November 28, 1975

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:

The Commission has issued the enclosed Amendment No. 15 to Facility License No. DPR-44 for Peach Bottom Atomic Power Station Unit 2. The amendment includes Change No. 15 to the Technical Specifications and is in response to your requests dated July 9, 1975 and Supplements thereto dated September 10, 1975, October 1, 24 and 30, 1975, and November 7, 18 and 20, 1975.

The amendment authorizes operation of Peach Bottom Unit 2 (1) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), (2) with modified operating limits based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's regulations, and (3) with a modification to the Low Pressure Coolant Injection System (LPCIS) authorized by Amendment No. 14 to the license.

Based on our review of the information submitted by you on the proposed modifications to the LPCIS for Peach Bottom Unit 2, we have determined that Peach Bottom Unit 2 can operate with the LPCIS modification provided that at a future outage, date to be proposed by Philadelphia Electric Company (PECO) and approved by the NRC staff, separate, redundant and independent buses would replace the existing swing buses supplying the valve motor operators for the LPCIS injection valves. Also included in Amendment No. 15 are additional surveillance requirements on the swing buses and associated electrical systems which are to be implemented following the completion of the proposed LPCIS modifications.

The Commission also has issued the enclosed Order for Modification of License which authorizes operation of Peach Bottom Unit 2 with plugged bypass flow holes, subject to the conditions set forth in Change No. 15 issued with Amendment No. 15 to the license, in accordance with your application dated November 7, 1975 as supplemented. This Order superSedes the December 27, 1974 Order for Modification of License and the November 4,

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SATeets	GLear	VStello
11/24/75	11/26/75	11/ /75

KRG

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SURNAME ➤	HVerdery:kmf		KRGoller	RSBoyd	EGCase	BCRusche
DATE ➤	11/24/75	11/25/75	11/26/75	11/26/75	11/ /75	11/28/75

1975 Order for Modification of License in their entirety.

The Commission's staff has evaluated the potential for environmental impact associated with operation of Peach Bottom Unit 2 in accordance with this license amendment. From this evaluation the staff has determined that there will be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to that action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal also are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

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

Sincerely,

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

Enclosures:
1. Amendment No. 15
2. Order for Modification of License
3. Negative Declaration
4. Environmental Impact Appraisal
5. Safety Evaluation
6. Federal Register Notice

cc w/enclosures:
See next page

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DATE➤						

1975 Order for Modification of License in their entirety.

The Commission's staff has evaluated the potential for environmental impact associated with operation of Peach Bottom Unit 2 in the manner set forth in item (2) of the second paragraph above. From this evaluation the staff has determined that there will be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to that action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal also are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

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

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George Lear, Chief
Operating Reactors Branch #3
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1. Amendment No. 15
2. Order for Modification of License
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5. Safety Evaluation
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric & Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company (the licensees) dated July 9, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.(C).2 of Facility License No. DPR-44 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 15."

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 Reactor Licensing

Attachment:
Change No. 15 to the
Technical Specifications

Date of Issuance: November 28, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 15

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace pages i, ii, iii, iv, 5, 6, 9 through 28, 33 through 36, 53, 54, 65, 66, 69, 70, 71, 72, 81, 82, 91, 92, 103, 104, 110a, 111, 112, 133a, 133b, 140, 140a, 141, 142, 142a, 149, 150, 159 and 160. (No change made on pages ii, iii, 6, 12, 23, 36, 53, 65, 70, 72, 82, 112, 141, 150 and 159.)

Add page 218a.

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1.0 DEFINITIONS (Cont'd.)

Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958)

2. Maximum Total Peaking Factor - The Maximum Total Peaking Factor (MTPF) is the lowest Total Peaking Factor which limits a fuel type to a Linear Heat Generation Rate (LHGR) corresponding to the 15
operating limit at 100% power.

3. Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core.

4. Total Peaking Factor - The ratio of the maximum fuel rod surface heat flux in an assembly to the average surface heat flux of the core.

5. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

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

4. Instrment Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during

by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation or protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

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

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

9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

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

    (a) Initiating - A logic that receive signals from channels and produces decision outputs to the actuation logic.

    (b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective

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

Specification:

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

The existence of a minimum
15 critical power ratio MCPR less then 1.06 shall constitute violation of the fuel cladding integrity safety limit.

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded. 15

Objective

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

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

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

$$S \leq 0.66 W + 54\%$$ 15

where:

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

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A (cont'd.)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of 2.63, the setting shall be modified as follows:

$$S \leq (0.66\ W + 64\%)\ \frac{2.63}{MTPF}$$

Where: 5

MTPF = The value of the existing maximum total peaking factor

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

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

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

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

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

LIMITING SAFETY SYSTEM SETTING

B. APRM Rod Block Trip Setting

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

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 15
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 2.63, the setting shall be modified as follows:

$$S_{RB} \leq (0.66\ W + 42\%)\ \frac{2.63}{MTPF}$$

where:

MTPF = The value of the existing maximum toal peak factor.

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.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING	
	F. Scram--low condenser vacuum	$\geq$23 inches Hg vacuum
	G. Scram--main steam line isolation	$\leq$10 percent valve closure
	H. Main steam isolation valve closure --nuclear system low pressure	$\geq$850 psig
	I. Core spray and LPCI actuation--reactor low water level	$\geq$378 in. above vessel zero (-159.5 in. indicated level)
	J. HPCI and RCIC actuation--reactor low water level	$\geq$490 in. above vessel zero (-49.5 in. indicated level)
	K. Main steam isolation valve closure --reactor low water level	$\geq$490 in. above vessel zero (-49.5 in. indicated level)

2.1 BASES: (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973. 15

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2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

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

The details of the analysis which shows compliance with the ASME Code requirements is 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 each unit. 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 1260 psig for Peach Bottom Unit 3 and 1292 for Peach Bottom Unit 2 if a neutron flux scram is assumed. This results in 115 psig and 83 psig margins respectively to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient (turbine trip with bypass valve 15
failure to open) assuming a turbine trip scram is presented in FSAR paragraph 14.5.1.2 and Figure 14.5.16 for Peach Bottom Unit 3 and in Section 7.2 and Figures 7-2 and 7-3 of NEDO-21104 for Peach Bottom 2. These analyses show that the 11 relief valves limit pressure at the safety valves to 62 psig and 49 psig, respectively 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.

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

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

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

B. Daily during reactor power operation, the peak heat flux and peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specification 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds 2.63.

15

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
	D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels. The trip system may be in the untripped position for no more than eight hours per functional trip period for this testing.

4.1 BASES (Cont'd)

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20 X 10-6 failure/hour. The bi-stable trip circuits are predicted to have unsafe failure rate of less than 2×10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above, and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. The flow biasing network is functionally tested at least once per month and in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semi-conductor devices and detectors that drift or lose sensitivity.

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 peak heat flux 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 of the peak heat flux 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.

It is highly improbable that in actual operation with MTPF at 2.63 that MCPR will be as low as 1.21. Usually with peaking factors of this magnitude the peak occurs low in the core in a low quality region where the initial heat flux is very high. The MCPR design power shape (TPF = 2.43) assumes that the peak occurs higher in the core and represents the worst combination of individual peaking factor magnitude and shape, from a MCPR consideration that can be expected to occur in the core. Therefore, with TPF 15
< 2.43 there are not technical specification requirements for calculating MCPR. With TPF greater than 2.43 the daily requirement for calculating MCPR 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 pattern is approached insures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

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 High Water Level	$\leq$+45 in. indicated level	2 Inst. Channels	Trips HPCI and RCIC turbines.
1	Reactor Low Level (inside shroud)	$\geq$+312 in. above vessel zero (2/3 core height)	2 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
2	Containment High Pressure	$1 < p < 2$ psig	4 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
1	Confirmatory Low Level	$\geq$+6 in. indicated level	2 Inst. Channels	ADS Permissive 9
2	High Drywell Pressure	$\leq$2 psig	4 HPCI Inst. Channels 4 RHR & Core Spray Inst. Channels	1. Initiates Core Spray LPCI; HPCI. 2. Initiates starting of Diesel Generators

1.1 BASES: FUEL CLADDING INTEGRITY LIMIT

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.06. MCPR $>$ 1.06 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforation or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the conditions intended by design for planned operation.

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

Onset of transition boiling results in a decrease in heat transfer from 15
the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables; i. e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The Safety Limit (MCPR) of 1.06 has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR $\geq$ 1.21) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit 1.06 is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. Because of the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.06 would not produce boiling transition.

1.1 BASES (Cont'd)

However, if boiling transition were to occur, clad perforation would not be expected. Clad temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Peach Bottom Atomic Power Station Units 2 and 3 fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (MCPR = 1.06), operation is constrained to a maximum LHGR of 18.5 kW/ft. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 2.63. For the case of the MTPF exceeding 2.63, operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram settings as required by specification 2.1.A.1.

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

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the 15
pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analysis 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. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressure below 800 psia is conservative.

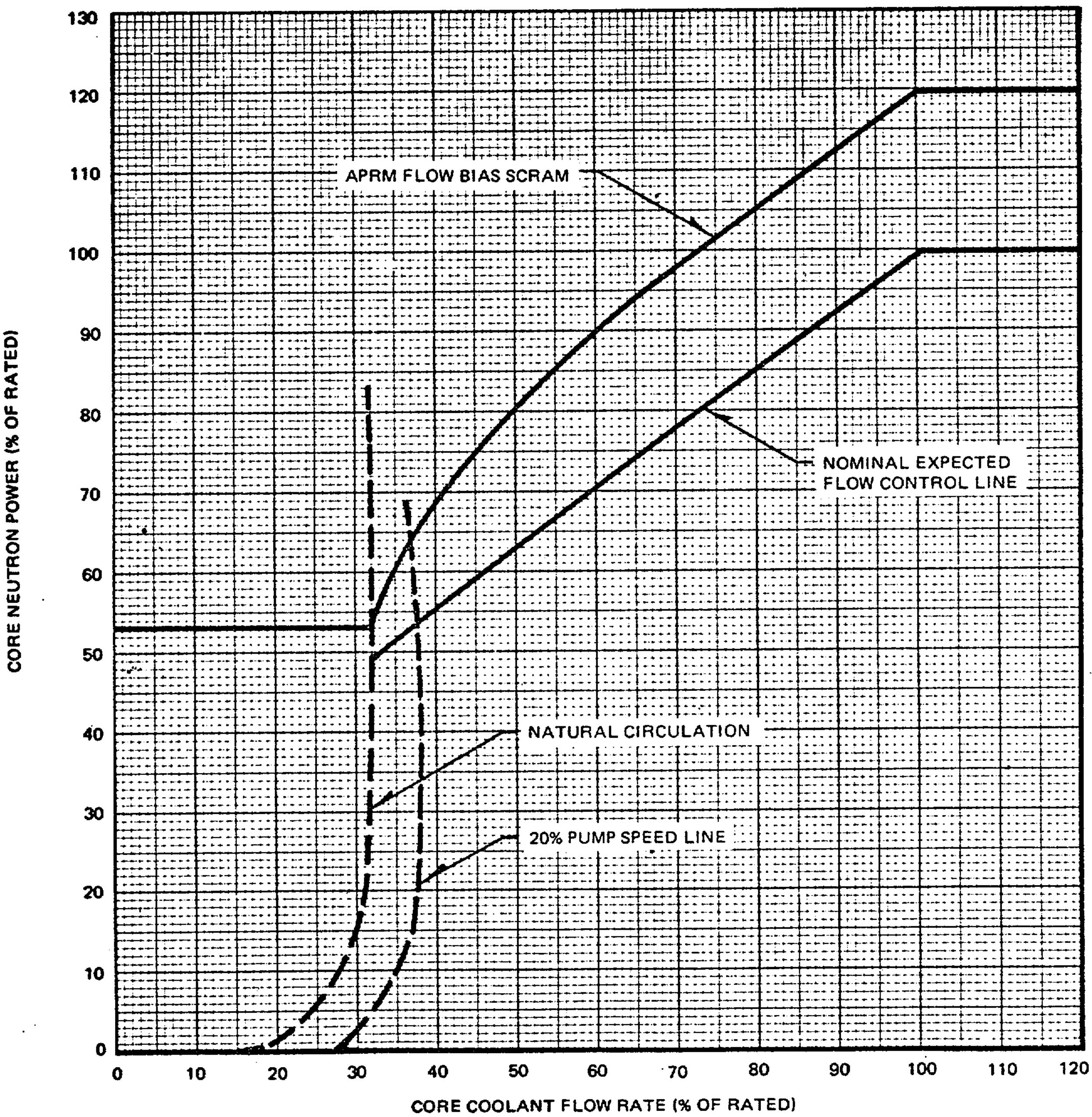
C. Reactor Water Level (Hot or Cold Shutdown Conditions)

For the fuel in the core during periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

The safety limit has been established at 17.7 inches above the top of the irradi[illegible]o provide a point which can be monitored and also provide adequa[illegible]

D. References

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958. 15



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APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

FIGURE 1.1-1

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 15
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 1.21 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. 15

A. Neutron Flux Scram

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (3293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during 15
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 15
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 15
Specification 2.1.A.1, when the maximum total peaking factor is greater than 2.63.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR $>$ 1.06 when the transient is initiated from MCPR $>$ 1.21.

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 15
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.)

already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig. 15

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accomodate the increase in power level, the scram trip setting is also 15
ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to assure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most 15
severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analyses by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.06. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

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. 15
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 15
peaking factor exceeds 2.63, 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 15
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 15
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 [redacted] the hydraulic control system piping. Two turbine first stage pressure switches for each trip system initiate automatic bypass of the turbine control valve fast clo ure scram when the first stage pres80re is below that required to produce 0% of rated power. Control valve closure time is approximately twice as long as that

2.1 BASES: (Cont'd)

system piping. Two turbine first stage pressure switches for each trip system initiate automatic bypass of the turbine control valve fast closure scram when the first stage pressure is below that required to produce 30% of rated power. Control valve closure time is approximately twice as long as that 15
for stop valve closure. No significant change in MCPR occurs.

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2.1 BASES (Cont'd)

F. Main Condenser Low Vacuum Scram

To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vacuum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 850 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I.J.K. Reactor Low Water Level Set Point for Initiation of HPCI and RCIC, Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram set point and initiation set points. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

TABLE 3.2.B

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 dry well pressure, starts LPCI and Core Spray pumps. 15
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. 15
2	Reactor Drywell Pressure	$\leq$ 2 psig	4 Inst. Channels	1. In conjunction with Low-Low Reactor Water Level, 120 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS).

TABLE 3.2.B

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
1	Core Spray Sparger to Reactor Pressure Vessel d/p	5 ($\pm$ 1.5) psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break.
2	Condensate Storage Tank Low Level	$\geq$5' above tank bottom	2 Inst. Channels	Provides interlock to HPCI pump suction valves.
2	Supression Chamber	$\leq$5" above normal water level	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.

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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
1	RCIC Turbine High Flow	$\leq$ 450" H_2O (2)	2 Inst. Channels	
2	RCIC Turbine Compartment Wall	$\leq$ 200 deg. F (2)	4 Inst. } 16 Inst.	
6	RCIC Steam Line Area Temp.	$\leq$ 200 deg. F (2)	12 Inst. }	
2	RCIC Steam Line Low Pressure	100 >P >50 psig (2)	4 Inst.	
1	HPCI Turbine Steam Line High Flow	$\leq$ 225" H_2O (3)	2 Inst. Channels	

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
4 (5)	HPCI Steam Line Low Pressure	$100 > p > 50$ psig (3)	4 Inst.	
2	HPCI Turbine Compartment Temperature	$\leq$ 200 deg. F (3)	4 Inst. } 16 Inst.	
4	HPCI Steam Line Area Temperature	$\leq$ 200 deg. F (3)	8 Inst. } 16 Inst.	
2	HPCI/RHR Valve Station Area Temperature	$\leq$ 200 deg. F (3)	4 Inst. } 16 Inst.	
1 per 4KV Bus	4KV Emergency Bus Undervoltage Relay	25% $\pm$ 5% of Rated Voltage		1. Trips all loaded breakers 2. Fast transfer permissive 3. Dead bus start of diesel
1 per 4KV Bus	4KV Emergency Bus Sequential Loading Relay	95% $^{+0\%}_{-10\%}$ of Rated Voltage		Permits sequential starting of vital loads
2 per 4KV Bus	Emergency Transformer Undervoltage	60% $\pm$ 5% of Rated Voltage		1. Trips emergency transformer feed to 4KV emergency bus 2. Fast Transfer permissive
1	LPCI Cross-Connect Position	NA	1 Inst.	Initiates annunciation when valve is [illegible]sed 5

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS subsystem is required by Section 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument set point corresponds to 18" above the top of active fuel.

5. HPCI has only one trip system for these sensors.

TABLE 4.2.B

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Chec
1) Reactor Water Level	(1)	Once/3 months	Once/day
2) Drywell Pressure	(1)	Once/3 months	None
3) Reactor Pressure	(1)	Once/3 months	None
4) Auto Sequencing Timers	NA	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	None	None
[illegible])			
7) Core Spray Sparger d/p	(1)	once/6 months	Once/day
8) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
9) Steam Line High Temp. (HPCI & RCIC)	(1)	Once/operating cycle	Once/day
10) Safeguards Area High Temp.	(1)	Once/3 months	None
11) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
12) HPCI Suction Source Levels	(1)	Once/3 months	None
13) 4KV Emergency Power System Voltage Relays	Once/operating cycle	Once/5 year	None
14) ADS Relief Valves Bellows Pressure Switches	Once/operating cycle	Once/operating cycle	None
15) LPCI/Cross Connect Valve Position	Once/refueling outage	N/A	N/A

PBAPS

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TABLE 4.2.B

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Logic System Functional Test (4) (6)	Frequency
1) Core Spray Subsystem	Once/6 months
2) Low Pressure Coolant Injection Subsystem	Once/6 months
3) Containment Cooling Subsystem	Once/6 months
4) HPCI Subsystem	Once/6 months
5) HPCI Subsystem Auto Isolation	Once/6 months
6) ADS Subsystem	Once/6 months
7) RCIC Subsystem Auto Isolation	Once/6 months
8) Area Cooling for Safeguard System	Once/6 months

3.2 BASES (Cont'd)

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

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

Temperature is monitored at four (4) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" direct current control bus and two (2) by "B" direct current control bus. Each pair of sensors, e.g., "A" or "B", at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves.

The trip settings of $\leq$ 300% of design flow for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

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

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

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

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.6. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block. 15

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

3.2 BASES (Cont'd)

[illegible]RM rod block function i[illegible]ased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.6. 15

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. 15

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION

3.3.B (cont'd)

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

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

   a. Both RBM channels shall be operable:
   or

   b. Control rod withdrawal shall be blocked:
   or

   c. The operating power level shall be limited so that
15 the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

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

Above 950 psig

[illegible] Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	5.0

SURVEILLANCE REQUIREMENTS

4.3.B (cont'd)

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

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

C. Scram Insertion Times

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

LIMITING CONDITION FOR OPERATION

3.3.C (Cont'd.)

Below 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.475
20	1.10
50	2.0
90	5.0

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

Above 950 psig

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.398
20	0.954
50	2.120
90	5.300

Below 950 psig

% Inserted from Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.504
20	1.166
50	2.12
90	5.300

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

SURVEILLANCE REQUIREMENT

4.3.C (Cont'd.)

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

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

3.3 and 4.3 BASES (Cont'd.)

A limiting control rod pattern is a pattern which results in the
core being on a thermal hydraulic limit (i.e., operating on a 15
limiting value for APLHGR, LHGR, or MCPR as defined in Technical
Specifications 3.5.I., 3.5.J., and 3.5.K.) During use of such
patterns, it is judged that testing of the RBM system prior to
withdrawal of such rods to assure its operability will assure
that improper withdrawal does not occur. It is the responsibility
of the Reactor Engineer to identify these limiting patterns and
the designated rods either when the patterns are initially
established or as they develop due to the occurrence of inoperable
control rods in other than limiting patterns. Other personnel
qualified to perform this function may be designated by the station
superintendent.

3.3 and 4.3 BASES (Cont'd.)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.06. 15
The limiting power transient is that resulting from turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure 3.6.14) with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains 15
greater than 1.06.

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

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C. In addition the control rod drop accident has been analyzed in NEDO 10527 and its supplements 1 & 2 for the scram times given in specification 33.C.

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

3.3 and 4.3 BASES (Cont'd.)

identified as the resistance to drive motion by an internal control rod drive filter. The filter had been loaded by foreign material, probably accelerated by construction debris. The sudden changes in drive scram performance which were observed at that plant were due to stepwise release into reactor coolant of particulate matter as the reactor and subsystems were subsequently started up. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Dresden 2 has currently 27"B" type drives. Approximately 4718 drive tests have been recorded to date. Data documenting the successful performance of the modified drive has been submitted to the AEC with a letter from Commonwealth Edison Company to the Commission dated November 6, 1972 with the subject of the letter being Proposed Changes to Quad-Cities Power Station Operating License, including Appendices A and B, DPR 29 and 30, AEC Dkts 50-254 and 50-265.

Although the cause and cure of the dirt problem were known at the time of the writing of the Dresden 3 Tech Specs, the progressive surveillance requirement was incorporated into that technical specification to ostensibly detect any other unforeseen drive problems. The possibility of this being a temporary requirement may be inferred from the provision for review of all surveillance requirements after the first operating cycle.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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-C or 3.5.1-D as applicable. If at any time during steady state operation it is determined by normal surveillance that the
15 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 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

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = 0.026

LT = Total core length = 12 ft Unit 2
= 12.167 ft Unit 3

L = Axial position above bottom of core

4.5.I. Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power. This daily requirement is relaxed provided 15
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 15
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 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 $\geq$ 1.20 at rated power and flow. For core flows other than rated the MCPR shall be $\geq$ 1.20 times k_f, where k_f is as shown in Figure 3.5.1-E. If at any time during steady
15 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 within the prescribed limits. Surveillance and corresponding 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 15
daily requirement is relaxed provided there has been no significant change in power level or distribution as determined by the reactor engineer.

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 LHGR

This specification assures that the peak cladding temperature following
the postulated design basis loss-of-coolant accident will not exceed
the limit specified in the 10 CFR Part 50, Appendix K. The peak cladding 15
temperature 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. Since expected local
variations in power distribution within a fuel assembly affect the
calculated peak clad temperature by less than $\pm$ 20°F relative to the
peak temperature for a typical fuel design, the limit on the average
linear heat generation rate is sufficient to assure that calculated
temperatures are within the 10 CFR Part 50, Appendix K limit. The limit- 15
ing value for APLHGR is shown in Figures 3.5.1-A, B, C and D.
15

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 15
fuel burnup, or control rod movement has caused changes in power distribu-
tion. 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 15
considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ration (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will 15
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, thermal hydraulic analysis indicated

3.5.K. BASES (Cont'd.)

that the resulting MCPR value is in excess of requirements. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. 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. 15

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.

4.5 BASES

Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with frequent tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, caused the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

4.5 I&J Surveillance Requirements Bases

Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup or control rod movement has cuased changes in power distribution. Since changes due to burnup are slow and only a few control rods are moved daily, a daily check of power distribution is adequate.

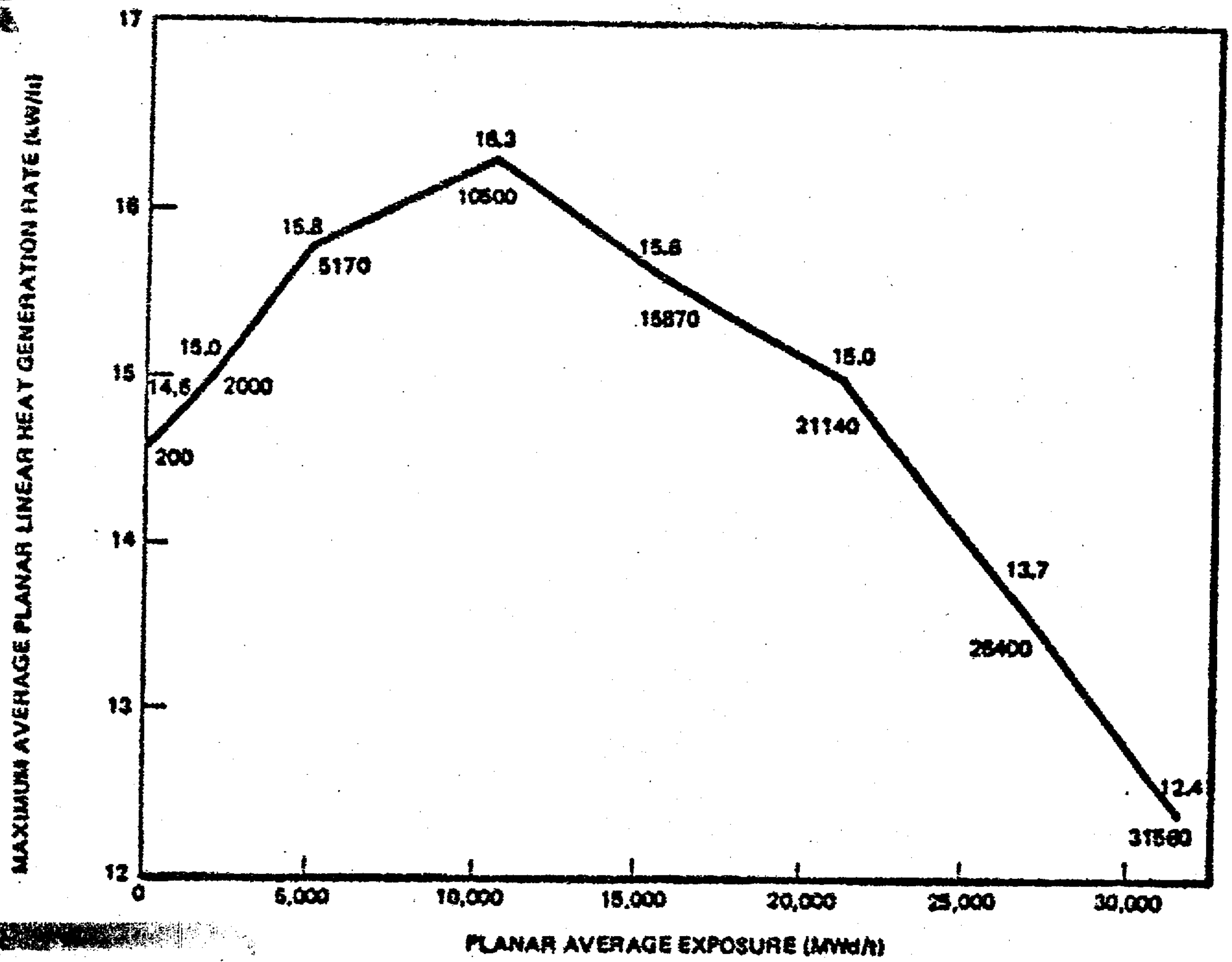
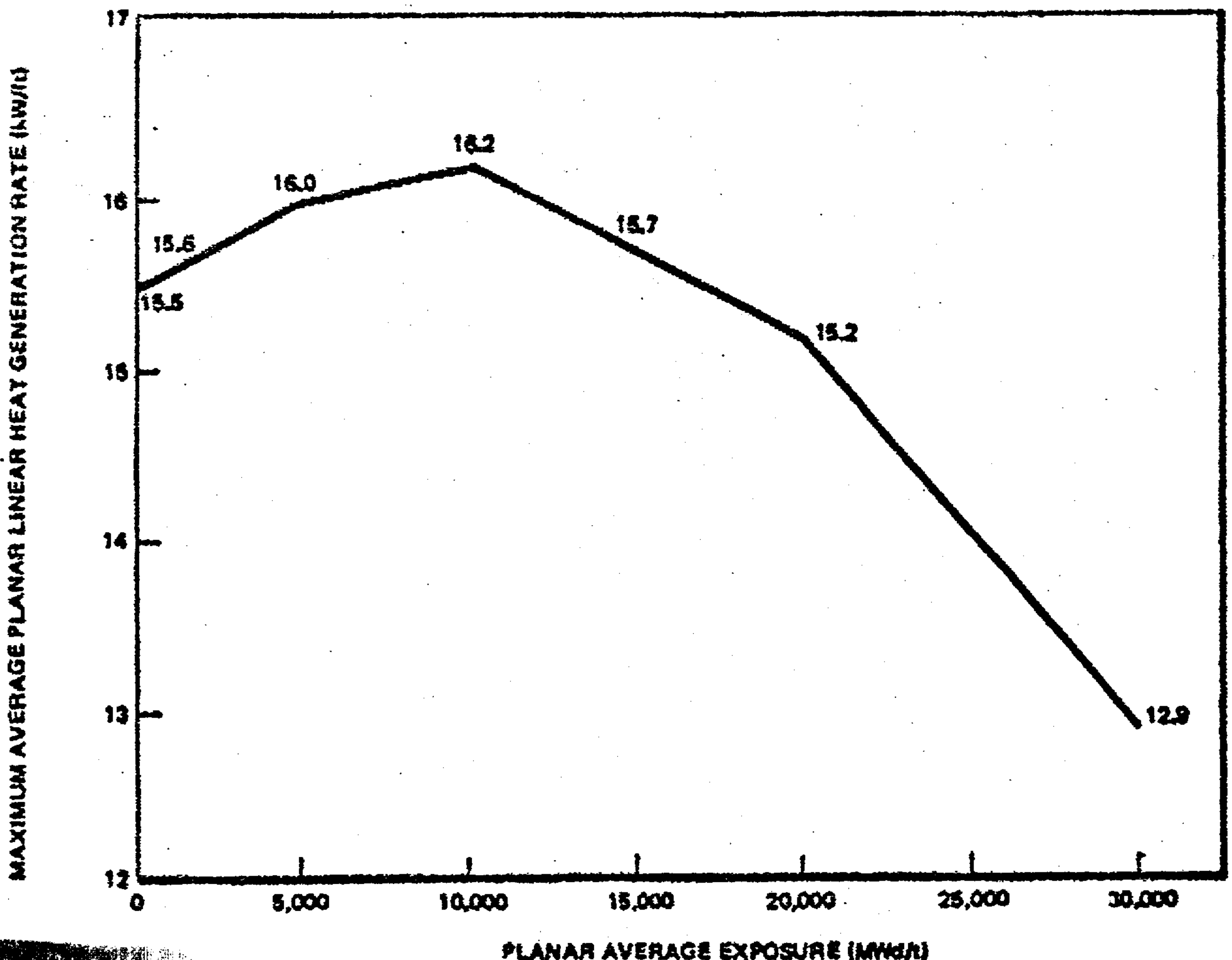


Figure 8-10a. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure



3.5.6.B Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.9.A	4.9.A (Cont'd.)

3.9.A

4.9.A (Cont'd.)

3. Swing Buses

   a. Every two months the swing buses supplying power to the Low Pressure Coolant Injection System (LPCIS) valves shall be tested to assure that the transfer circuits operate as designed. 15

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.F (cont'd) pump may not exceed 122% the speed of the slower pump when core power is 80% or more of rated power or 135% the speed of the slower pump when core power is below 80% of rated power. 2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed. 3. The reactor shall not be operated for a period in 15 excess of 24 hours with one recirculation loop out of service. 4. If Specification 3.6.F.1 cannot be met one recirculation pump shall be tripped.	
3.6.G <u>Structural Integrity</u> The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. The reactor shall be maintained in a Cold Shutdown condition until each indication of a defect has been investigated and evaluated.	4.6.G <u>Structural Integrity</u> The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

Table 4.6.1

IN-SERVICE INSPECTION PROGRAM FOR PEACH BOTTOM UNITS 2 AND 3

Area of Interest (See Notes)

Item No.	Category	Comp. & Parts to be Examined	Method	Exam Schedule in 5-year Period	Exam Schedule in 10-year Interval	Remarks
Reactor Vessel						
1.1	A	Longitudinal and circumferential welds in core region	Volumetric	5% of circumferential 10% of longitudinal welds	5% of circumferential 10% of longitudinal welds	
1.2	B	Code	Volumetric	5% of 1/2 of the gore welds in the closure head; 5% of circumferential welds above sacrificial shield; 10% of longitudinal welds above sacrificial shield	5% of all circumferential welds 10% of all longitudinal welds	
1.3	C	Code	Volumetric	33% of head-to-flange weld; 33% of vessel-to-flange weld including ligaments	100% of head-to-flange weld and 100% of vessel-to-flange weld including ligaments	
1.4	D	Code	Volumetric	One recirculation outlet nozzle; five recirculation inlet nozzles; two steam nozzles; two feed water nozzles; one core spray nozzle; two instrument nozzles	All nozzles in vessel greater than 2 inches in size	Results of volumetric examination of inner nozzle radii from the vessel O.D. are not definitive except for high-strain locations
1.5	E-1	Code	----------	----------------------	----------------------	There are no penetrations in this category due to exclusion criteria of paragraph IS-120 (d)
1.6	E-2	Code	Visual	12% of penetrations	25% of penetrations	The examinations will be performed when the system is pressurized after refueling and prior to operation
1.7	F	Code	Visual and surface and volumetric	The dissimilar metal welds on the nozzles enumerated in Item 1.4	All dissimilar metal welds on nozzles	The pre-service examination will indicate if these welds are amenable to ultrasonic examination. Welds not amenable to ultrasonic examination will be radiographically examined when techniques are developed and/or the system is drained for other reason
1.8	G-1	Code	Volumetric and visual or surface	33% of all studs and nuts	100% of all studs and nuts	
1.9	G-1	Code	Volumetric	33% of ligaments in vessel	100% of ligaments in vessel	
1.10	G-1	Code	Visual	33% of washers and bushings	100% of washers and bushings	Bushings will be examined only if studs are removed for examination
1.11	G-2	Code	Visual	33% of bolting	100% of bolting	
1.12	H	Code	Volumetric	5% of weld	10% of weld	
1.14	I-1	Code	Visual	Three patches	Six patches	
1.15	N	Code	Visual	Two examinations	Four examinations	
Piping 4.1	F	Code	Visual and surface and volumetric	33% of dissimilar welds	All dissimilar welds	
4.2	J	Code	Visual and volumetric	12% of pipe butt welds including 1 foot of adjacent longitudinal welds	25% of pipe butt welds including 1 foot of adjacent longitudinal welds	It is intended that the volumetric examination be performed by ultrasonics. Weld not amenable to ultrasonic examination will be examined by radiography with the exception of those in pipe which cannot be drained without draining the reactor. These welds will be examined if the reactor is drained for some other reason
4.3	G-1	Code	----------	----------------------	----------------------	Not applicable
4.4	G-2	Code	Visual	33% of bolts, nuts, and studs	100% of bolts, nuts, and studs	
4.5	K-1	Code	Visual and volumetric	12% of supports	25% of supports	
4.6	K-2	Code	Visual	33% of supports	100% of supports	Excluded are supports whose failure satisfies the exclusion criteria of paragraph IS-120 (d)
Pumps						
5.1	L-1	Code	----------	----------------------	----------------------	Not applicable
5.2	L-2	Code	Visual	None	None	A pump will be examined if disassembled for other reasons
5.3	F	Code	----------	----------------------	----------------------	Not applicable
5.4	G-1	Code	Visual and volumetric	33% of bolts, studs, and nuts, including ligaments between threaded holes	100% of bolts, studs, and nuts, including ligaments between threaded holes	
5.5	G-2	Code	Visual	33% of bolts, studs, and nuts	100% of bolts, studs, and nuts	
5.6	K-1	Code	Visual and volumetric	12% of supports	25% of supports	
5.7	K-2	Code	Visual	33% of supports	100% of supports	Excluded are supports whose failure satisfies the exclusion criteria of paragraph IS-120 (d)
Valves						
6.1	M-1	Code	----------	----------------------	----------------------	Not applicable
6.2	M-2	Code	Visual	----------------------	----------------------	Only valves requiring maintenance will be examined
6.3	F	Code	----------	----------------------	----------------------	Included in piping inspections
6.4	G-1	Code	Visual and volumetric	33% of bolts, studs, and nuts, including ligaments between threaded holes	100% of bolts, studs, and nuts, including ligaments between threaded holes	
6.5	G-2	Code	Visual	33% of bolts, studs, and nuts	100% of bolts, studs, and nuts	
6.6	K-1	Code	Visual and volumetric	12% of supports	25% of supports	
6.7	K-2	Code	Visual	33% of supports	100% of supports	Excluded are supports whose failure satisfies the exclusion criteria of paragraph IS-120 (d)

Notes

Area of Interest

Item No.	
1.1	Pressure-Containing Welds in Reactor Pressure Ve
1.2	Pressure-Containing Welds in Remainder of Vessel
1.3	Pressure-Containing Welds, Vessel-to-Flange and Head-to-Flange
1.4	Pressure-Containing Nozzles in Vessels
1.5	Pressure-Containing Welds in Vessel Penetrations
1.6	Pressure-Containing Welds in Vessel Penetrations
1.7	Pressure-Containing Dissimilar Metal Welds
1.8	Pressure-Retaining Bolting
1.9	Pressure-Retaining Bolting
1.10	Pressure Retaining Bolting
1.11	Pressure Retaining Bolting
1.12	Vessel External Supports
1.13	Vessel Interior Clad Surfaces
1.14	Vessel Interior Clad Surfaces
1.15	Interior Surfaces and Internal Components of Reactor Vessels
4.1	Pressure-Containing Dissimilar Metal Welds
4.2	Pressure-Containing Welds in Piping
4.3	Pressure-Retaining Bolting
4.4	Pressure-Retaining Bolting
4.5	Support Members and Structures for Piping, and Pumps
4.6	Support Members and Structures for Piping, and Pumps
5.1	Pressure-Containing Welds in Pump Casings
5.2	Pump Casings
5.3	Pressure-Containing Dissimilar Metal Welds
5.4	Pressure-Retaining Bolting
5.5	Pressure-Retaining Bolting
5.6	Support Members and Structures for Piping, and Pumps
5.7	Support Members and Structures for Piping, and Pumps
6.1	Pressure-Containing Welds in Valve Bodies
6.2	Valve Bodies
6.3	Pressure-Containing Dissimilar Metal Weld
6.4	Pressure Retaining Bolting
6.5	Pressure Retaining Bolting
6.6	Support Members and Structures for Pipin and Pumps.
6.7	Support Members and Structures for Pipin and Pumps.

3.6.E & 4.6.E BASES

Jet Pumps - Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser increases the cross sectional flow area for blowdown following the postulated design basis double-ended recirculation line break. Therefore, if a failure occurs, repairs must be made to assure the validity of the calculated consequences.

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Section 4.6.G.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

3.6.F & 4.6.F BASES

Jet Pump Flow Mismatch

15

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.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PHILADELPHIA ELECTRIC COMPANY (Peach Bottom Atomic Power Station, Unit 2)	))))))	Docket No. 50-277

ORDER FOR MODIFICATION OF LICENSE

I.

Philadelphia Electric Company (PECO or Licensee) is the holder of Facility Operating License No. DPR-44 which authorizes operation of Peach Bottom Atomic Power Station Unit 2 (Unit 2 or the Facility) at steady-state reactor core power levels not in excess of 3295 megawatts thermal (rated power). The Facility is a boiling water reactor (BWR) located at the Licensee's site in Peach Bottom, York County, Pennsylvania.

II.

1. On July 23, 1975, the Nuclear Regulatory Commission (the Commission) issued an "Order for Modification of License" (40 F.R. 32179 of July 31, 1975) which confirmed a plan for limited additional operation of the facility. As explained in the Order of July 23, 1975, the Facility's channel box wear, as indicated by the noise-to-signal ratio recorded by the traversing incore probe (TIP), had exceeded the threshold for remedial action. The remedial action, confirmed by the Order, limited operation of the facility at not more than 40 percent of rated core flow and with a maximum fuel bundle power of 3.35 MWt. In addition, the Order permitted operation up to full flow and power for a brief period of time needed to collect flow vibration data and to conduct fuel preconditioning. The Order further stipulated that the Licensee was to shutdown the facility following approximately 45 equivalent full

flow days from June 21, 1975 unless within that period certain specified tests have been completed which demonstrated the efficacy of the 40% flow limit.

2. By letter dated October 24, 1975, the Licensee proposed a plan, previously discussed with the NRC staff, setting forth a course of remedial action, which would allow operation with flow rates above 40 percent of rated flow and maximum bundle power above 3.35 MWt. The plan would involve shutdown of the reactor and appropriate replacement of worn channel boxes and plugging of the core support plate bypass holes. The reactor was shutdown on October 31, 1975, for visual inspection of the channel boxes and the necessary repairs.

3. By its letter dated September 29, 1975, the Licensee provided details relating to the fuel channel inspection program and the installation of core bypass flow plugs in the lower core plate and supplied analyses to demonstrate the adequacy of the procedures for plug installation. Additionally, by its letter dated October 24, 1975, the Licensee referenced modifications previously approved and implemented at the Duane Arnold and Vermont Yankee reactors.

4. On November 4, 1975, the Commission issued an "Order for Modification of License" (40 F.R. 52671 November 11, 1975) that approved the repair program and authorized the installation of bypass hole plugs in the facility's lower core plate. As discussed in the November 4, 1975 Order, the NRC staff concluded that the plugs will reduce the vibration of the instrument thimbles caused by flow through the bypass holes. By telecon on November 18, 1975, Philadelphia Electric

Company confirmed that the licensee's inspection and repair program was completed. The inspection program resulted in the rejection of 128 channel boxes with unacceptable wear as defined in the repair program. These channel boxes were replaced. Eighty-four channel boxes with indications of wear, but within the criteria of the repair program, were also replaced. Philadelphia Electric Company also confirmed that all flow bypass holes in the core plate were plugged.

5. By letters dated November 7,[1/] 18, and 20, 1975, the licensee provided analyses, including an emergency core cooling performance analysis, for reactor power operation with the plugs installed in the bypass holes. The November 7, 1975 letter supplemented letters of July 9, September 10, October 1 and 30, 1975 related to ECCS analyses.

6. The Commission's staff has reviewed the analyses submitted by the licensee on November 7, 1975 and supplements thereto to support operation with the bypass flow hole plugs installed. As discussed in the Commission's concurrently issued Safety Evaluation for Amendment No. 15 to the license the proposed operation with plugs will require modified limits relating to emergency core cooling system performance. The modified limits specified in the concurrently issued Amendment No. 15 would be based upon an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR Section 50.46.

1/ Copies of the November 7, 1975 filing by the Licensee are available for public inspection in the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and are being placed in the Martin Memorial Library, 189 E. Market Street, York, Pennsylvania.

The amendment would modify various limits established in accordance with the Commission's Interim Acceptance Criteria, and would, with respect to the facility, terminate the further restrictions imposed by the Commission's December 27, 1974 Order for Modification of License (40 F.R. 1772, January 9, 1975), and would impose instead, limitations established in accordance with the Commission's Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 10 CFR Section 50.46. The amendment would also revise the Technical Specifications to permit operation of the facility using operating limits based on the General Electric Thermal Analysis Basis (GETAB) and with the Low Pressure Coolant Injection System modified in accordance with the licensee's application for license amendment dated July 9, 1975 as supplemented.

It also should be noted that plugs identical to those to be used in Peach Bottom Unit 2 have been installed in the Cooper, Vermont Yankee, Duane Arnold and Pilgrim reactors. The plugs installed in Vermont Yankee were removed during a refueling operation after ten months of successful service. No abnormalities or loose pieces were reported. Vermont Yankee has since reinstalled the plugs.

7. Based on our review of the licensee's submittals of November 7, 18, and 20, 1975, and the prior related experience at the Pilgrim and Vermont Yankee reactors, the NRC staff concluded in its concurrently issued Safety Evaluation that operation of Peach Bottom Unit 2 in accordance with the

additional restrictions set forth in Amendment No. 15 to the License would provide reasonable assurance that the public health and safety would not be endangered.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License No. DPR-44 is hereby amended by substituting the following provisions for the provisions set out in the Commission's Orders for Modification of License dated December 27, 1974 and November 4, 1975:

1. Operation of Peach Bottom Atomic Power Station Unit 2 with plugged bypass flow holes is hereby authorized subject to the conditions set forth in the concurrently issued Amendment No. 15 to the Facility License No. DPR-44 incorporating Change No. 15 to the Technical Specifications, and

2. A monitoring program using LPRM and TIP traces and available accelerometers on incore instrument guide tubes shall be performed for the purpose of detecting any instrument tube - channel box interaction.

FOR THE NUCLEAR REGULATORY COMMISSION

Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 28th day of November, 1975.

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE NO. DPR-44

PEACH BOTTOM ATOMIC POWER STATION UNIT 2

DOCKET NO. 50-277

The U. S. Nuclear Regulatory Commission (the Commission) has reviewed a change to the Appendix A Technical Specifications of Facility Operating License DPR-44 as proposed by the licensee, Philadelphia Electric Company. This change would authorize the licensee to operate the Peach Bottom Atomic Power Station Unit 2 in York County, Pennsylvania, with certain revisions to the present limiting conditions for operation as specified in Appendix A of the referenced license. These revisions would result from implementing the Acceptance Criteria for the Emergency Core Cooling Systems for Light Water Power Reactors (ECCS) as specified in Section 50.46 of Part 50 CFR. No revisions to the Environmental Technical Specifications (Appendix B) were requested in connection with the proposed action.

The proposed action would be carried out in conjunction with a planned shutdown for the purpose of correcting a channel box wear problem within the reactor core. The proposed ECCS action would result in a reduced power level of no more than 15 percent for no more than 12 months.

The Commission's Division of Reactor Licensing has evaluated the expected environmental impact of the proposed change. On the basis of this appraisal, the Commission has concluded that an environmental impact statement is not warranted for this particular action. There would be no environmental impact attributable to the proposed action other than those impacts described in the Commission's Final Environmental Statement for Peach Bottom, Units 2 and 3, issued April 1973. The environmental impact appraisal is 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 East Market Street, York, Pennsylvania.

Dated at Rockville, Maryland, this 21 day of November 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Gordon K. Dicker

Gordon K. Dicker, Chief
Environmental Projects Branch 2
Division of Reactor Licensing

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 15 TO DPR-44

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION UNIT 2

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated July 9, 1975, the Philadelphia Electric Company (the licensee) proposed changes to the Technical Specifications in Appendix A of Facility License No. DPR-44. The licensee provided further supportive information by letters dated September 10, 1975, and November 7, 18, and 20, 1975.

The proposed change would incorporate the "Acceptance Criteria for the Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (ECCS) as specified in Section 50.46 of Part 50 CFR into the operating license for Peach Bottom Unit 2. Philadelphia Electric Company is presently licensed to operate Peach Bottom Unit 2 at power levels up to 3,293 megawatts thermal. The proposed action would result in a decrease in the power level amounting to less than 15 percent for no longer than 12 months. The Commission's staff has independently reviewed the expected environmental impact of the proposed action.

2. Environmental Impacts of Proposed Action

In the absence of any significant change in power levels, there would be no change in cooling water requirements. Further, there would be no change in radioactive effluents or thermal effluents from normal operation or post accident conditions. The restrictions on heat generation rates will require careful control of fuel operating history; however, there should be no reduction in total burnup resulting from the revised ECCS evaluation methods. It is not anticipated that the issuance of this change to the Appendix A Technical Specifications would affect the cost-benefit balance nor would it require changes in the Environmental Technical Specifications in Appendix B of the license.

No environmental impacts are expected other than those described in the Commission's Final Environmental Statement for the Peach Bottom Atomic Power Station, Units 2 and 3, issued April 1973. The Commission's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected release rates from the total quantity of nuclear fuel within the reactor units. The proposed action would not affect the total quantity of fuel used at Peach Bottom. No increases in radiation doses to humans or other biota are expected.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than those impacts described in the Final Environmental Statement, issued April 1973. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

DATE: 11/21/75

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

(CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS)

AND

ORDER FOR MODIFICATION OF LICENSE

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION UNIT 2

DOCKET NO. 50-277

1.0 Introduction

Philadelphia Electric Company has proposed to operate Peach Bottom Atomic Power Station Unit 2 under the following conditions:

(1) with plugged bypass flow holes in the lower core support plate as requested in its submittal dated October 24, 1975 and supplements dated November 7 and 20, 1975;

(2) using limits based on the General Electric Thermal Analysis Basis (GETAB) as requested in its submittal dated July 9, 1975 and supplements dated October 1 and 30, November 18 and 20, 1975;

(3) using modified operating limits based on an acceptable evaluation model that conforms with Section 50.46 of 10 CFR Part 50 as requested in its submittal dated July 9, 1975 and supplements dated September 10, October 30, November 7, 18 and 20, 1975; and

(4) with a modification to the low pressure coolant injection system (LPCIS) as requested in its submittal dated July 9, 1975.

2.0 Nuclear Design

The primary nuclear effect caused by plugging the bypass flow holes is an increased bypass void fraction and a reduction in the average in-channel void fraction. The in- and out-of-channel void fraction changes give a net increase in the core average void fraction.

At steady state conditions, the increased bypass void fraction results in a small reduction in the maximum local peaking factor within a fuel bundle and an increase in the local bundle power calculational uncertainty. Another consequence of the reduced bypass flow is a small reduction in the infinite multiplication factor of uncontrolled fuel.

The presence of voids in the bypass region affects the relationship between the travelling incore probe (TIP) signal and the local bundle power. The TIP signal is reduced by the presence of voids and could lead to an underprediction of the peak heat flux. The relationship of the power in the four bundles surrounding a TIP instrument tube and the TIP signal as a function of bypass voids was determined by the General Electric Company (GE) by performing three group, two-dimensional diffusion theory calculations. A correction factor was developed and algorithms for computing the bypass void fraction and for making appropriate corrections in the local bundle power have been incorporated in the process computer.

The uncertainty in the local bundle power caused by bypass voids is taken into account in determining the minimum critical power ratio (MCPR) safety limit. The TIP uncertainty introduced by the bypass voids is zero in the bottom half of the core and increases from 4.08% at the core mid-plane to 5.21% at the core exit.

After the bypass flow holes are plugged, the fuel will be placed in its original core location. The following observations can be made:

(1) the control rod worths are not significantly changed and, consequently, the previous results of the control rod drop analysis remain valid,

(2) the shutdown margin will remain the same as previously analyzed,

(3) the standby liquid control system reactivity insertion rate and magnitude will not be affected.

We conclude that the analysis of the nuclear performance of the plant with plugged bypass holes is acceptable.

3.0 Mechanical Design

The only mechanical design change in the reactor is the use of plugs to fill the bypass flow holes[1]. The plug consists of two

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

stainless steel parts (body and shaft) which are connected by an Inconel spring. The shoulder of the body rests on the top of the core plate along the rim of a one-inch bypass hole and is pressed down by the spring. An equal and opposite force is applied on the shaft. A stainless steel latch is connected to the bottom of the shaft by means of a pin. This latch is free to rotate about the pin and latches the shaft to the core plate. The spring exerts a minimum of 35 pounds on the body and latch and a maximum of 46 pounds (with the worst tolerance combination).

Removal of a plug can be accomplished by applying about 500 pounds of force and deforming the latch plastically. More than 10 plugs were removed in tests performed at the GE test facility with consistent latch deformations without damaging other parts.

Plugs identical to those to be used in Peach Bottom Unit 2 have been installed in the Vermont Yankee, Duane Arnold and Pilgrim reactors. The plugs installed in Vermont Yankee were removed during a refueling operation after 10 months of successful service. No abnormalities or loose pieces were reported. Vermont Yankee has since reinstalled the plugs.

Pressure differentials across the core plate during normal steady state operation and following a steam line break accident are expected to be on the order of 20 to 32 psi. These loads together with the spring preload will produce yielding of the latch in bending but will be significantly below about 500 pounds of force necessary for removing the plug. The 1973 GE full scale flow mockup test shows that, with up to 40 psi differential pressure, there is negligible leakage flow through the plugged holes. No plug vibration was observed during the test and no apparent deformation on the latch was evident after the test. No fatigue and plastic strain ratcheting is expected since the plant power cycle during the anticipated service period will be minimal.

Stainless steel and Inconel are compatible with other reactor internals and are not expected to introduce any unusual oxidation and stress corrosion problems. The flux level at the core plate elevation is estimated to be quite low and an insignificant reduction in ductility due to irradiation is anticipated. GE has performed creep tests with both Inconel springs and stainless steel latches and found that stress relaxation or creep deformation were insignificant. The tests were performed at 550°F.

Philadelphia Electric Company presented to the NRC staff a summary of channel inspections on BWR-2s and BWR-3s. These older plants have instrument tubes similar to Peach Bottom Unit 2, but no bypass flow holes in the core support plate. The bypass flow for these enters through clearances in the assembly end fittings, which is similar to the proposed Peach Bottom Unit 2 configuration with plugged bypass holes. One hundred sixty-four channels (adjacent to instrument tubes and source tubes) were inspected during normal fuel outages in 7 plants. No significant channel wear was observed at the corners adjacent to the instrument tubes.

General Electric has a design criteria for channel box wastage of 0.010 inches for the lower 80 inches of the channel and 0.020 inches for the remaining length. All of the channels (new and old) in the core will meet whis requirement. Channels with observed acceptable wear on the corner will not be reinserted in the core next to an in-core instrument where additional wear could occur during subsequent reactor operation.

Based on a review of the design, the test rig, the installation methods and primarily the previously successful operating experience at Vermont Yankee and Pilgrim, we conclude that the plugs will not fail so as to result in loose parts in the core or result in unplugging of the bypass flow holes. Also, we conclude that the installed plugs will substantially reduce the instrument tube vibration, due to flow through the bypass holes, sufficient to preclude any unacceptable wear for at least one fuel cycle.

4.0 Thermal Hydraulic Design

The licensee's letter of July 9, 1975, and supplements dated October 1 and 30, November 7, 18 and 20, 1975, requested that the license for Peach Bottom Unit 2 be amended to include operating limits based on the General Electric Thermal Analysis Basis (GETAB) described in the General Electric report NEDO-10958[2]. The analyses are based on a core loading with 7 x 7 fuel and with the bypass flow holes plugged.

The proposed changes involve the adoption of a new transition boiling correlation termed GEXL which would replace the Hench-Levy critical heat flux correlation as the basis for determining the thermal-hydraulic conditions which would result in a departure from nucleate boiling. One of the safety requirements for light water cooled nuclear reactors is prevention of damage to the fuel cladding. To prevent damage to the fuel cladding, light water

(2) "General Electric BWR Thermal Analysis Basis (GETAB) Data Correlation and Design Application," NEDO-10958 and NEDE-10958 (Proprietary), November 1973.

cooled reactors must be designed and operated such that during normal operation and anticipated transients the heat transfer rate from the fuel cladding to the coolant are sufficient to prevent overheating of the fuel cladding. Although transition boiling would not necessarily result in damage to boiling water reactors (BWR) fuel rods, historically it has been used as a fuel damage limit because of the large reduction in heat transfer rate when film boiling occurs. A critical power ratio (CPR) is defined which is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest. The MCPR is the critical power ratio corresponding to the most limiting fuel assembly in the core. The fuel assembly power at which boiling transition would be predicted to occur, using the GEXL correlation, is termed the critical power. The GEXL transition boiling correlation is more recent than the previously used Hench-Levy critical heat flux correlation and is based on an extensive data base. The methods for applying the GEXL correlation to determine thermal limits has been termed the General Electric Thermal Analysis Basis (GETAB). We have accepted the GEXL correlation and the GETAB methods in a previous report[3] as a basis for establishing the safety limit and limiting conditions for operation related to prevention of fuel damage for general Electric BWR 8 x 8 and 7 x 7 fuel. To apply GETAB to the Technical Specifications involves establishing (1) the fuel damage safety limit, (2) limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and (3) limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied. We have evaluated the thermal margins for Peach Bottom Unit 2 based on the NEDO-10958 report[2] and plant specific input information provided by the licensee. As described below, we conclude that the calculated consequences of the anticipated abnormal transients do not violate the thermal and plastic strain limits of the fuel.

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(3) "Review and Evaluation of GETAB (General Electric Thermal Analysis Basis) for BWRs," Division of Technical Review, Directorate of Licensing, United States Atomic Energy Commission, September, 1974.

4.1 Fuel Cladding Integrity Safety Limit-MCPR

The safety limit on MCPR is based on the GETAB statistical analysis which assures that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The uncertainties in the core and system operating parameters and the GEXL correlation, Table 5-1 of the licensee submittal, [1] combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. These uncertainties are the same as or more conservative than those reported in NEDO-10958[2] and NEDO-20340[4]. The uncertainty of the bypass void effect on TIP readings accounts for the additional uncertainty due to the bypass void content resulting from plugging the core support plate bypass holes. The reactor core selected for the GETAB statistical analyses is a typical core (251" diameter vessel/764 fuel assemblies). This typical core is of the same reactor class as the Peach Bottom Unit 2 core. The bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during operation of Peach Bottom Unit 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, a MCPR of 1.06, is acceptable for the Peach Bottom Unit 2 plant current fuel cycle with plugged bypass holes.

4.2 LIMITING CONDITION FOR OPERATION - MCPR

Various transient events will reduce the required operating limit 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 those transient analyses which show a significant decrease in MCPR. The types of transients evaluated were losses of flow, pressure and power increases, and coolant temperature decreases. The main factors affecting the plant transient analyses are the moderator void coefficient of reactivity, the Doppler coefficient of reactivity, and the full power scram reactivity function. The Doppler coefficient of reactivity is affected by the changes in the moderator density in the fuel channel and bypass region primarily through changes in the Dancoff-Ginsburg rod shadowing effect. This effect is small and insignificantly affects the Doppler coefficient of reactivity. The full power scram

(4) General Electric, "Process Computer Performance Evaluation Accuracy," NEDO-20340, and Amendment 1, NEDO-20340-1, dated June, 1974 and December, 1974.

reactivity function for the end-of-cycle with plugged bypass flow holes indicates a total scram worth of -35.60 dollars. This is less total scram worth than the previously determined value and is due primarily to a recalculation of the end-of-cycle reactivity. However, the initial scram reactivity addition rate which is important to transient analyses is about the same as previously used.

The moderator void coefficient of reactivity used in the safety analyses of Peach Bottom Unit 2 with plugged bypass flow holes is more negative than used in the FSAR for two reasons. The first cause is a renormalization of the void coefficient calculations based on analyses of operating BWR data. This effect, of the order of 15 ro 20 percent, is unrelated to the plugging of the bypass flow holes. The second cause is the increase in the amount of voids present in the bypass region after the bypass flow holes are plugged. The most limiting transient is a rod withdrawal error transient. The analysis was initiated from rated power and the transient was terminated by the Rod Block Monitor. The decrease in MCPR is 0.15 which is the limiting change in thermal margin. As a result, the steady state MCPR must be equal to or greater than 1.21 to satisfy the safety limit MCPR of 1.06. The calculated change in MCPR for the second most severe transient, loss of feedwater heating, is 0.14.

We conclude that the proposed MCPR value of 1.21, the limiting condition for operation, is acceptable for Peach Bottom Unit 2 with plugged bypass holes.

4.3 LIMITING CONDITIONS FOR OPERATION, MCPR, AT LESS THAN RATED POWER AND FLOW

The limiting transient at less than rated power and flow condition is the recirculation pump speed control failure. The Technical Specifications would require the licensee to maintain MCPR greater than 1.21 times the K_f factor for core flows less than rated. The K_f factor curves were generically derived and assure that most limiting transient, a speed control increase, occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the limiting conditions for operation, MCPR, at less than rated power and flow are acceptable.

5.0 OVERPRESSURE TRANSIENTS

The licensee has reanalyzed the worst case overpressure transient for operation with the bypass flow holes in the lower core support plate plugged. The transient analyzed was the closure of all main steam isolation valves with a high neutron flux scram. The assumptions used in the analysis were: Operation at 105% of power, end-of-cycle scram reactivity insertion rate curve and one safety/relief valve fails to operate. The reanalysis predicts a peak pressure at the vessel bottom of 1292 psig which is 83 psi below the code allowable pressure. The reanalysis and calculated pressure margin are acceptable.

6.0 EMERGENCY CORE COOLING SYSTEMS ANALYSIS

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's Regulations "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 reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of Section 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.

On July 9, 1975 the licensee submitted an evaluation of the ECCS performance for the design basis pipe break for Peach Bottom Units 2 and 3 along with an amendment requesting changes to the Technical Specifications for Peach Bottom Units 2 and 3 to implement the results of the evaluation.[5] The licensee incorporated further information relating to the details of the ECCS evaluation, by referencing an appropriate lead plant analysis,[6] to show compliance with the Section 50.46 criteria and Appendix K to 10 CFR Part 50.

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of the Peach Bottom Atomic Power Station Units 2 and 3 dated December 27, 1974.

The background of the staff review of the GE ECCS models and their application to Peach Bottom Unit 2 is described in the staff Safety Evaluation Report (SER) for these facilities dated December 27, 1974 issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October, 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Peach Bottom Unit 2 evaluation which is covered by this SER properly conforms to the accepted model.

With respect to reflood and refill computations, the Peach Bottom Unit 2 analysis was based on a modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations, as described in the December 27, 1974 SER. The Peach Bottom Unit 2 evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

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(5) Letter from E. J. Bradley, Philadelphia Electric Company, to B. C. Rusche, USNRC, dated July 9, 1975.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August, 1974. We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

The additional analyses (performed on the lead plant, Brunswick Unit 2[6] and incorporated by reference) supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as the worst single failure for the Peach Bottom Unit 2 design. The limiting break continues to be the complete severence of the recirculation suction line assuming a failure of the LPCI injection valve.

The November 7, 1975 submittal contains the ECCS analysis for operation with the plugged bypass flow holes. The results for this ECCS Appendix K calculation show a peak cladding temperature of 2200°F; a peak local oxidation of 7.5%, and a maximum core average hydrogen generation of 0.2% for the worst large size pipe break assuming failure of the LPCI injection valve (the worst single failure). The calculations show a peak cladding temperature of 1770°F, and a peak local oxidation of less than 1.0% for the worst small size pipe break area (0.07 ft^2) assuming failure of the HPCI system (the worst concurrent single failure). We have reviewed the evaluation of ECCS performance submitted by PECO for Peach Bottom Unit 2 with plugged bypass holes and conclude that the evaluation has been performed wholly in conformance with the requirements of Section 50.46. Therefore, operation of the reactor would meet the requirements of Section 50.46 provided that operation is limited to the maximum planar linear heat generation rates (MAPLHGR) of figures 8-10A, and 8-10B of the PECO submittal dated November 7, 1975, and to a minimum critical power ratio (MCPR) greater than 1.17. The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.17. However, a more restrictive technical specification limits operation of the reactor to a MCPR of 1.21 based on consideration of a rod withdrawal error transient. A statement should be added to the bases for the limiting condition of operation indicating the MCPR used in the ECCS performance evaluation.

(6) Brunswick 2 Loss-of-Coolant Analysis, Conformance with 10 CFR Part 50 Appendix K, dated June, 1975.

Certain operating conditions presently allowed are not in conformance with the analysis performed in accordance with Section 50.46. Certain changes must be made to the proposed technical specifications to conform with the evaluation of ECCS performance. 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.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small size line breaks the Technical Specifications will not permit continuous operation with any ADS valve out of service except as with other ECCS equipment one valve may be out of service for seven days.

Based on the above, we conclude that with the Technical Specifications discussed above operation of the reactor will meet the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's Regulations.

7.0 LPCIS Modification

The NRC staff previously issued a safety evaluation for the LPCIS modification at Peach Bottom dated November 5, 1975[7]. The acceptability of the LPCIS modification was addressed in that document with exception that the electrical distribution changes and wiring changes proposed by the licensee had not been completely evaluated at that time.

In the existing onsite power system for Peach Bottom, the train A Low Pressure Coolant Injection (LPCI) valves are powered from swing bus N210025A which has swing capabilities between MCC 20B36 and MCC 20B38 connected to diesel generators A & C, respectively. The train B LPCI valves are powered from swing bus N210025B which has swing capabilities between MCC 20B37 and MCC 20B39 connected to diesel generators B & D, respectively.

(7) Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment No. 14 to Facility Operating License No. DPR-44, November 5, 1975.

Our review of this swing bus concept has shown it to be unacceptable as proposed. There are certain undetectable failures within the transfer circuitry that, if present when the bus transfer were required, would prevent the bus from transferring to its alternate source. There are also certain single failures that could tie the two diesel generators together through either of the swing buses. We informed the applicant that this design was unacceptable and that separate and independent buses would be required to bring the 480-volt portion of the onsite emergency power system into conformance with the recommendations of Regulatory Guide 1.6. However, we will allow the licensee reasonable time to institute their proposed design after the submission (and acceptance by the NRC staff) of their modified design.

In order to alleviate the problem of potential undetected failure prior to completion of required electrical modifications, we shall place a technical specification requirement on these transfer circuits that they be tested bi-monthly and that the test procedure used be submitted for our approval. Within one month after completion of required mechanical modifications, we will also require the licensee to perform a functional test of the existing electrical scheme to verify that the two independent undervoltage schemes on each swing bus have been set with sufficient margin such that the swing buses will not be transferred to their alternate source when the maximum voltage dip occurs during diesel generator load sequence and verification that the protective relaying and circuit breaker coórdination between each swing bus and the diesel generators are within their design specification. The licensee has agreed to make the necessary modifications to his present design so that the modified design meets Regulatory Guide 1.6. We find this commitment to be an acceptable short term solution to the electrical power system problem. We will review the design changes proposed to meet Regulatory Guide 1.6 when they are submitted.

The loop selection logic circuitry of the LPCI system will be removed from the control room panels. Removal of this logic circuitry allows both injection valves to open, given an accident signal, no matter where the pipe break is located. This situation of opening both injection valves requires that the RHR crosstie valve remain closed during normal plant operations and accident conditions. The applicant has proposed that the keylock switch on the control room panel which operates the crosstie valve will be changed from keylock open to keylock close, and the crosstie valve circuit breaker at the motor control center cubical is padlocked open with the valve closed. An annunciator will be added to alarm whenever the crosstie valve is open. We find these proposed changes to be an acceptable method of assuring that this valve will remain closed during normal plant operation and accident conditions and are, therefore, acceptable.

Due to the elimination of the loop selection logic, the accident initiation signals have been rewired to direct (1) both LPCI injection valves to open, (2) both recirculation loop discharge valves to close when reactor pressure decreases to an appropriate setting and (3) LPCI pumps to start from two divisions instead of one (i.e., each pump and valve will receive a one-out-of-two logic initiation) upon detection of accident conditions.

The LPCI system redundant injection valves, pumps and recirculation valves are controlled by a-c control power relays in their control circuitry. These relays are in turn controlled by redundant 125-volt d-c output relays provided in each actuation train in the LPCI logic panels. This assures that failure of the 120-volt d-c power supply of either train will not prevent operation of any valve and pump in either train. Separation has been provided within the logic panels and wiring between the two logic panels is run through separate conduit. Separation of A & B circuits is maintained by the conduit so that any assumed failure of a conduit run will not prevent the operation of the redundant or associated control systems. We conclude that these design changes do not compromise the separation and independence of the two safety trains and are acceptable.

With regard to the emergency electrical distribution system provided for operation of the modified LPCI system, we find that the use of swing buses is not acceptable since there are certain undetectable failures that can exist within the transfer circuitry that if present when the bus transfer is required, would prevent the bus from transferring to its alternate source. However, we conclude that the operation of Peach Bottom Unit 2 with the LPCIS modification completed and with the existing emergency power distribution system is acceptable on a short term basis since Technical Specification 4.9.A.3 requires the licensee to test the swing buses every two months. The licensee has agreed to make the necessary modifications to his present design so that the modified design meets Regulatory Guide 1.6. We find this commitment to be an acceptable short term solution to the electrical power system problem. We will review the design changes proposed to meet Regulatory Guide 1.6 when they are submitted.

8.0 Inspection Program

During the November, 1975 outage Philadelphia Electric Company (PECO) performed an inspection of all Peach Bottom Unit 2 fuel bundle channel boxes from locations adjacent to in-core instrument tubes. The inspection revealed 128 channel boxes with an unacceptable amount of wear. These channel boxes were replaced. Eighty-four (84) channel boxes were worn less than the amount established for replacement. PECO replaced these channel boxes as well, but retained them for future use in locations which are not adjacent to instrument channels. There were no perforations of any channel box. One channel box inspected had a 12 inch hairline crack in the lower 80 inches of the channel. The crack had no separation and displayed 3/8 inch maximum wear width at the top of the crack.

The core instrument tube adjacent to the one cracked channel was inspected and was found to be acceptable. Based on the results of the inspection and replacement program we have concluded that the condition of the installed channel boxes and in-core instrument tubes are acceptable.

9.0 Technical Specification and License Changes

The specific proposed changes to the Technical Specifications and license which we consider acceptable are itemized below:

Section 1.0 Definitions

Subsections would be added which would define maximum total peaking factor, critical power ratio and minimum critical power ratio. We would also modify the definition of minimum critical power ratio from that proposed by PECO. We have discussed these changes with PECO and they do not object. The new definitions are needed to be consistent with the revised format of the limits discussed below.

Section 1.1 Fuel Cladding Integrity Safety Limits

Subsection 1.1.A for operation with reactor primary system pressure greater than 800 psig or core flow greater than or equal to 10% of rated would be revised to state a MCPR safety limit.

Subsection 1.1.B would be revised to limit core thermal power to 25% or less of rated thermal power when reactor pressure is less than or equal to 800 psig or core flow is less than 10% of rated.

These changes are consistent with the GETAB analyses discussed earlier in this safety evaluation.

Sections 2.1.A.1 and 2.1.B APRM Flux Scram and Rod Block Trip Settings

The existing specifications would be changed to reflect a MCPR limitation rather than a MCHFR limitation so that the specifications would be consistent with the GETAB analysis.

Section 3.3.B.5 Control Rods Limiting Condition for Operation

The existing specification 3.3.B.5.c would be revised from a MCHFR limitation to a MCPR limitation so that the specification would be consistent with the GETAB analysis.

Section 3.5.E. Automatic Pressure Relief Subsystem Limiting Condition for Operation

The existing specification allows continued operation for up to 30 days after one relief valve of the automatic depressurization system (ADS) is made or found to be inoperable. The loss-of-coolant accident analyses submitted in accordance with 10 CFR 50.46 were based on the assumption that all ADS valves operated for small line breaks with HPCI failure. Because the analyses submitted do not support extended periods of operation with one ADS valve out of service, we would reduce the time the valve can be out of service to 7 days. This is consistent with out of service times for other ECCS equipment. We have discussed this change with the PECO staff and they did not object.

Section 3.5.I Average Planar LHGR Limiting Conditons for Operation

The average planar linear heat generation limits would be revised to be consistent with the analyses performed in accordance with 10 CFR Part 50.46 for operation with plugged bypass holes in the lower core plate.

Section 3.5.R Minimum Critical Power Ratio (MCPR)

Subsection 3.5.K would be added to place operating MCPR limits on the fuel. The limits are consistent with the GETAB analyses discussed earlier in this report and require a MCPR more limiting than that needed to satisfy the requirements of the LOCA analysis.

Section 3.6.F Jet Pump Flow Mismatch

This section would be modified to limit operation to a period of twenty-four hours when one recirculation loop is out of service. The additional restriction would be consistent with the analysis discussed earlier in this report.

Section 4.9 Auxiliary Electrical System

Additional surveillance requirements would be added to check the operability of the swing buses which power the LPCIS injection valves as discussed earlier in this report.

GETAB Bases

The bases would also be changed to discuss the justification for the revised specifications itemized above.

Proposed Reporting Exclusion

The specifications proposed by the licensee would exclude reporting, as a reportable occurrence, operation in excess of the limiting MAPLHGR, local LHGR and MCPR values providing corrective action was taken upon discovery. We would not include these provisions. We believe that such events should be reported in conformity with the Technical Specifications.

APRM Flux Trip, APRM Rod Block and RBM Rod Block Settings

The specifications proposed by the licensee would change the primary coolant flow referenced trip settings to provide greater operating margin. The licensee has not fully justified the proposed revisions. Therefore we would retain the existing flow referenced limits.

Instrument Tube-Channel Box Interaction Surveillance

Excessive instrument tube-channel interaction previously has been determined from the noise level in the LPRM signals. The plugged bypass flow holes are expected to affect the noise content of the LPRM signals. The noise content in the 1.4 to 3 Hz frequency range caused by vibration of the LPRM instrument tube should be reduced relative to the power dependent noise content. Some increase in the boiling noise, 5 to 50 Hz range, is expected because of boiling in the bypass water region.

Before the plant was shutdown in 1975, extensive LPRM time traces, TIP traces, and power spectral density (PSD) calculations were obtained for a number of combinations of power and flow. These data will provide a basis for evaluating the efficiency of plugging the bypass flow holes. After reactor startup, comparison of similar measurements with pre-shutdown data will be made to confirm that the mechanical vibration of the instrument tubes has been substantially reduced.

The licensee has agreed to provide NRC with a plan for monitoring instrument tube-channel box interaction. The monitoring would be performed on a periodic basis using the available LPRM and TIP traces and the available accelerometers on core instrument guide tubes. This monitoring program should be required by the licensee.

9.0 CONCLUSION

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

DATED: November 28, 1975

For further details with respect to this action, see (1) the application for amendment dated July 9, 1975, September 10, October 30, November 7, 18 and 20, 1975, (2) Amendment No. 15 to License No. DPR-44, with Change No. 15, (3) the Commission's related Safety Evaluation, and (4) the Commission's Negative Declaration dated (which is also being published in the Federal Register) and associated Environmental Impact Appraisal. 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 Reactor Licensing.

Dated at Bethesda, Maryland, this 28th day of November, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

D. M. Elliott, Acting Chief
Operating Reactors Branch #3
Division of Reactor Licensing

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-277

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC & 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. 15 to Facility Operating License No. DPR-44 issued to Philadelphia Electric Company, Public Service Electric & 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 2, located in Peach Bottom, York County, Pennsylvania. The amendment is effective as of its date of issuance.

The amendment modifies the provisions in the Technical Specifications relating to Limiting Conditions for Operation associated with the Emergency Core Cooling System (ECCS) and Reactor Core Power Distribution Limits; and provides for modification of the ECCS to improve its performance in accordance with the licensees' application for amendment dated July 9, 1975.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on August 18, 1975 (40 F.R. 34647). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.