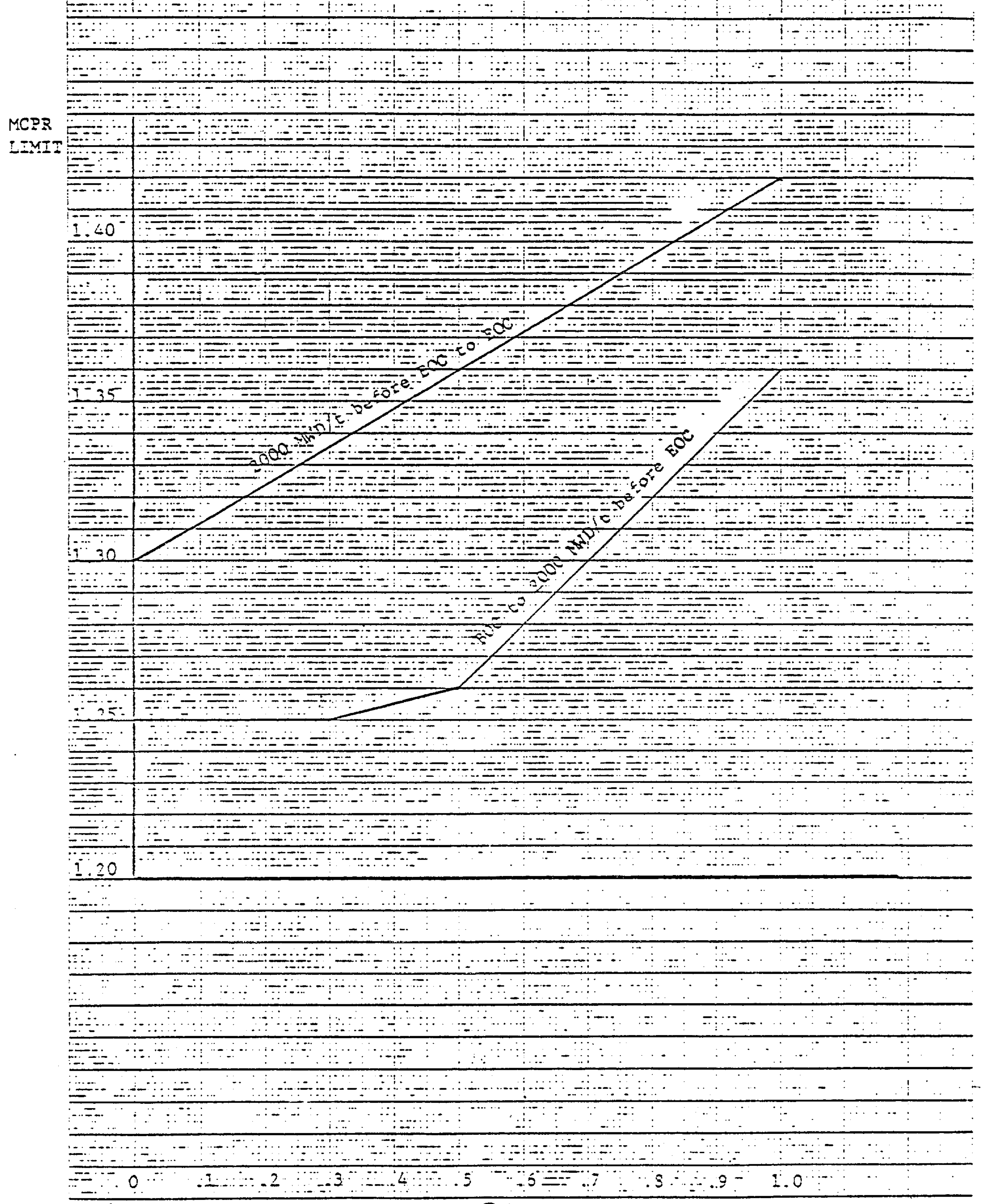


Figure 3.5.K.2 - MCPR Operating Limit Vs. τ

P 8X8R & P1A fuel



8X8 Fuel, Type H

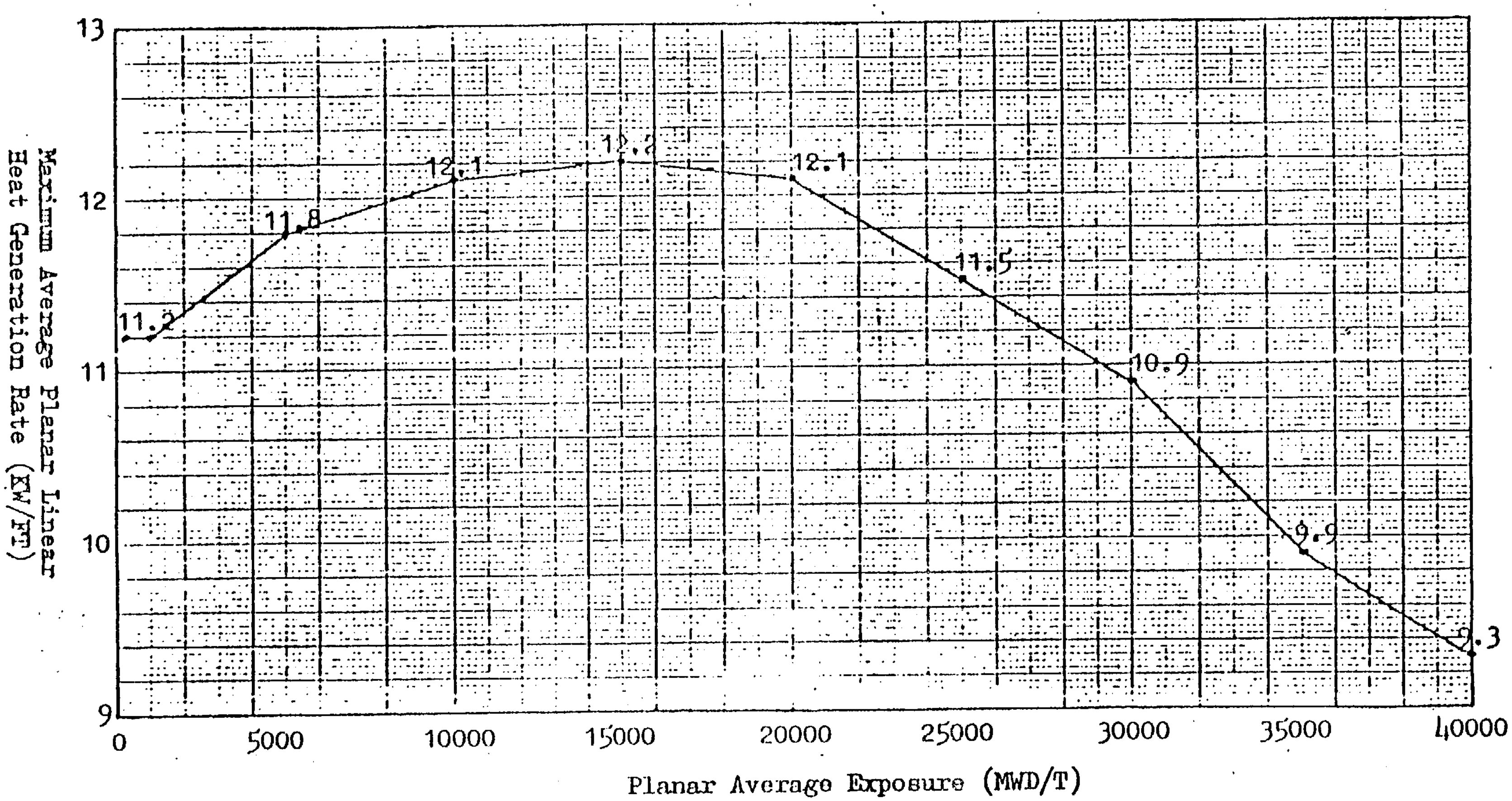


Figure 3.5.1.C Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

The average level of irradiation of the irradiated fuel from the reactor shall not exceed 33,000 MWD/MTU in accordance with Tables S-3 and S-4 of 10 CFR Part 51.

8X8 Fuel, Type L

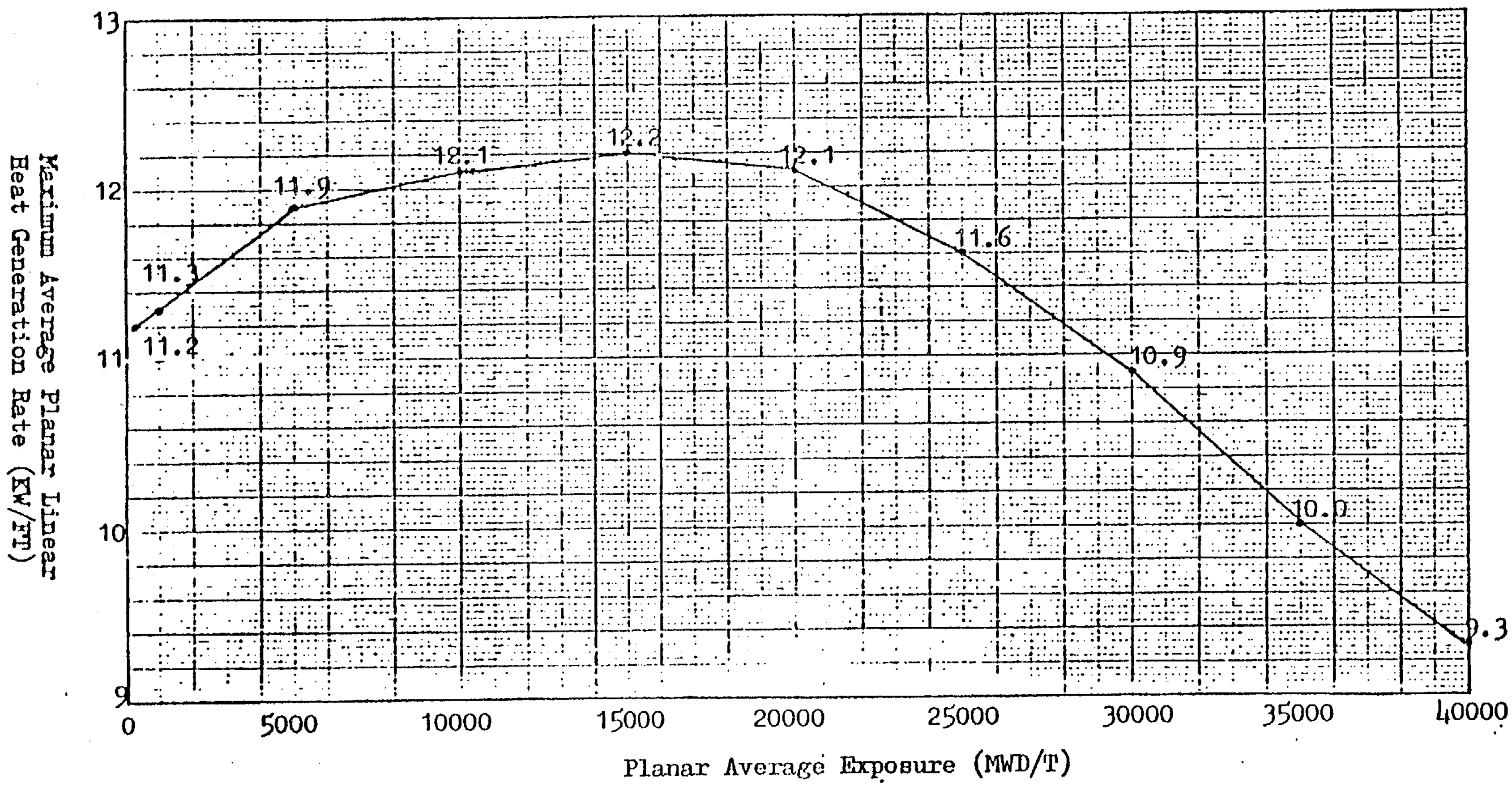


Figure 3.5.1.D Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

The average level of irradiation of the irradiated fuel from the reactor shall not exceed 33,000 MWD/MTU in accordance with Tables S-3 and S-4 of 10 CFR Part 51.

PEACH BOTTOM UNIT 3

8X8 PTA Fuel

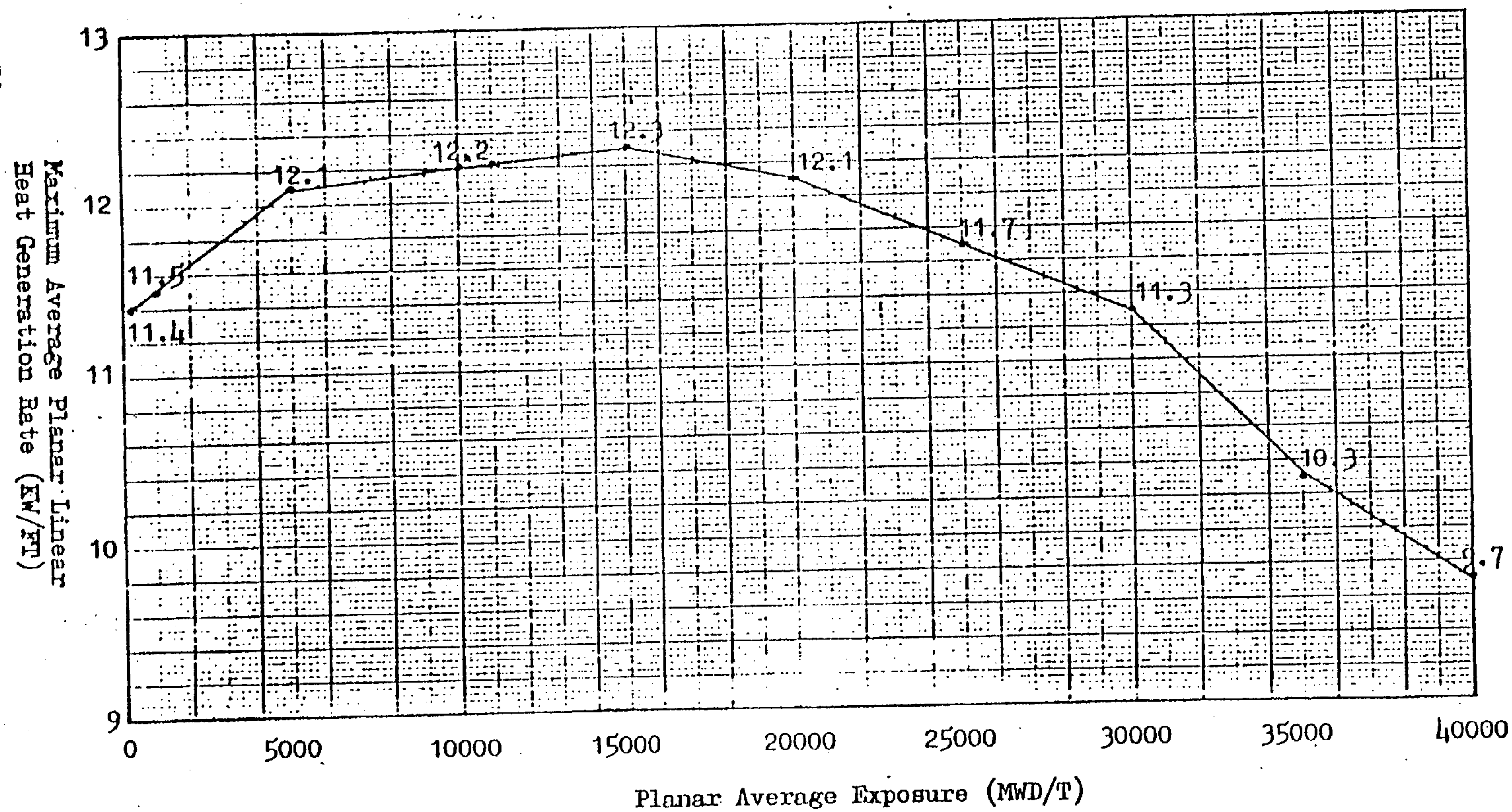
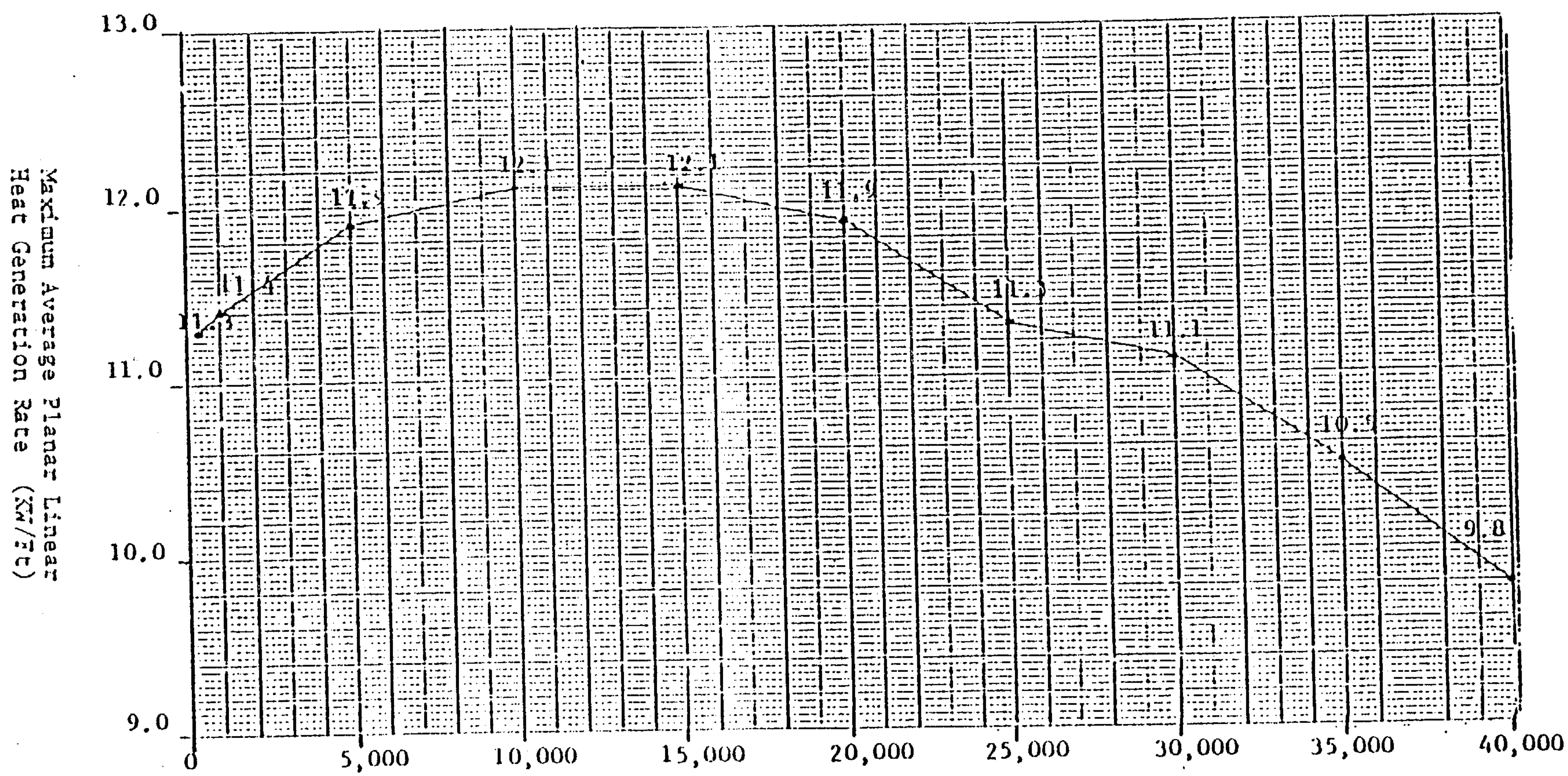


Figure 3.5.1.F Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

The average level of irradiation of the irradiated fuel from the reactor shall not exceed 33,000 MWD/MTU in accordance with Tables S-3 and S-4 of 10 CFR Part 51.

PEACH BOTTOM UNIT 3

8 X 8R FUEL



PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.G MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

The average level of irradiation of the irradiated fuel from the reactor shall not exceed 33,000 MWD/MTU in accordance with Tables S-3 and S-4 of 10 CFR Part 51.

PEACH BOTTOM UNIT 3

P8 x 8R FUEL, TYPE P8DRB284H

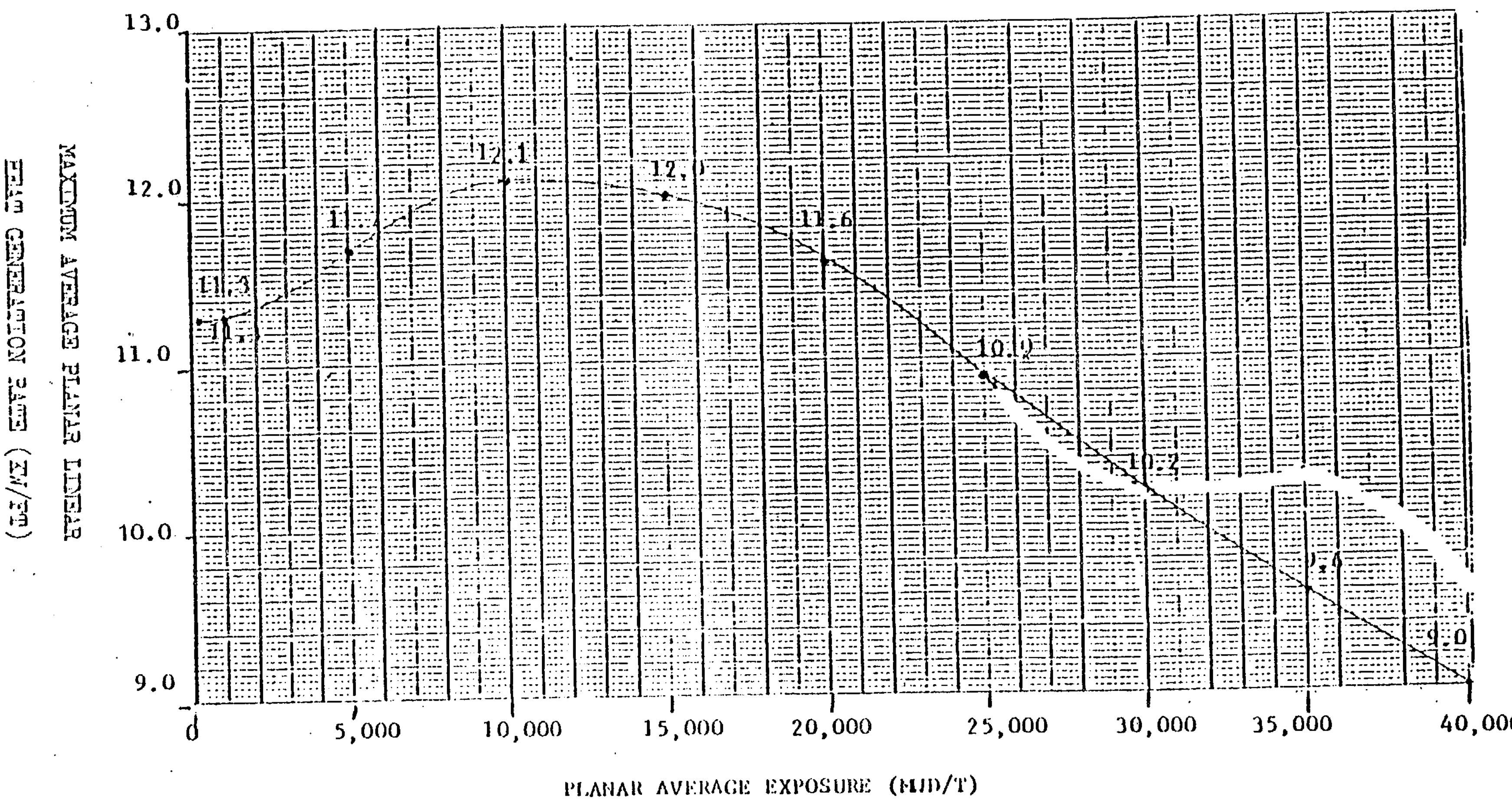


FIGURE 3.5.1.H MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

The average level of irradiation of the irradiated fuel from the reactor shall not exceed 33,000 MWD/MTU in accordance with Tables S-3 and S-4 of 10 CFR Part 51.

PEACH BOTTOM UNIT 3

P 8X8R FUEL, TYPE P8DRB299

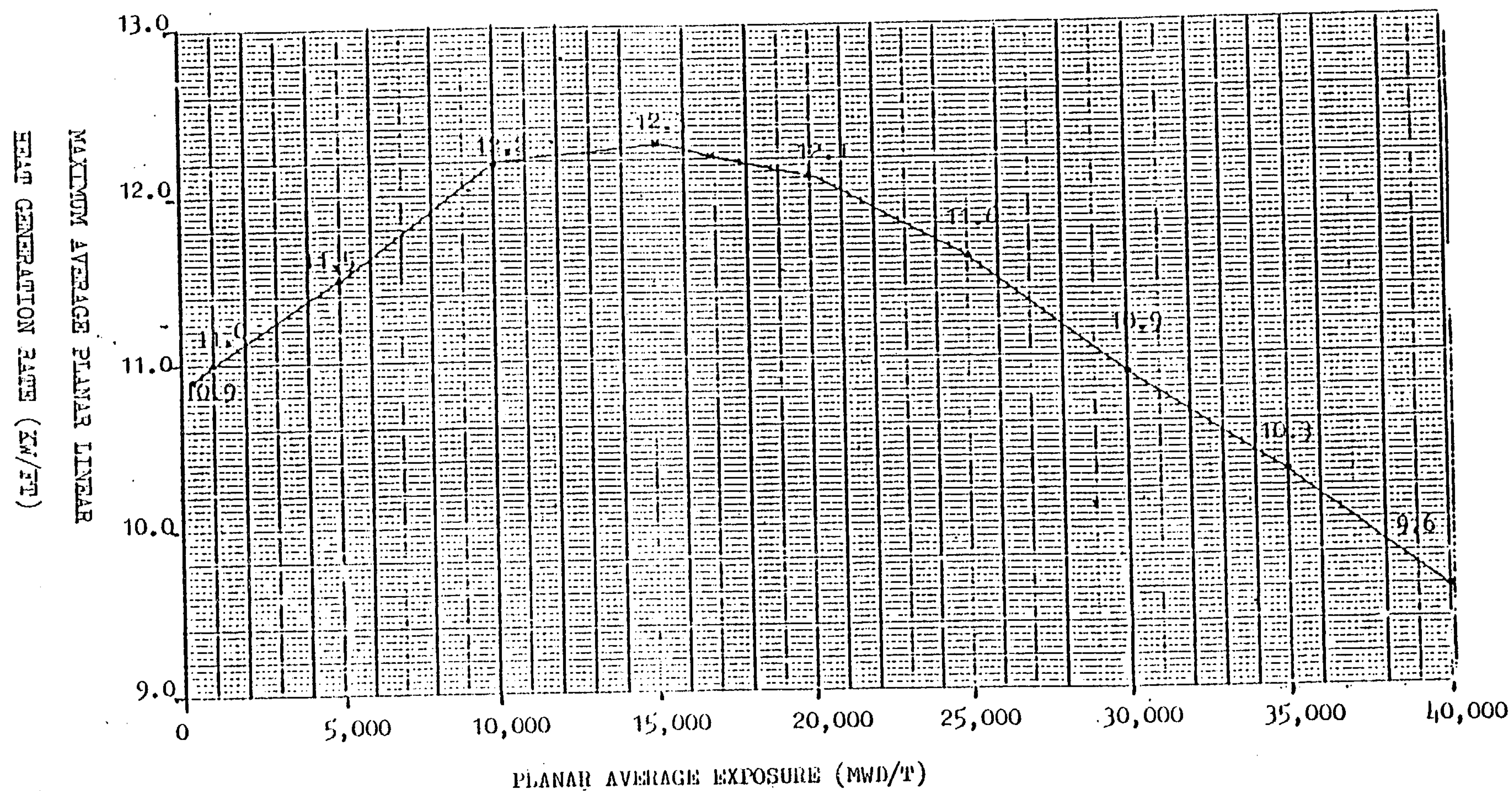


FIGURE 3.5.1.I MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE VERSUS PLANAR AVERAGE EXPOSURE

The average level of irradiation of the irradiated fuel from the reactor shall not exceed 33,000 MWD/MTU in accordance with Tables S-3 and S-4 of 10 CFR Part 51.

PEACH BOTTOM UNIT 3

P 8X8R GENERIC

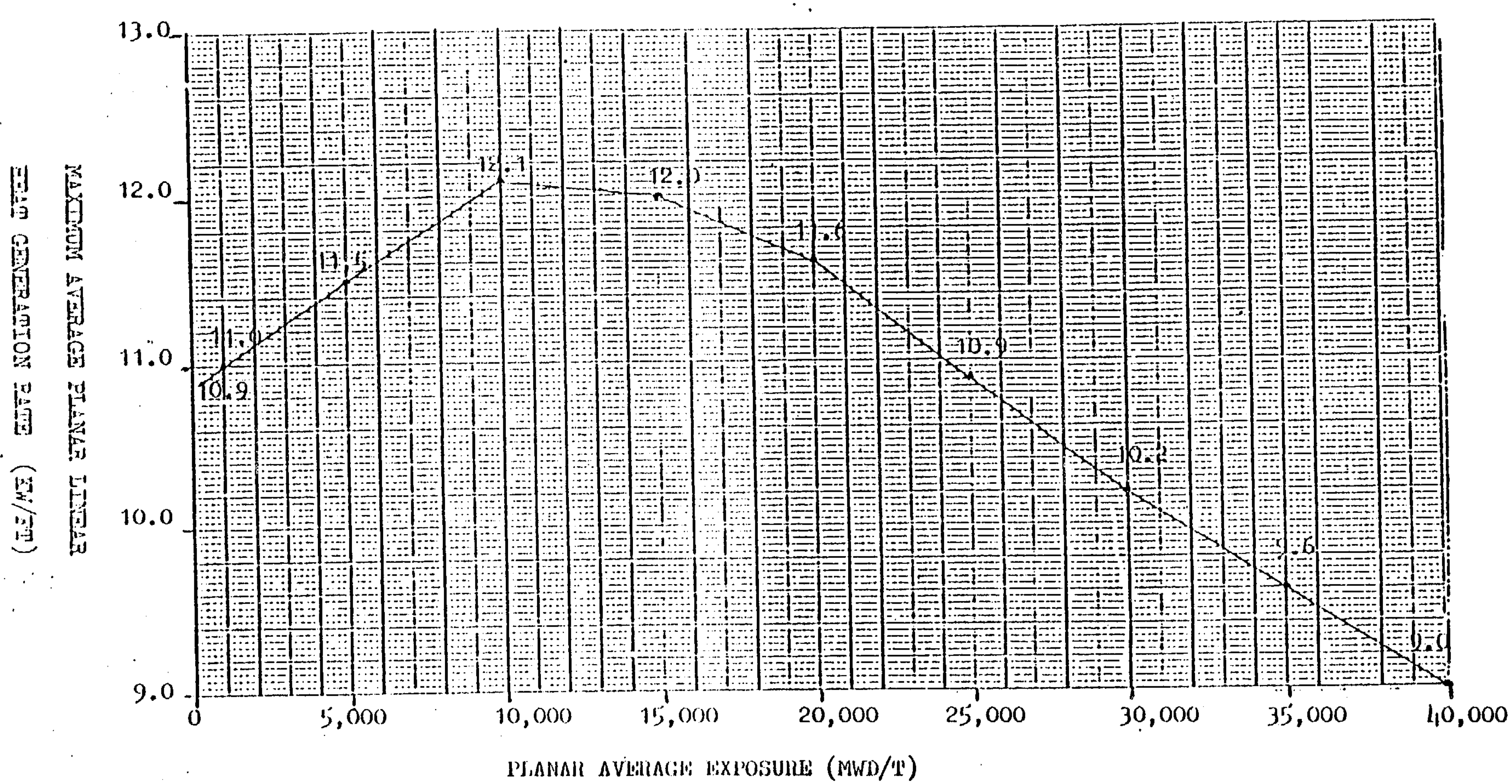


FIGURE 3.5.1.J MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS PLANAR AVERAGE EXPOSURE

The average level of irradiation of the irradiated fuel from the reactor shall not exceed 33,000 MWD/MTU in accordance with Tables S-3 and S-4 of 10 CFR Part 51.

3.6.A & 4.6.A. Bases (Cont'd)

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figure 3.6.1, 3.6.2, or 3.6.3 in conjunction with Figure 3.6.4. Note: Figure 3.6.3 includes an additional 40°F margin required by 10 CFR 50 Appendix G.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be removed and tested to experimentally verify the values used for Figure 3.6.4.

As described in paragraph 4.2.5 of the Safety Analysis report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The plant safety analyses (Ref: NEDE-24011-P-A) state that all MSIV valve closure - Flux scram is the event which satisfies the ASME Boiler and Pressure Code requirements for protection from the consequences of pressure in excess of the vessel design pressure. The reactor vessel pressure code limit of 1375 psig, given in Subsection 4.2 of the FSAR, is well above the peak pressure produced by the above overpressure event.

3.6.D & 4.6.D <u>BASES</u>

<u>Safety and Relief Valves</u>

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

The details of the analysis which 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 3 with a total capacity of 79.51% of rated steam flow. The analysis of the worst overpressure transient demonstrates margin to the code allowable overpressure limit of 1375 psig.

To meet the power generation design basis, the total pressure relief system capacity of 79.51% has been divided into 65.96% safety/relief (11 valves) and 13.55% safety (2 valves). The analysis of the plant isolation transient shows that the 11 safety/relief valves limit pressure at the safety valves below the setting of the safety valves. Therefore, the safety valves will not open.

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

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

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

5.2 REACTOR

A. The core shall consist of not more than 764 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.

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 during operation.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. DPR-56

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

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

DOCKET NO. 50-278

Introduction

By letter dated September 30, 1980, Philadelphia Electric Company (PECo or the licensee) made application to modify the Technical Specifications (TSs) for Peach Bottom Atomic Power Station, Units 2 and 3, to permit an extension of the maximum average planar linear heat generation rate (MAPLHGR) from 30,000 to 40,000 megawatt days per short ton of uranium (MWd/T). By letter dated May 20, 1981, we issued TSs extending the MAPLHGR for Unit 2 only. This license amendment evaluates the requested change for Unit 3.

In addition, by letter dated March 30, 1981, as supplemented April 24, June 30 and July 15, 1981, the licensee made application to modify the TSs for Peach Bottom Unit 3 to permit operation with the reload number 4 core (Cycle 5).

Evaluation

Thermal-Hydraulic Design

Peach Bottom Unit 3 Reload 4 consists of 216 new P8x8R fuel bundles which have drilled lower tie plates and finger springs to regulate bypass flow. This makes a total of 764 bundles with drilled lower tie plates for Reload 4 (or Cycle 5). Reload 4 has a total of 489 P8x8R fuel bundles, 252 8x8R fuel bundles and 23 8x8 fuel bundles. Assumed cycle exposure is increased from 17,160 MWd/T (Reload 3) to 18,208 MWd/T (Reload 4). Also, for operational flexibility, Minimum Critical Power Ratio (MCPR) operating limits with different options were provided in the proposed TSs. Our review consisted of the following: (a) Fuel Cladding Integrity Safety Limit, (b) Operating Limit MCPRs (OLMCPRs), (c) Thermal-Hydraulic Stability, and (d) TS modifications.

The objective of the review is to confirm that the thermal-hydraulic design of the reload core has been accomplished using acceptable methods, that it provides an acceptable margin of safety from conditions which would lead to fuel damage during normal operation and anticipated operational transients, and is not susceptible to thermal-hydraulic instability.

Fuel Cladding Integrity Safety Limit

As stated in Ref. 4, the minimum allowable critical power ratio for core-wide or localized transients is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid boiling transition. There has been no change in the safety limit MCPR for Peach Bottom Unit 3 from Cycle 4 to Cycle 5.

OLMCPRs

Various transients could serve to reduce the actual MCPR below the intended safety limit MCPR (SLMCPR) during Cycle 5 operation. The most limiting of these operational transients have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR).

The transients evaluated were the generator load rejection without bypass, the feedwater controller failure, loss of 100°F feedwater heating, the control rod withdrawal error and the fuel loading error. Transients were analyzed on the basis of the initial conditions given in Section 6 of Ref. 2. The initial MCPR assumed in the calculation of the ΔCPR for the generator load rejection without bypass at End of Cycle (EOC)-2000 MWd/T for the PTA/P8x8R fuel is 0.01 below the OLMCPR. The licensee has provided justification for this assumption in Ref. 7. This is acceptable to the NRC staff. The ΔCPR values given in Section 9 (Ref. 2) are plant-specific deterministic values calculated by using the ODYN transient code (Ref. 6). The value of ΔCPRs for the same fuel types (8x8 and 8x8R) for Cycle 5 is 0.26 compared to 0.23 for Cycle 4, and for P8x8R and PTA fuel for Cycle 5 is 0.29 compared to 0.25 for Cycle 4. This difference is due to the use of the (Ref. 8) ODYN transient code compared to the REDY code used in Cycle 4. We have evaluated the ODYN code and found it acceptable for transient analyses of the Cycle 5 core.

Fuel Loading Error ΔCPRs

The licensee stated (Ref. 2) that the mislocated bundle loading error event analysis will no longer be reported for each cycle as per Ref. 5. We have accepted this for current Peach Bottom reloads (Ref. 5a). The licensee has done the rotated bundle loading error event analysis based on the new analysis procedure described in Ref. 6. Analysis shows that the rotated bundle results in a MCPR greater than the safety limit of 1.07, and we find this analysis acceptable.

Rod Withdrawal Error (RWE) ΔCPRs

RWE ΔCPRs given in Section 10 of Ref. 2 were calculated using previously approved methods (Ref. 4). The ΔCPR at the Rod Block Monitor (RBM) setpoint of 107% is 0.20 for Cycle 5 compared to 0.13 for Cycle 4. This difference is due to less P8x8R fuel loaded in Cycle 5 compared to Cycle 4 and a different loading pattern (Ref. 2) in Cycle 5. RWE ΔCPR is not limiting in Cycle 5. Therefore, the RWE ΔCPR analysis is acceptable to the NRC staff.

Establishing OLMCPRs

The ΔCPRs calculated above were adjusted to reflect either "Option A" or "Option B" ΔCPRs by employing the conversion method described in Ref. 10. These adjustments are based on conservative factors. The MCPR for the event is determined by adding the ΔCPR to the safety limit. Section 11 (Ref. 2) presents both the MCPRs for the non-pressurization events as well as the adjusted MCPRs (Option A and Option B) for the pressurization events.

a) MCPRs were adjusted for Option B for all plants choosing to operate under Option B which meet all scram specifications given in Ref. 9.

b) MCPRs were determined by a linear interpolation between the Option A MCPR and the Option B MCPR for all plants choosing to operate under Option B which do not meet the scram time specification. This interpolation is based on the tested measured scram time and is described in Ref. 9.

c) MCPRs were adjusted for Option A according to Ref. 9. This option is to be used if the surveillance requirement of the TSs to scram time test control rods is not performed.

We have reviewed all the OLMCPR results discussed above. These results are consistent with the previous Cycle 4 analysis and are more conservative for Cycle 5 than Cycle 4; therefore we find these results acceptable.

Thermal-Hydraulic Stability

The results of the thermal-hydraulic stability analysis (Ref. 2) show that the channel hydrodynamic and reactor core stability decay ratios at the natural circulation-105% rod line intersection are below the stability limit. Decay ratio for Cycle 5 was 0.87 as compared to 0.90 for Cycle 4. Because the operation in the natural circulation mode will be prohibited by the TSs, there will be added margin to the stability limit and we conclude this is acceptable.

Evaluation of TS Changes

The licensee has submitted proposed changes to the Peach Bottom Unit 3 TSs (Ref. 3). These changes: (1) identify the operating limits for all fuel types for Cycle 5 operation, (2) incorporate MAPLHGR limits for the Reload 4 fuel and extended exposure MAPLHGR limits for the Reload 2 and Reload 3 fuel, (3) add a generic MAPLHGR curve for P8x8R fuel to reduce need for future cycle-dependent revisions.

OLMCPR TSs For All Fuel Types

Based on our thermal-hydraulic design evaluation in this Safety Evaluation, changes in the TSs are found to be acceptable except that we have modified TS Figure 3.5.K.1, Page 142 (Ref. 3) MCPR operating limit vs. τ for 8x8 and 8x8R fuel. This modification makes the MCPR operating limit more conservative. This change was agreed to by the licensee.

MAPLHGR Limit TS Curves

For all fuel types the licensee proposed to extend the burnup time from 30,000 to 40,000 MWd/T. The licensee has stated (Ref. 13) that they comply with General Electric letter (Ref. 12) for the MAPLHGR limits. Therefore, as stated in Ref. 11, we find the proposed extended exposure MAPLHGR limits for the Reload 2 and Reload 3 fuel acceptable. The licensee's proposed MAPLHGR limits for the Reload 4 fuel and the generic MAPLHGR curve for the P8x8R fuel to reduce the need for future cycle dependent revisions have been done with currently approved methods and are in compliance with Ref. 12; therefore, we conclude these revised curves are acceptable.

Change in Control Rod Scram Time TS

This proposed TS change calls for a 3.5 second average scram insertion time, rather than 5.0 second average scram insertion time for the 90% rod insertion from the fully withdrawn position. This change is in conformance with the Cycle 5 unique transient analysis input presented in Reference 2.

Our review of the reduction in average scram insertion times indicates that there is no effect on the course of previously analyzed transients. Based on both the current use of these scram times in the General Electric Standard TSs and the lack of change on transient results, we conclude that this change is acceptable.

Evaluation of Fast Scram Control Rod Drive (FSCRD) Program During Cycle 5

In order to assist General Electric Corporation in developing a control rod drive system for the BWR/6 design, PECo has been using a single FSCRD in Unit 3 starting in Cycle 2. We have previously evaluated the use of this drive and found it acceptable. In order to accumulate long term exposure of the drive in an operating reactor environment, the licensee proposed to extend its use through Cycle 5. Previous operating experience has been favorable. The original drive, used in Cycle 2, has been removed, disassembled and inspected. The inspection provided support for continued use of a FSCRD in Unit 3 through Cycle 5. The current FSCRD was installed during Cycle 3. These drives have no effect on the parameters used in the safety analyses. We conclude, based on the above, that continued use of a FSCRD in Unit 3 during Cycle 5 is acceptable.

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.

There is, however, an environmental consideration related to the amendment. 10 CFR 51.20g(2)(iii) states, in part, "The average level of irradiation of the irradiated fuel from the reactor does not exceed 33,000 megawatt days per metric ton and...". The TS curves specify burnup in megawatt days per short ton. A short ton is 2,000 pounds and a metric ton is 2,205 pounds, thus a metric ton is 1.1 times greater than a short ton.

In a previous Safety Evaluation performed for the Browns Ferry Nuclear Plant, Units Nos. 1 and 2, dated October 6, 1980, we extended the irradiation to 40,000 megawatt days per short ton. This is the same request made by PECo for Peach Bottom Unit No. 3 in this amendment. We found that the Browns Ferry fuel when irradiated to 40,000 megawatt days per short ton did not exceed an average level of burnup of 33,000 megawatt days per metric ton. The Peach Bottom 3 fuel is bounded by the evaluation done for the Browns Ferry fuel. We conclude, based on the bounding Browns Ferry analysis, that the proposed burnups to 40,000 MWD/Short Ton do not exceed the 10 CFR Part 51.20 limits of 33,000 MWD/Metric Ton.

To ensure that the environmental considerations in 10 CFR 51 are evaluated, if MAPLHGR limits are extended in the future, we have, with the licensee's concurrence, added a note to the TS Figures related to MAPLHGR limits, stating the requirement of 10 CFR 51.20.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 16, 1981

References

1. Letter, E. J. Bradley (PECo) to H. R. Denton (NRC), dated March 30, 1981.

2. Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 3, Reload No. 4, Y1003J01A20, February 1981.

3. "Proposed Technical Specification Changes", along with technical descriptions and bases for such changes, March 30, 1981, enclosure of Ref. 1.

3a. Letter, S. L. Daltroff (PECo) to J. F. Stolz (NRC), dated June 30, 1981.

4. "General Electric Boiling Water Reactor Generic Reload Fuel Application", July 1979, (NEDE-24011-P-A-1).

5. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Change in General Electric Methods for Analysis of Mislocated Bundle Accident", November 14, 1980.

5a. Letter, W. V. Johnston to T. A. Ippolito, "Change in General Electric Analysis of Mislocated Bundle Accident", dated April 14, 1981.

6. Safety Evaluation Report (letter), D. G. Eisenhut (NRC) to R. E. Engel (GE), MFN-200-78, dated May 3, 1978.

7. "General Electric Boiling Water Reactor Generic Reload Fuel Application", January 1979 (NEDE-24011-P-A, Amendment 2).

8. Letter, R. P. Denise (NRC) to G. G. Sherwood (GE), January 23, 1980.

9. Letter (with attachment), R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model", September 5, 1980.

10. Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "ODYN Adjustment Method for Determination of Operating Limits", January 19, 1981.

11. Letter, L. S. Rubenstein (NRC) to T. M. Novak (NRC), "Extension of General Electric Emergency Core Cooling System Performance Limits", dated June 25, 1981.

12. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Additional Information Regarding Extension of Emergency Core Cooling System Performance Limits", dated May 28, 1981.

13. Letter, PECo to NRC dated June 30, 1981.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-278

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. 79 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 Technical Specifications for operation of the Peach Bottom Atomic Power Station, Unit No. 3 (the facility) located in York County, Pennsylvania. The amendment is effective as of its date of issuance.

The revised Technical Specifications permit Cycle 5 operation of the facility with the Reload No. 4 core, and extend the maximum average planar linear heat generation rate for the Cycle 5 fuel from 30,000 megawatt days per short ton of uranium (MWd/T) to 40,000 MWd/T.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated September 30, 1980 and March 30, 1981, as supplemented April 24, June 30, and July 15, 1981, (2) Amendment No. 79 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 Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 16th day of September 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile

Morton B. Fairtile, Acting Branch Chief
Operating Reactors Branch #4
Division of Licensing