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BJones (4)
BScharf (15)
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JRBuchanan
OPA (Clare Miles)
Gray files
extra cps (5)

6/11/76

Docket No. 50-277

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

Gentlemen:

In response to your requests dated March 22, 1976 and May 13, 1976, with supplement dated May 7, 1976, the Commission has issued the enclosed Amendment No. 23 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station Unit No. 2.

The amendment consists of changes in the Technical Specifications that authorize operation with (1) up to 188 GE 8 x 8 reload fuel assemblies, (2) four Lead Test Assemblies, (3) twelve developmental fuel channels, (4) holes drilled in the lower tieplate of all reload 8 x 8 fuel bundles, and (5) a modified rod sequence control system. Additionally, Technical Specifications related to the Core Spray and Low Pressure Coolant Injection (LPCI) Systems have been modified to reflect the additional reflood credit given for the LPCI system modification approved for Peach Bottom Unit No. 2 in Amendment No. 15 to the license.

We request that you submit a summary report of your findings concerning the Lead Test Assemblies and developmental channels at the next refueling outage.

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

Sincerely,

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

Enclosures:
1. Amendment No. 23 to License No. DPR-44
2. Safety Evaluation
3. Federal Register Notice

9

SEE PREVIOUS PAGE (YELLOW) FOR CONCURRENCE CHAIN

OFFICE ➤	ORB#3 See next page	ORB#3	OELD	ORB#3	DOR:ad/ORs	
SURNAME ➤	CParrish	EVerdery:kmb	SSohinki	GLear	KRGoller	
DATE ➤	6/ /76	6/ 11 /76	6/ /76	6/ 11 /76	6/ /76	

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

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Docket No. 50-277

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

Sincerely,

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

Enclosures:
1. Amendment No. 23 to License No. DPR-44
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

OFFICE➤	ORB#3	ORB#3	OELD SM	ORB#3 DE	DOR:AD/ORs
SURNAME➤	CParrish CP	EVerdery:kmb	5-50H	GLear	KRGoller KRG
DATE➤	6/3/76	6/3/76	6/9/76	6/10/76	6/11/76

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

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

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. DPR-44

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

   A. The applications for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company (the licensees), dated March 22, 1976 and May 13, 1976, with supplement dated May 7, 1976, 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 application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   E. After weighing the environmental aspects involved, 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: June 11, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 23

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace pages 10, 11, 14, 15, 15a, 15b, 17, 18, 20, 33, 35, 37, 38, 39, 40, 54, 66, 73, 74, 101, 102, 109, 110, 119, 120, 124, 125, 126, 127, 133a, 133b, 134, 139, 140, 140a, 140b, 140c, 141a, 157, 158, 160, 241, and 242 with the attached revised pages. Delete pages 15c and 15d. Add new pages 140d, 142c and 142d. No change has been made on pages 38, 39, 120, 158, and 242.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1.A (cont'd)

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

a. $S \leq (0.66\ W + 54\%)\ \frac{2.63}{\text{MTPF}}$ for 7x7 fuel

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

MTPF = The value of the existing maximum total peaking factor

A = 2.63 for 7x7 fuel and 2.44 for 8x8 fuel. 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.

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.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia) When the reactor pressure is $\leq$ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.	B. APRM Rod Block Trip Setting $S_{RB} \leq 0.66 + 42\%$ where: S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt) W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr). In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified to the more limiting of the 2 values determined by the following: 1. $S_{RB} \leq (0.66\ W + 42\%)\ \frac{2.63}{\text{MTPF for 7x7 fuel}}$ 2. $S_{RB} \leq (0.66\ W + 42\%)\ \frac{2.44}{\text{MTPF for 8x8 fuel}}$ MTPF = The value of the existing maximum total peaking factor A = 2.63 for 7x7 fuel and 2.44 for 8x8 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 17.7 in. above the top of the normal active fuel zone.	C. Scram and isolation--reactor low water level $\geq$538 in. above vessel zero (0" on level instruments) D. Scram--turbine stop $\leq$10 percent valve closure E. Scram--turbine control valve fast closure on loss of control oil pressure. 500 < P < 850 psig.

The required input to the statistical model are the uncertainties listed on Table 1.1-1, the nominal values of the core parameters listed in Table 1.1-2, and the relative assembly power distribution shown in Figure 4-l of Reference 1.

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

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

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

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a 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 bundle power corresponds to a core thermal power of more than 50%. Therefore a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1A or 1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design.

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

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

D. Reactor Water Level (Shutdown Condition)

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

E. References

1. "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal with Partial Installation of the Alternate Flow Path for Peach Bottom Atomic Power Station Unit No. 2, License No. DPR-44, Docket No. 50-277," NEDO-21172, Revision 1, March 1976.

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

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

Table 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
Bypass void effect on TIP	3.58 (core midplane) 4.08 (core exit)
R Factor	1.6
Critical Power	3.6

Table 1.1-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

Core Thermal Power	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1078 ft^2
R-Factor	1.098 (7 x 7)
	1.100 (8 x 8)

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

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

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

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

For analyses of the thermal consequences of the transients a MCPR of 1.28 for 7x7 fuel and 1.31 for 8x8 fuel is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady state operation without forced recirculation will not be permitted, 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. <u>Neutron Flux Scram</u>

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

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

The scram trip setting must be adjusted to assure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than 2.63 for 7x7 fuel and 2.44 for 8x8 fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.06 when the transient is initiated from MCPR > 1.28 for 7x7 fuel and > 1.31 for 8x8 fuel.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that

2.1 BASES (Cont'd.)

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.06. This rod block trip setting, which automatically varies with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds 2.63 for 7x7 fuel and 2.44 for 8x8 fuel, thus preserving the APRM rod block safety margin.

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

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

D. Turbine Stop Valve Closure Scram

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

E. Turbine Control Valve Scram

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

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

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

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

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit 2. The analysis of the worst overpressure transient, (3-second closure of all main steamline isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1286 psig for Peach Bottom Unit 2 if a neutron flux scram is assumed. This results in a 89 psig margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in Section 6.3 and Figure 6-7 of NEDO-21172, Revision 1 for Peach Bottom Unit 2. These analyses show that the 11 relief valves limit pressure at the safety valves to 39 psig below the setting of the safety valves. Therefore, the safety valves will not open.

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

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

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

LIMITING CONDITION FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

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

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

B. 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 2.63 for 7x7 fuel and 2.44 for 8x8 fuel.

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

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

Table 3.1.1 (Cont'd.)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

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

NOTES FOR TABLE 3.1.1

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

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

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

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

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

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

3. Bypassed when reactor pressure is < 600 psig.

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

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

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

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

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

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

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

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 = 2.63 for 7x7 fuel and 2.44 for 8x8 fuel. MTPF is the value of the existing maximum total peaking factor.

13. See Section 2.1.A.1.

4.1 BASES (Cont'd.)

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; e.e., in the period of a month a maximum drift of 0.4% could occur, thus providing for adequate margin.

For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable.

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

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

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Reactor Low Pressure	300-350 psig	4 Inst. Channels	Permissive for opening Core Spray and LPCI Admission valves. Coincident with high drywell pressure, starts LPCI and Core Spray pumps. Permissive for closing Recirculation Discharge Valve.
1	Reactor Low Pressure	$50 \leq P \leq 75$ psig	2 Inst. Channels	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.
2	Reactor Drywell Press Pressure	≤ 2 psig	4 Inst. Channels	1. In conjunction with Low-Low Reactor Wate Level, 120 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS).

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)	$\leq [0.66W + 42]\left(\frac{A}{MTPF}\right)$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Startup Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 2.5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq [0.66W + 41]\left(\frac{A}{MTPF}\right)$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (3)	$\geq$ 2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$ 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

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 ≐ 2.63 for 7x7 fuel and 2.44 for 8x8 fuel. MTPF is the value of the existing maximum total peaking factor.

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)

3.3.B (cont'd.)

B. Control Rods

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

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

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

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

4.3.B (cont'd.)

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

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION

3.3.B (cont'd.)

c. (deleted)

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

SURVEILLANCE REQUIREMENT

4.3.B (cont'd.)

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

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

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

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

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

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

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

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

3.3 and 4.3 BASES (cont'd.)

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

3. The Rod Worth Minimizer (RWM) and sequence mode 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-21172, Revision 1.

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

   Up to 50% rod density (either sequence A or B control rods fully withdrawn and the other sequence fully inserted), the sequence mode of the RSCS restricts the maximum positive reactivity which can be added to the core due to a dropped control rod by control rod selection. Between 50% rod density and 30% of rated power, the group notch mode of the RSCS restricts the reactivity worth by requiring movement of control rods such that rods assigned to each notch group are kept within one notch of each other.

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

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

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

3.4 BASES

STANDBY LIQUID CONTROL SYSTEM

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

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 600 ppm of boron in the reactor core in less than 125 minutes. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a 2.7% Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the zenon poison peak.

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

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

3.4 BASES (Cont'd.)

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

The solution is kept at least 10°F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

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

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

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

Objective:

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

Specification:

A. Core Spray and LPCI Subsystems

1. Two independent Core Spray Subsystems (CSS) shall be operable with each subsystem comprised of:

   a. (Two 50%) capacity centrifugal pumps.

   b. An operable flow path capable of taking suction from the suppression pool and transferring the water to the spray sparger in the reactor vessel.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

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

Objective:

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

Specification:

A. Core Spray and LPCI Subsystems

1. Core Spray Subsystem Testing.

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.A Core Spray and LPCI Subsystems (cont'd.)	4.5.A Core Spray and LPCI Subsystems (cont'd.)
Both CSS shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Shutdown condition except as specified in 3.5.A.2 and 3.5.F.3 below.	Item — Frequency (d) Pump flow rate Both loops shall deliver at least 6250 gpm against a system head corresponding to a reactor vessel pressure of 105 psig. — Once/3 months (e) Core Spray Header ΔP Instrumentation Check — Once/day Calibrate — Once/3 months (f) Operability check to ensure that pumps will start and that injection valves will open. — In accordance with 4.5.A.2, 4.5.A.4 and 4.5.A.5
2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystems are operable.	2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI subsystems shall be demonstrated to be operable in accordance with 4.5.A.1(f) and 4.5.A.3(e) within 24 hours, and at least once per 72 hours thereafter until the inoperable core spray subsystem is restored to operable status. 3. LPCI Subsystem Testing shall be as follows: (a) Simulated Automatic Actuation Test — Once/Operating Cycle (b) Pump Operability — Once/1 month

LIMITING CONDITIONS FOR OPERATION

3.5.A <u>Core Spray and LPCI Subsystems</u> (Cont'd)

3. Two independent Low Pressure Coolant Injection (LPCI) subsystems will be operable with each subsystem comprised of:

   a. (Two 33-1/3%) capacity pumps,

   b. An operable flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel, and

   c. During power operation the LPCI system cross-tie valve closed and the associated valve motor operator circuit breaker locked in the off position.

Both LPCI subsystems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Shutdown Condition, except as specified in 3.5.A.4 and 3.5.A.5 below.

4. From and after the date that one of the four LPCI pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days provided that during such seven days the remaining active components of the LPCI subsystems, and all active components of both core spray subsystems are operable.

5. From and after the date that one LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems and the remaining LPCI subsystem are operable.

SURVEILLANCE REQUIREMENT

4.5.A <u>Core Spray and LPCI Subsystems</u> (Cont'd)

	<u>Item</u>	<u>Frequency</u>
(c)	Motor Operated valve operability	Once/month
(d)	Pump Flow Rate	Once/3 months

Each LPCI pump shall deliver 10,000 gpm against a system head corresponding to a vessel pressure of 20 psig based on individual pump tests.

(e)	Operability check to ensure that pumps will start and injection valves will open	In accordance with 4.5.A.2, 4.5.A.4 and 4.5.A.5

4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operable, the remaining LPCI pumps and associated flow paths and both core spray subsystems shall be demonstrated to be operable in accordance with 4.5.1(f) and 4.5.A.3(e) within 24 hours and at least once per 72 hours thereafter until the LPCI subsystem is restored to operable status.

5. When it is determined that one of the LPCI subsystems is inoperable, both core spray subsystems and the remaining LPCI subsystem shall be demonstrated to be operable within 24 hours, and at least once per 72 hours thereafter until the LPCI subsystem is restored to operable status.

LIMITING CONDITIONS FOR OPERATION

3.5.A Core Spray and LPCI Subsystems (cont'd.)

6. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 36 hours.

B. Containment Cooling Subsystem (HPSW)

1. Except as specified in 3.5.B.2, 3.5.B.3, 3.5.B.4 and 3.5.F.3 below, all containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Condition.

2. From and after the date that any two HPSW pumps are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days, unless such pump is sooner made operable, provided that during such thirty days all other active components of the containment cooling subsystem are operable.

SURVEILLANCE REQUIREMENT

4.5.A Core Spray and LPCI Subsystems (cont'd.)

B. Containment Cooling Subsystem (HPSW)

1. Containment Cooling Subsystem Testing shall be as follows:

	Item	Frequency
(a)	Pump & Valve Operability	Once/3 months
(b)	Pump Capacity Test. Each HPSW pump shall deliver 4500 gpm at 280 psig.	After pump maintenance and every 3 months
(c)	Air test on drywell and torus headers and nozzles.	Once/5 years

2. When it is determined that any two HPSW pumps are inoperable, the remaining components of the containment cooling subsystems shall be demonstrated to be operable immediately and weekly thereafter.

3.5.I Average Planar LHGR

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

3.5.J Local LHGR

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

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

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

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

LT = Total core length = 12 ft

L = Axial position above bottom of core

4.5.I. Average Planar LHGR

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

4.5.J. Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power. This daily requirement is relaxed provided there has been no significant change in power level or distribution as determined by the reactor engineer.

LIMITING CONDITION OF OPERATION

3.5.J. Local LHGR (Cont'd.)

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

3.5.K. Minimum Critical Power Ratio (MCPR)

During steady state power operation, MCPR shall be $\geq$ 1.28 for 7x7 fuel and $\geq$ 1.31 for 8x8 fuel at rated power and flow. For core flows other than rated the MCPR shall be $\geq$ 1.28 times k_f for 7x7 fuel and $\geq$ 1.31 times k_f for 8x8 fuel where k_f is as shown in Figure 3.5.1-E. If at any time during steady state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

SURVEILLANCE REQUIREMENT

4.5.K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power. This daily requirement is relaxed provided there has been no significant change in power level or distribution as determined by the reactor engineer.

3.5.A BASES

Core Spray and LPCI Subsystems

Core Spray Subsystem (CSS)

The CSS is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two redundant loops each provide adequate core cooling capacity for all break sizes from 0.2 ft^2 up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the Automatic Depressurization System (ADS).

The CSS specifications are applicable whenever irradiated fuel is in the core because the CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and also provides a source for flooding of the core in case of accidental draining.

With one CSS inoperable, the demonstrated operability of the redundant full capacity CSS and the full capacity Low Pressure Coolant Injection system provides assurance of adequate core cooling and justifies the specified 7 days out-of-service period.

The surveillance requirements provide adequate assurance that the CSS will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

Low Pressure Coolant Injection System (LPCIS)

The LPCIS is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two loops each with two pumps provide adequate core flooding for all break sizes from 0.2 ft^2 up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The LPCIS specifications are applicable whenever there is irradiated fuel in the reactor vessel because LPCIS is a primary source of water for flooding the core after the reactor vessel is depressurized.

With one LPCIS pump inoperable, or one LPCIS loop inoperable, adequate core flooding is assured by the demonstrated operability of the redundant LPCIS pumps or loop, and both CSS loops. The reduced redundancy justifies the specified 7 day out-of-service period.

The surveillance requirements provide adequate assurance that the LPCI will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

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3.5.E BASES (Cont'd.)

With one ADS valve known to be incapable of automatic operation, four valves remain operable to perform their ADS function. However, since the ECCS Loss-of-Coolant Accident analysis for small line breaks assumed that all five ADS valves were operable, reactor operation with one ADS valve inoperable is only allowed to continue for seven (7) days provided that the HPCI system is demonstrated to be operable and that the actuation logic for the (remaining) four ADS valves is demonstrated to be operable. The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

F. Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Since the system cannot be pressurized during refueling, the potential need for core flooding only exists and the specified combination of the core spray or the LPCI system can provide this. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI subsystem, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. If a water hammer were to occur at the time at which the system were required, the system would still perform its design function. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

3.5 BASES (Cont'd)

H. Engineered Safeguards Compartments Cooling and Ventilation

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

I. Average Planar Linear Heat Generation Rate (APLHGR)

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

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

The calculational procedure used to establish the APLHGR shown on Figures 3.5.1.A, B, F and G is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 4. Differences in the Peach Bottom analysis as compared to previous analyses performed with Reference 4 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.5.1-A, B, F and G; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 5, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 3.5-1.

J. Local LHGR

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

Densification analyses for 8x8 fuel are presented in Section 3.3.4.3 and Appendix B of NEDO-20360, Supplement 3.

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.5.K are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients presented in Reference 6. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

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

The limiting transients which determine the required steady state MCPR limits are: for 7x7 fuel - rod withdrawal error transient, and for 8x8 fuel - turbine trip with failure of the bypass valves. These transients yield the largest ΔCPR for each class of fuel. When added to the safety limit MCPR of 1.06, the required minimum operating limit MCPR's of specification 3.5.K are obtained.

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

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

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

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

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

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

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

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

The analysis of the rod withdrawal error for Peach Bottom Unit 2 considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor which is operating with the limiting control rod pattern as discussed above. This rod pattern is shown in Figure 6-9 and 6-10 of Reference 6.

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

Analysis of turbine trip with failure of the bypass valves is presented in Section 6.3 of Reference 6. Input data and operating conditions used in this analysis are shown in Table 6-1 of Reference 6.

L. 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, 1974.

6. "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal with Partial Installation of the Alternate Flow Path for Peach Bottom Atomic Power Station Unit 2, License No. DPR-44, Docket No. 50-277", NEDO-21172, Revision 1, March 1976.

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

TABLE 3.5-1

PEACH BOTTOM 2 (Alternate Flow Path)

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power	3440 MWt which corresponds to 105% of rated steam flow
Vessel Steam Output	14.049×10^6 lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Design Basis Recirculation Line Break Area	4.28* and 1.0
Recirculation Line Break Area for Small Breaks	1.0 and 0.07

FUEL PARAMETERS:

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (kW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio
Initial Core	7 x 7	18.5	1.5	1.17
Reload No. 1	8 x 8	13.4	1.4	1.17

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

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

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

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

4.5.L MCPR Limits for Core Flows Other than Rated

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

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

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

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow rate (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was

PEACH BOTTOM UNIT 2

RELOAD 1

8x8 Fuel

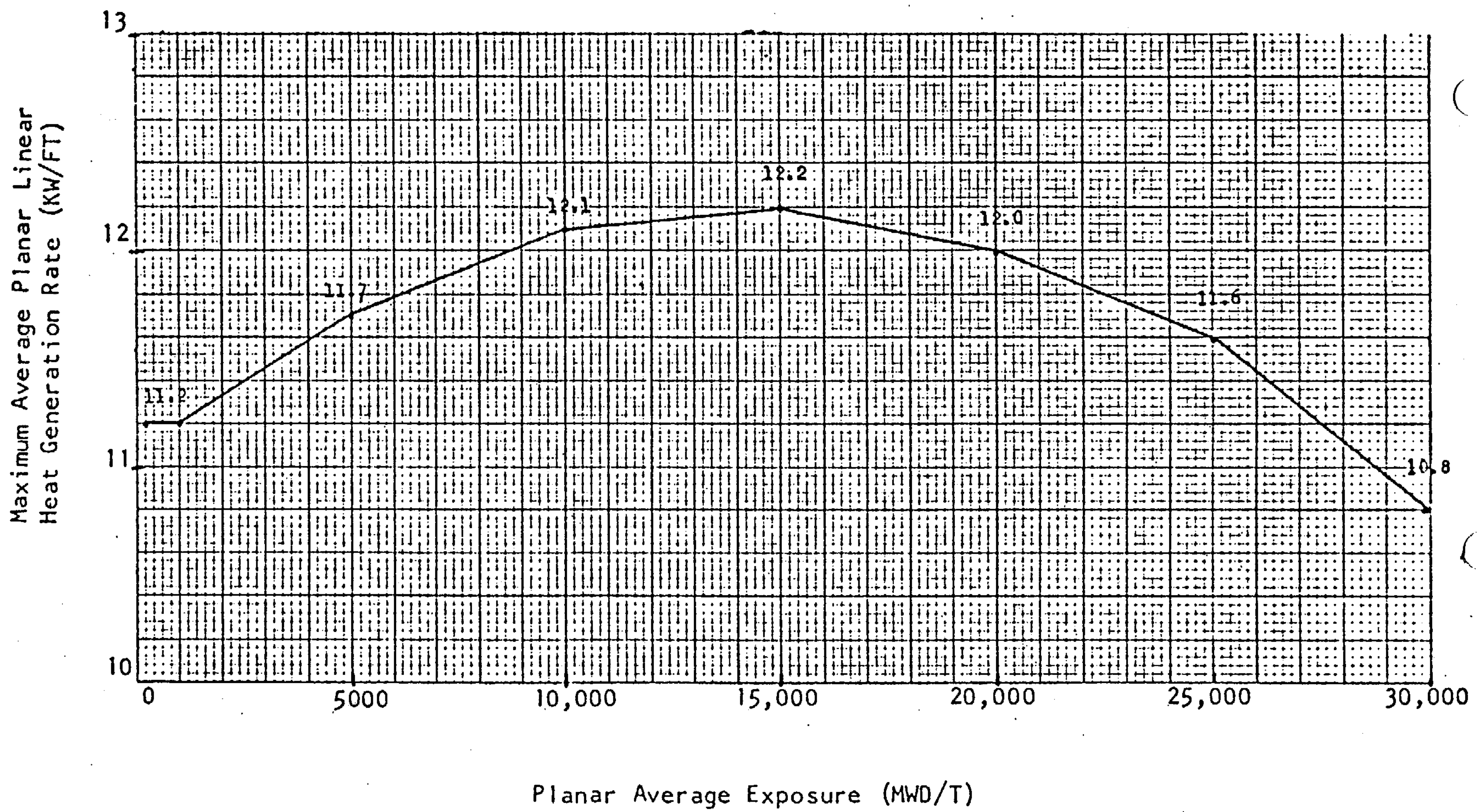

Figure 3.5.1.F Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

RELOAD 1

8x8 LTA Fuel

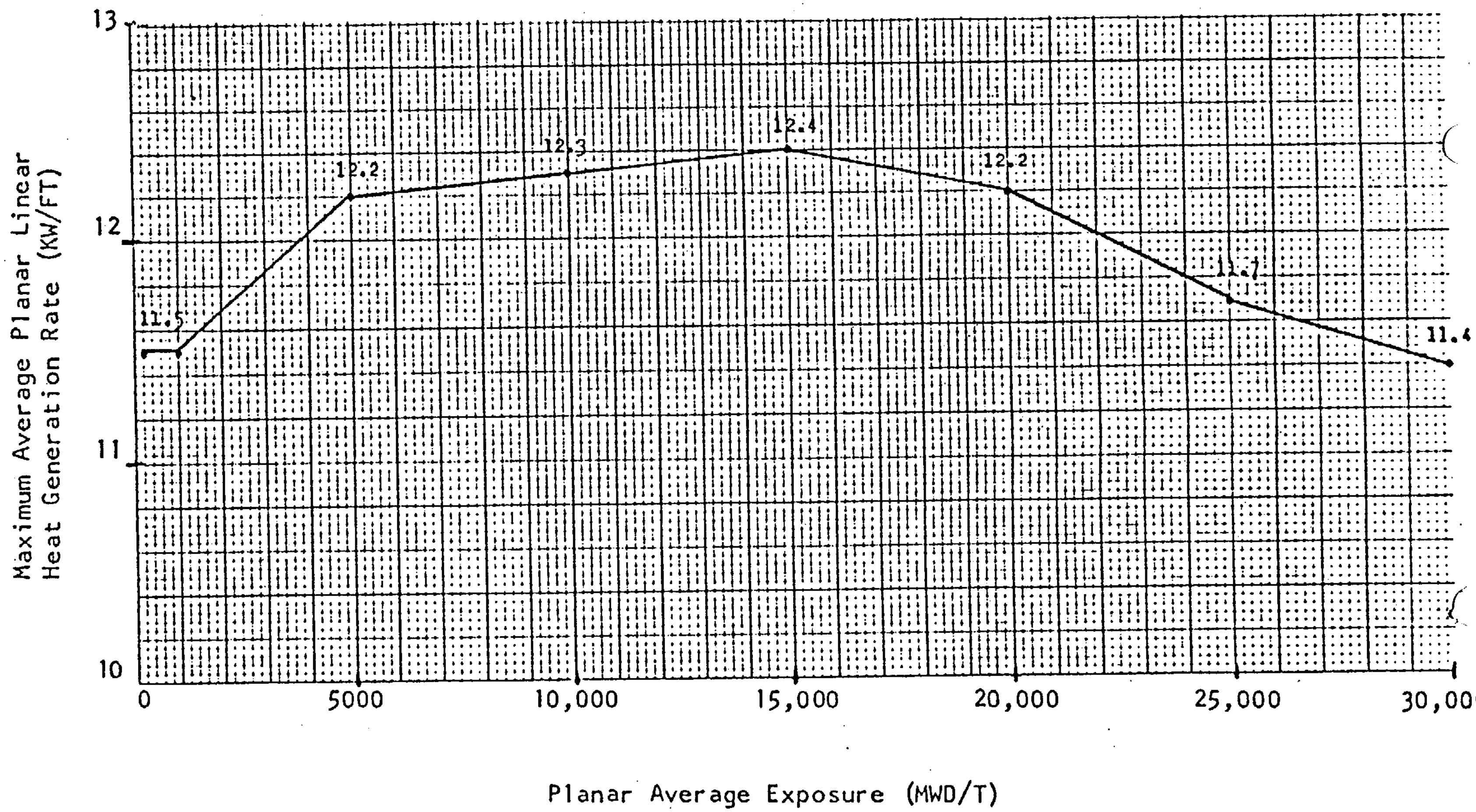

Figure 3.5.1.G Maximum Average Planar Linear Heat Generation Versus
Planar Average Exposure

3.6.D & 4.6.D BASES

Safety and Relief Valves

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

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

Thirteen safety/relief valves have been installed on Peach Bottom Unit 2 with a total capacity of 78.2% 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 1286 psig if a neutron flux scram is assumed. This results in a 89 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the power generation design basis, the total safety/relief capacity of 78.2% has been divided into 64.5% relief (11 valves) and 13.7% safety (2 valves). The analysis of the plant isolation transient (Turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in Section 6.3 of NEDO-21172, Revision 1 for Peach Bottom Unit 2. This analysis shows that the 11 relief valves limit pressure at the safety valves to 39 psig below the setting of the safety valves. Therefore, the safety valves will not open.

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

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

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

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

3.6.F & 4.6.F BASES

Jet Pump Flow Mismatch

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is below 50% of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, continuous reactor operation under such conditions should not be permitted until the necessary analyses have been performed, evaluated and determined acceptable. The reactor may, however, operate for periods up to 24 hours with one recirculation loop out-of-service. This short time period permits corrective action to be taken and minimizes unnecessary shutdowns which is consistent with other Technical Specifications. During this period of time the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients.

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. 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.

5.3 REACTOR VESSEL

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

5.4 CONTAINMENT

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

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

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

5.5 FUEL STORAGE

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

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

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

5.6 SEISMIC DESIGN

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO FACILITY LICENSE NO. DPR-44

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. 2

DOCKET NO. 50-277

1.0 Introduction

By letter dated March 22, 1976 and supplemented by letters dated May 7 and 13, 1976, Philadelphia Electric Company requested an amendment to Facility Operating License No. DPR-44. The amendment would modify the Technical Specifications for the Peach Bottom Atomic Power Station Unit No. 2 to permit operation of the facility with (1) up to 188 General Electric (GE) 8 x 8 reload fuel bundles, (2) four Lead Test assemblies (LTA), (3) twelve developmental fuel channels, (4) holes drilled in the lower tie plate of all reload fuel bundles to provide an alternate bypass flow path, and (5) modification to the Rod Sequence Control System (RSCS).

2.0 Background

The licensee has proposed to reload the Peach Bottom Unit No. 2 reactor with (1) up to 188 GE 8 x 8 bundles with 100 mil channel wall thickness and an average enrichment of 2.74 wt% U-235, (2) four LTA's, and (3) twelve developmental channels. All 1-inch bypass flow holes in the core support plate are plugged. The lower tie plates in the reload assemblies will be drilled with two 9/32 inch holes to provide an alternate bypass flow path.

The documentation submitted in support of the proposed reload relate to the GE BWR Reload 1 licensing submittal for Peach Bottom for the 8 x 8 fuel (References 1 and 2), the LTA's (References 3 and 4), and the developmental channels (References 5 and 6). References 1, 3 and 5 were originally submitted when the licensee proposed to drill all assemblies; however, the licensee revised his proposal such that only reload assemblies will be drilled and submitted References 2, 4 and 6. By letters dated March 22, 1976, and May 7, 1976, the licensee also submitted proposed Technical Specification changes (Reference 7).

The licensee also proposed a modification to the RSCS to replace the group "C" mode with the group notch mode that would be completed during the current refueling outage.

The NRC staff is also imposing more restrictive requirements on the operability of the Low Pressure Coolant Injection (LPCI) System. The additional core reflood capability provided by the LPCI modification and for which credit was given in the current ECCS analysis can be assured only if the LPCI System is available. The changes being implemented on LPCI system operability will ensure that the LPCI system will be available to perform its intended function. We have discussed and agreed upon these changes with the licensee's representatives.

3.0 Evaluation

3.1 Nuclear Characteristics

The information presented in the licensing submittal closely follows the guidelines of Appendix A of NEDO-20360 (Reference 8). Although later supplements to this report are undergoing review by the staff, this topical has been found acceptable for use for reactors containing 8 x 8 reload fuel. Up to 188 GE 8 x 8 reload fuel bundles with an average enrichment of 2.74% by weight will be loaded throughout the core. Sixty-eight of the reload fuel bundles have high gadolinia content (8D274H) and 120 have a low gadolinia content (8D274L). In addition, four LTA's will be loaded and are expected to be operated for four full reactor cycles. The LTA's have a total fueled length of 150 inches compared to 144 inches for a conventional 8 x 8 bundle. The top six inches and the bottom 4 inches in each LTA contain natural uranium pellets. The remaining 140 inches contain pellets of 2.73 percent enrichment. The core contains a total of 764 fuel bundles. Thus, about 25 percent of the fuel bundles are being replaced for the reload.

The loading pattern consists of the 8 x 8 reload bundles scattered throughout the core and the four LTA's symmetrically placed about the core centerline. The high gadolinia reload bundles are loaded in the interior of the core while the low gadolinia reload bundles are loaded at the outer portions of the core. The four LTA's have a low gadolinia content and are therefore loaded at the outer portions of the core. The data in Reference 2 indicate that the nuclear characteristics of the Reload 1 8 x 8 core (including the LTA's) are similar to the previous core. Thus, the total control system worth, temperature, and void dependent behavior of the reconstituted core will not differ significantly from those values previously reported for the Peach Bottom Unit No. 2 reactor. The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.38% Δk subcritical in the most reactive operating state with the most reactive rod fully withdrawn and with all the others fully inserted. For Cycle 2 the minimum shutdown margin is 0.011 Δk, which occurs at the beginning of the cycle.

The information presented in Reference 2 indicates that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by 0.027 Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for Peach Bottom is that the effective multiplication factor (k_{eff}) of the fuel as stored in the fuel storage rack is equal to, or less than, 0.90 for normal conditions. This is achieved if the uncontrolled k_{eff} of a single fuel bundle is less than 1.30 at 65°C (Reference 8). The 8 x 8 8D274H and 8D274L fuel bundles, at both the zero exposure and the peak reactivity point, have a maximum k_{eff} of 1.216 and 1.238, respectively and, therefore, meet the fuel storage requirement for Peach Bottom.

The full power scram reactivity curves used for the Reload 1 cycle are shown in Figure 6.6 of Reference 2. The scram curves used in the anticipated transient analyses include a design conservatism factor of 0.8.

The void and Doppler coefficients of reactivity for the Reload 1 cycle are given in Table 5-1 of Reference 2. The void coefficient of reactivity at the core average void fraction of 41 percent varies from -13.75 to -12.3×10^{-4} $(\Delta k/k)/\Delta V$. The Doppler coefficient at a fuel temperature of 650°C varies from -1.214 to -1.116×10^{-5} $(\Delta k/k)/\Delta T$.

The use of natural uranium in the extremities of the LTA is intended to provide a more efficient use of the total amount of uranium in the core. The nuclear characteristics of the LTA's are not significantly different from the standard 8 x 8 8D274L fuel bundles (Reference 4).

Thus, based on our review of the information presented in the Peach Bottom Unit No. 2 licensing submittal, and the generic 8 x 8 reload report (Reference 8), we conclude that the nuclear characterisitcs and performance of the reconstituted core for Cycle 2 will not differ significantly from that of the first fuel cycle.

3.2 Mechanical Design

The two types of Reload 1 fuel have the same mechanical configuration and fuel bundle enrichments as the 8D264L and the 8D274H fuel assemblies described in the 8 x 8 generic reload report (Reference 8), except for channel wall thickness and the holes drilled in the lower tie plate. The channel wall thickness for reload fuel is nominal 0.100 inches (standard product line fuel channels have nominal 0.080 inch wall thickness). Two 9/32 inch holes are drilled in the lower tie plate of the reload assemblies to provide bypass flow.

The four LTA's are similar in outline dimensions to the reload fuel. The LTA's however, contain two water rods and each fuel rod contains 10 inches of natural uranium.

The developmental channels are identical to the 100 mil channels in the reload fuel, except for variations in the heat treatment process history and channel thickness (120 mil only). The various heat treatments of the developmental channels will investigate improved resistance to channel corrosion in the BWR environment.

As supplemented with information required by our status report (Reference 9) on the GE generic report evaluation, the generic 8 x 8 reload report (Reference 8) which is under review, has been found acceptable for use for reactors containing 8 x 8 reload fuel. The thicker 0.100 (and 0.120) inch wall channels will result in larger margins for withstanding operating loads. On the basis of our review of the generic 8 x 8 reload report and the reload submittal we conclude that the Peach Bottom Unit No. 2 Reload 1 is acceptable.

3.3 Thermal-Hydraulics

The GE generic 8 x 8 fuel reload topical report (Reference 8) and GETAB (Reference 10) are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of the GETAB establishes:

(1) the fuel damage safety limit,

(2) the limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and

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

We have evaluated the Peach Bottom Unit 2 Cycle 2 thermal margins based on the GETAB report and plant specific input information provided by the licensee. The staff evaluation of these margins is reported herein.

3.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition during abnormal operational transients. The uncertainties in the core and system operating parameters and the GEXL correlation (Table 4-1 of Reference 2) combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for Peach Bottom Cycle 2 are the same or more conservative than those used in GETAB (revision of Table IV-1 of NEDO-10958 (Reference 11)). The analysis includes an additional uncertainty in the standard deviation for the TIP readings (3.58-4.08%) due to the bypass void content resulting from plugging of the bypass holes. The generic core selected for the GETAB statistical analysis is a typical 251/764 core and therefore applies to Peach Bottom Unit No. 2. The generic GETAB statistical analysis results are conservative since the bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during the second cycle of operation of the Peach Bottom Unit No. 2 reactor. This results in a conservative value of the MCPR which meets the 99.9% criterion. We conclude that the proposed fuel integrity safety limit MCPR of 1.06 is acceptable for Peach Bottom Cycle 2.

3.3.2 Operating Limit MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit MCPR of 1.06 is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔMCPR). The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR (References 2 and 4). The types of transients evaluated were overpressure, feedwater temperature decrease, coolant flow increase, etc. The most limiting transient in these categories for the 8 x 8 and LTA fuel was the turbine trip without bypass, without two (recirculation) pump trip (assuming end of cycle scram reactivity insertion rates and initial operating conditions of 104.5% of rated power and 100% rated flow), resulting in a ΔMCPR of 0.25. For the 7 x 7 fuel, the most limiting transient is the rod withdrawal error transient, resulting in a ΔMCPR of 0.22. Addition of these ΔMCPR's to the safety limit MCPR of 1.06 gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the operating limit MCPR's are 1.28 for 7 x 7 fuel and 1.31 for 8 x 8 and LTA fuel.

The transient analyses were evaluated with scram reactivity insertion rates that include a design conservatism factor of 0.80. The initial conditions used for the worst operational transient are acceptable. The initial MCPR's assumed in the transient analyses were equal to or greater than the established operating limit MCPR's.

A GE study (Reference 10) has shown that the required operating MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peak was assumed to be representative of beginning-of-cycle conditions and to be located at the midplane (axial peak-to-peak average of 1.40).

The R-factors, which are a function of the local power peaking, assumed in the analyses are also representative of beginning-of-cycle conditions. The values used are 1.098 for 7 x 7 fuel, 1.100 for 8 x 8 fuel and 1.045 for LTA's. During the cycle the local peaking, and therefore the R-factor, is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

Conservatism was applied in the determination of the required operating limit MCPR because the assumed axial and local peaking factors were representative of the beginning of the fuel cycle. This is the worst consistent set of axial and local peaking factors.

Analyses have shown that the operating limit MCPR's of 1.28 for 7 x 7 fuel and 1.31 for 8 x 8 and LTA fuel assure that the fuel cladding integrity safety limit is not exceeded during anticipated abnormal operational transients. Hence, we conclude that the operating limit MCPR's of 1.28 for 7 x 7 fuel and 1.31 for 8 x 8 and LTA fuel are acceptable.

3.3.3 Rod Withdrawal Error

The rod withdrawal error transient is discussed in References 2 and 4 in terms of worst case conditions. Assumptions and descriptions of the rod withdrawal event are given in Reference 8. The information in these two references indicates that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) set at 107% of initial power level will terminate the transient while the critical power ratio is equal to 1.06 for the 7 x 7 fuel, 1.13 for the 8 x 8 fuel and greater than 1.43 for the LTA's. Therefore, the rod withdrawal error transient is limiting for the 7 x 7 fuel for Peach Bottom Unit No. 2, Cycle 2. We conclude that the analysis performed for this localized transient and the predicted consequences of this localized transient are acceptable.

3.3.4 Operating MCPR Limits for Less Than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to the limiting conditions for operation stated in paragraph 3.5.k of the Technical Specifications. This requires that for core flows less than the rated flow, the licensee maintain the MCPR greater than the operating minimum values. The minimum MCPR values for less than rated flow are the rated flow values (1.28 for 7 x 7 and 1.31 for 8 x 8 and LTA) multiplied by the respective K_f factors appearing in Figure 3.5.1-E of the Technical Specifications. The K_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the calculated consequences of the anticipated abnormal transients do not violate the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

3.4 Accident Analysis

3.4.1 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 Section 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 cooling 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 results.

In NEDO-21104 (Reference 12), the licensee submitted a re-evaluation of the Peach Bottom Unit No. 2 ECCS performance. The plant analyzed had all the bypass holes plugged and the LPCI modification. The results of the analysis are applicable to the 7 x 7 fuel.

The loss-of-coolant accident was reanalyzed for the Reload 1 8 x 8 fuel with 100 mil channels and for the LTA's; the results are presented in References 2 and 4. The analyses showed compliance with the criteria in 10 CFR Section 50.46 and Appendix K of 10 CFR Part 50. The Technical Specification changes (Reference 7) which were submitted include new MAPLHGR curves for the 8 x 8 fuel (Figure 3.5.1.F) and the LTA fuel (Figure 3.5.1.G) applicable to fuel Cycle 2.

Although the licensee performed a complete LOCA analysis for the LTA in accordance with Appendix K to 10 CFR Part 50 and submitted his results in Reference 4, the NRC staff performed only a limited review of that analysis because (1) there are only four assemblies in the Peach Bottom Unit No. 2 core, and (2) the predicted peak clad temperature is more than 400°F below the limit of 2200°F. A more detailed review of the analysis based upon Appendix K, 10 CFR Part 50 and the associated model adjustments will be performed, if the LTA's are more numerous in a future application for a core reload.

We conclude that operation of the reactor in accordance with the Technical Specifications will meet the requirements of 10 CFR Section 50.46.

3.4.2 Steamline Break Accident

The steamline break accident analysis (for breakes outside containment) as presented by the licensee is acceptable based on our generic review of NEDO-20360 (Reference 8). Steamline breaks inside containment have been discussed in Section 3.4.1.

3.4.3 Fuel Loading Error

Fuel loading errors are discussed in References 2 and 4 for an 8 x 8 and an LTA fuel bundle placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in References 2 and 4 indicates that a fuel loading error results in a peak linear heat generation rate (LHGR) of 16.5 kW/ft for 8 x 8 and LTA and a minimum critical power ratio (MCPR) greater than 1.10 for 8 x 8 and 1.02 for LTA in the misplaced fuel bundle during steady state operation.

The peak LHGR is less than that required to exceed the specified fuel design limits. Fuel damage criteria are not exceeded during this accident. Fuel bundles adjacent to a misplaced fuel bundle will be negligibly affected. We conclude that the consequences of a fuel loading error are acceptable.

3.4.4 Control Rod Drop Accident

The control rod drop accident for the Peach Bottom Unit No. 2 reloaded core is within the bounding analysis presented in Reference 8. The Doppler coefficient of reactivity, the accident reactivity shape and magnitude function, and the rod drop scram reactivity functions are compared with the technical bases presented in Reference 8. This analysis is performed for Doppler coefficients of reactivity at the beginning of the Reload 1 fuel cycle, at both cold (20°C) and hot (286°C) startup conditions. It is shown by Figures 6-1, 6-2, 6-3, 6-4 and 6-5 of Reference 2 that the maximum values of the parameters for this reloaded core will not exceed the bounding values. The results presented in Reference 2 also apply to the LTA fuel.

Therefore, we conclude that the consequences of a control rod drop accident from any in-sequence control rod during startup will be below the design limit of 280 cal/gm.

3.4.6 Fuel Handling Accident

With respect to fuel handling accidents, in References 2 and 4, the applicant noted that the description and analyses of this event provided in the FSAR and discussed in the generic 8 x 8 reload report (Reference 8) are applicable to this reload. That is, the total activity released to the environment and the radiological exposures for the 8 x 8 fuel will be less than those values presented in the FSAR for the 7 x 7 core. As identified in the FSAR the radiological exposures for this accident with 7 x 7 fuel are well below the guidelines set forth in 10 CFR Part 100. Therefore, it is concluded that the consequences of this accident for the 8 x 8 fuel will also be well below the 10 CFR Part 100 guidelines.

3.5 Overpressure Analysis

In reference 2 the licensee presented the results of an overpressure analysis to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure (these results are also applicable to the LTA fuel). The transient analyzed was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed for 104.5% power with the end of cycle scram reactivity insertion rate curve, scram initiated by high neutron flux, void reactivity applicable to this reload, no credit for the relief function of the safety/relief valves, with all safety valves operative. The results of this analysis indicate that the peak pressure at the vessel bottom would be 1286 psig. Furthermore, generic analysis (Reference 13) applied to Peach Bottom Unit No. 2 showed that for the aforementioned overpressure event, the failure of one safety valve would cause the maximum vessel pressure to increase by less than 20 psig. Hence the maximum peak pressure at the vessel bottom for MSIV closure with flux scram, no relief function of the safety relief valves and one failed safety valve is calculated to be less than 1306 psig; this results in about a 69 psi margin below the code allowable, which is acceptable to the staff.

3.6 Core Modification (Partial Drilling)

The NRC staff has previously approved the proposed modification to eliminate significant in-core vibration for facilities employing holes drilled in all fuel bundle lower tie plates (Reference 15). Concurrent with the evaluation for a "fully drilled" core, Reference 15 also considered and approved the mechanical and hydraulic affects of operating with only some of the fuel bundle lower tie plates drilled. The licensee's submittal was conservative in addressing the bypass-region-to-bundle flow rates and the core reflood rate for ECCS and is therefore acceptable. The past core modification surveillance is covered in Section 6.0.

3.7 Modifications to the Rod Sequence Control System (RSCS)

We have reviewed the licensee's submittal regarding the proposed modifications to the RSCS (References 4 and 7). The proposed modifications will upgrade the RSCS at Peach Bottom Unit No. 2 to the RSCS/BWR-4 design (Group Notch Control for plants with 25 wire probes) as reviewed on the Browns Ferry Dockets Nos. 50-259, 50-260 and 50-296.

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

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

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

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

4.0 Technical Specification Changes

The proposed Technical Specification changes (Reference 7) based on GETAB for Peach Bottom Unit No. 2 incorporate the Fuel Cladding Integrity Safety Limit MCPR and Operating Limit MCPR's as identified in Reference 2. (The LTA is 8 x 8 fuel, and the Safety Limit MCPR and Operating Limit MCPR for 8 x 8 fuel apply.)

The licensee in Reference 7 proposed to incorporate the effect of densification power spiking for 8 x 8 fuel into the maximum allowable LHGR without using a correction equation. The staff has not yet approved this concept, thus, the licensee will be required to use a correction equation to account for the effect of power spiking caused by fuel densification.

We find the proposed Technical Specification changes acceptable and consistent with the information in the Reload 1 licensing submittal.

5.0 Lead Test Assemblies and Developmental Channels

Based on our review of References 4 and 6, we approve the use of four Lead Test Assemblies and twelve developmental channels only in the Peach Bottom Unit No. 2 Reload 1 cycle. This approval does not allow expanded use of similar assemblies and channels in Peach Bottom Unit No. 2 or in other reactors without further staff review of their specific application.

In order to facilitiate future reviews in which LTA's are used as a major portion of the reload. the licensee is being requested to report the results of his findings concerning the LTA's and developmental channels at the next refueling outage.

6.0 Post Core Modification Surveillance

Since this application is among the first to incorporate partial drilling of the lower tie plates, we require that a surveillance program be implemented to assure that there is no adverse impacting of the LPRM's on the channel boxes. The surveillance program should include the following:

(1) Perform a detailed visual inspection of a statistically significant number of channel boxes for the first two refueling cycles.

(2) Monitor unfiltered TIP traces and report any anomalous behavior to the NRC.

(3) Install accelerometers on a number of in-core instrument source tubes, monitor the accelerometer signal at least monthly, and report any anomalous behavior to the NRC.

Further details of these surveillance requirements are discussed in Reference 15.

The licensee has committed to performing a surveillance program in accordance with requirements stated above. He will submit his proposed program to the NRC for review and approval.

7.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

8.0 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: June 11, 1976

REFERENCES

1. "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21172, February 1976. Appendix A to NEDO-20360, Revision 1, Supplement 3, September 1975.

2. "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal with Partial Installation of the Alternate Flow Path for Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21172, Revision 1, March 1976. Appendix A to NEDO-20360, Revision 1, Supplement 3, September 1975.

3. "Lead Test Assembly Supplemental Information for Reload 1 Licensing Submittal for Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21172, Supplement 1, Class 1, February 1976.

4. "Lead Test Assembly Supplemental Information for Reload 1 Licensing Submittal for Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21172, Revision 1, Supplement 1, Class 1, March 1976.

5. "Developmental Channels - Supplemental Information for Reload 1 Licensing Submittal for Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21172, Supplement 2, Class 1, February 1976.

6. "Developmental Channels - Supplemental Information for Reload 1 Licensing Submittal for Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21172, Revision 1, Supplement 2, Class 1, March 1976.

7. Philadelphia Electric Company Applications for Amendment of Facility Operating License DPR-44, March 1976 and May 1976.

8. "General Electric Generic Reload Licensing Application for 8 x 8 Fuel," Revision 1, Supplement 3, September 1975, NEDO-20360.

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

10. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, 73 NED9, Class 1, November 1973.

11. General Electric Letter (John A. Hinds) to U.S. Atomic Energy Commission (Walter Butler), "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO-10958 and NEDE-10958, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application." July 24, 1974.

12. "Peach Bottom Atomic Power Station Unit 2 Channel Inspection and Safety Analysis with Bypass Holes Plugged." NEDO-21104, 75NED62, Class 1, November 1975.

13. General Electric Letter (Ivan F. Stuart) to U.S. Nuclear Regulatory Commission (Victor Stello) "Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressures to Valve Operability," December 23, 1975.

14. Philadelphia Electric Company Letter (John L. Hankins) to U.S. Nuclear Regulatory Commission (George Lear), "Responses to April 20, 1976 Request for Additional Information Peach Bottom 2 Reload," May 7, 1976.

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-277

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 23 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 amendment will modify the provisions in the Technical Specifications to authorize operation with (1) up to 188 GE 8 x 8 reload fuel assemblies, (2) four Lead Test Assemblies, (3) twelve developmental fuel channels, (4) holes drilled in the lower tieplate of all reload 8 x 8 fuel bundles, and (5) a modified rod sequence control system.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the

FEDERAL REGISTER on April 26, 197C (41 FR 17435). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental 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 applications for amendment dated March 22, 1976 and May 13, 1976, and supplement dated May 7, 1976, (2) Amendment No. 23 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 Martin Memorial Library, 159 E. Market Street, York, Pennsylvania 17401.

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

Dated at Bethesda, Maryland, this 11th day of June, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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