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BJones (8)
BScharf (10)
JWetmore
ACRS (16)
CMiles
RDiggs
TERA
NSIC

MARCH 26 1980

Docket Nos. 50-277
and 50-278

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

Dear Mr. Bauer:

The Commission has issued the enclosed Amendment Nos. 65 and 64 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Unit Nos. 2 and 3. These amendments revise the Technical Specifications relating to core monitoring, core flooding and cooling system operability requirements and are in response to your application dated January 14, 1980.

During our review, we concluded that a minor modification in the wording that you proposed for Section 3.10.B was desirable to be more explicit in what was required for core monitoring. The changes which we proposed were discussed with and accepted by your staff.

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

Sincerely,

Original Signed by
T. A. Ippolito

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

Enclosures:
1. Amendment No. 65 to DPR-44
2. Amendment No. 64 to DPR-56
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

8004230409

OFFICE	ORB #3	ORB #3	AD:ORP	OELD	ORB #3	
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DATE	3/ /80	3/26/80	3/ /80	3/ /80	3/ /80	

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

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 65
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et al., (the licensee) dated January 14, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 65, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8004230414

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 26, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 65

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

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

Remove	Insert
35	35
47	47
132	132
139	139
165	165
227	227
228	228
231	231
232*	232

2. Add the following new pages:

132a
228*

*Pages 228a and 232 are included due to redistribution of material on the revised pages.

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective

To assure the operability of the reactor protection system.

Specification:

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

SURVEILLANCE REQUIREMENTS

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

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and present inadvertant criticality.

When there is no fuel in the reactor, the scram serves no function; therefore, the reactor protection system is not required to be operable.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference subsection 7.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e. an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

The APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in

LIMITING CONDITIONS FOR OPERATION

3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generators shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.

4. During a refueling outage, fuel and LPRM removal and replacement may be performed provided at least one of the following conditions below is satisfied:

SURVEILLANCE REQUIREMENTS

4.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable, al low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generators shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.F.3 (Cont'd)	4.5.F.2 (Cont'd)

a. Both core spray systems and the LPCI system shall be operable except that one core spray system or the LPCI system may be inoperable for a period of thirty days, or

b. The reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained at least 21 feet over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks and no work is being performed which has the potential for draining the reactor vessel.

3.5.E BASES (Cont'd.)

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

F. Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling capability is available at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Additionally, the specification provides minimum core flooding requirements during refueling operations. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

G. Maintenance of Filled Discharge Pipe

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

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment system.

Specification:

A. Primary Containment

1. Whenever the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified by 3.7.A.2, or when inoperability of the core spray systems, the LPCI and containment cooling subsystems is permissible as provided for in 3.5.F.3 and 3.5.F.4.b.

   a. Minimum water volume- 122,900 ft ^{3}

   b. Maximum water volume- 127,300 ft ^{3}

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

1. The suppression chamber water level and temperature shall be checked once per day.

2. a. Whenever there is indication of relief valve operation (except when the reactor is being shutdown and torus cooling is being established) or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

3. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation

4. A visual inspection of the suppression chamber interior, including water line regions shal be made at each major refueling outage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.10.A.5.b (Cont'd)

directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are available as defined in specification 3.10.B.

6. Any number of control rods may be withdrawn or removed from the reactor core provided the following conditions are satisfied:

a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

4.10.A

B. Core Monitoring

1. Except as specified in 3.10.B.2, 3.10.B.3, 3.10.B.4 and 3.10.B.5, during core alterations two SRM's shall be operable, one in the core quadrant where fuel or controls rods are being moved and one in the adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

B. Core Monitoring

1. Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B (Cont'd)	4.10.B (Cont'd)
a. The SRM shall be inserted to the normal operating level. (Use of special movable, dunking type detectors during fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)	2. Prior to unloading or reloading of fuel as provided for in sections 3.10.B.2 & 3.10.B.3, the SRM's shall be functionally tested. Prior to unloading of fuel, the SRM's should also be checked for neutron response.
b. The SRM shall have a minimum of 3 cps with all rods fully inserted in the core.	
2. Prior to unloading of fuel, the SRM's shall be proven operable as stated above; however, during unloading of fuel, the SRM count rate may drop below 3 cps, provided all control rods are full inserted and rendered electrically inoperable with the exception of the following provision. Individual control rods outside the periphery of the then existing fuel matrix may be electrically armed and moved after all fuel in the cell containing that control rod have been removed from the reactor core.	
3. Prior to reloading of fuel, two, three, or four fuel assemblies may be returned to their previous core positions adjacent to each of the 4 SRM's to obtain the required 3 cps. Until these assemblies are loaded, the SRM 3 cps count rate is not required.	
4. The SRM 3 cps count rate is not required with all fuel removed from the core.	
5. During the unloading and reloading of fuel, intermediate arrays of fuel shall always contain at least one SRM.	

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8 1/2' above the top of the fuel.

D. Heavy Loads Over Spent Fuel

Loads in excess of 1000 lbs (excluding the rigging and transport vehicle) shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

3.10 BASES (Cont'd)

The requirements for SRM Operability during these core alterations assure sufficient core monitoring.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and insures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

During unloading of fuel, it is permissible to allow the SRM count rate to decrease below 3 cps. Since all fuel moves during core unloading will reduce reactivity, the lower number of counts will not present a hazard. Requiring the SRM's to be functionally tested prior to fuel removal assures that the SRM's will be operable at the start of fuel removal. The daily response check of the SRM's ensures their continued operability until the count rate deminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during fuel removal, provided all rods that control fuel are fully inserted and electrically disarmed.

During core loading, the loading of adjacent assemblies around the four SRM's before attaining the 3 cps is permissible because these assemblies were in a subcritical configuration when they were removed and therefore will remain subcritical when the same assemblies are placed back into their previous positions. Since specification 3.10.A.2 requires that all control rods be fully inserted prior to loading fuel, inadvertent critical whenity is precluded during core loading.

C. Spent Fuel Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 8 1/2' above the top of the fuel is established because it provides adequate shielding and is well above the level to assure adequate cooling.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

B. Core Monitoring

Requiring the SRM's to be functionally tested prior to any core alteration assures that the SRM's will be operable at the start of that alteration. The daily response check of the SRM's ensures their continued operability.

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

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

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et al., (the licensee) dated January 14, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 64, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 26, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 64

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

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

Remove	Insert
35	35
47	47
132	132
139	139
165	165
227	227
228	228
231	231
232*	232

2. Add the following new pages:

132a
228a*

*Pages 228a and 232 are included due to redistribution of material on the revised pages.

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective

To assure the operability of the reactor protection system.

Specification:

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

SURVEILLANCE REQUIREMENTS

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 maximum fraction of limiting density 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 maximum fraction of limiting power density exceeds the fraction of rated power.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and present inadvertant criticality.

When there is no fuel in the reactor, the scram serves no function; therefore, the reactor protection system is not required to be operable.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference subsection 7.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e. an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

The APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generators shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.

4. During a refueling outage, fuel and LPRM removal and replacement may be performed provided at least one of the following conditions below is satisfied:

4.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable,al low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generators shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.F.3 (Cont'd)	4.5.F.2 (Cont'd)

a. Both core spray systems and the LPCI system shall be operable except that one core spray system or the LPCI system may be inoperable for a period of thirty days, or

b. The reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained at least 21 feet over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks and no work is being performed which has the potential for draining the reactor vessel.

3.5.E BASES (Cont'd.)

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

F. Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling capability is available at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Additionally, the specification provides minimum core flooding requirements during refueling operations. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

G. Maintenance of Filled Discharge Pipe

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

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment system.

Specification:

A. Primary Containment

1. Whenever the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified by 3.7.A.2, or when inoperability of the core spray systems, the LPCI and containment cooling subsystems is permissible as provided for in 3.5.F.3 and 3.5.F.4.b.

   a. Minimum water volume- 122,900 ft ^{3}

   b. Maximum water volume- 127,300 ft ^{3}

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

1. The suppression chamber water level and temperature shall be checked once per day.

2. a. Whenever there is indication of relief valve operation (except when the reactor is being shutdown and torus cooling is being established) or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

3. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation

4. A visual inspection of the suppression chamber interior, including water line regions shal be made at each major refueling outage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A.5.b (Cont'd)	4.10.A

directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are available as defined in specification 3.10.B.

6. Any number of control rods may be withdrawn or removed from the reactor core provided the following conditions are satisfied:

a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

B. Core Monitoring

1. Except as specified in 3.10.B.2, 3.10.B.3, 3.10.B.4 and 3.10.B.5, during core alterations two SRM's shall be operable, one in the core quadrant where fuel or controls rods are being moved and one in the adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

B. Core Monitoring

1. Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.

LIMITING CONDITIONS FOR OPERATION

3.10.B (Cont'd)

a. The SRM shall be inserted to the normal operating level. (Use of special movable, dunking type detectors during fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

b. The SRM shall have a minimum of 3 cps with all rods fully inserted in the core.

2. Prior to unloading of fuel, the SRM's shall be proven operable as stated above; however, during unloading of fuel, the SRM count rate may drop below 3 cps, provided all control rods are full inserted and rendered electrically inoperable with the exception of the following provision. Individual control rods outside the periphery of the then existing fuel matrix may be electrically armed and moved after all fuel in the cell containing that control rod have been removed from the reactor core.

3. Prior to reloading of fuel, two, three, or four fuel assemblies may be returned to their previous core positions adjacent to each of the 4 SRM's to obtain the required 3 cps. Until these assemblies are loaded, the SRM 3 cps count rate is not required.

4. The SRM 3 cps count rate is not required with all fuel removed from the core.

5. During the unloading and reloading of fuel, intermediate arrays of fuel shall always contain at least one SRM.

SURVEILLANCE REQUIREMENTS

4.10.B (Cont'd)

2. Prior to unloading or reloading of fuel as provided for in sections 3.10.B.2 & 3.10.B.3, the SRM's shall be functionally tested. Prior to unloading of fuel, the SRM's should also be checked for neutron response.

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8 1/2' above the top of the fuel.

D. Heavy Loads Over Spent Fuel

Loads in excess of 1000 lbs (excluding the rigging and transport vehicle) shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

3.10 BASES (Cont'd)

The requirements for SRM Operability during these core alterations assure sufficient core monitoring.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and insures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

During unloading of fuel, it is permissible to allow the SRM count rate to decrease below 3 cps. Since all fuel moves during core unloading will reduce reactivity, the lower number of counts will not present a hazard. Requiring the SRM's to be functionally tested prior to fuel removal assures that the SRM's will be operable at the start of fuel removal. The daily response check of the SRM's ensures their continued operability until the count rate deminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during fuel removal, provided all rods that control fuel are fully inserted and electrically disarmed.

During core loading, the loading of adjacent assemblies around the four SRM's before attaining the 3 cps is permissible because these assemblies were in a subcritical configuration when they were removed and therefore will remain subcritical when the same assemblies are placed back into their previous positions. Since specification 3.10.A.2 requires that all control rods be fully inserted prior to loading fuel, inadvertent critical whenity is precluded during core loading.

C. Spent Fuel Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 8 1/2' above the top of the fuel is established because it provides adequate shielding and is well above the level to assure adequate cooling.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

B. Core Monitoring

Requiring the SRM's to be functionally tested prior to any core alteration assures that the SRM's will be operable at the start of that alteration. The daily response check of the SRM's ensures their continued operability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 65 AND 64 TO FACILITY LICENSE NOS. DPR-33 AND DPR-56

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

PEACH BOTTOM ATOMIC POWER STATION UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

1.0 Introduction

By letter dated Jnauary 14, 1980, Philadelphia Electric Company (the licensee) requested amendments to Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. The proposed amendments would revise the Technical Specifications relating to core monitoring, core flooding and cooling system operability requirements for assuring the safe conduct of refueling operations under various plant conditions. The effect of the revised Technical Specifications would be (1) to allow dewatering of the suppression chambers (torus) during refueling outages, provided adequate core flooding inventory is available and no operations are being performed which have a potential for draining the reactor vessel, (2) to allow unloading and reloading the core under special conditions without having 3 counts per second (cps) on the source range monitors (SRMs) and (3) to allow the reactor protection system instrumentation to be inoperable when the reactor is completely defueled.

2.0 Discussion

When the plant is in a cold shutdown condition, the vessel is not pressurized. In such circumstances, the "suppression" function of the suppression pool is not needed. However, the suppression pool (also called the "torus) is also the source of water for the low pressure emergency core cooling systems. The present Technical Specifications require these systems to be operable and thus indirectly require the torus to contain a minimum quantity of water.

2.2 Evaluation

The core spray (CS) system is one of the low pressure emergency core cooling systems (ECCS) along with the low pressure coolant injection (LPCI) system, which protects the core in case of a loss-of-coolant accident (LOCA) when the reactor pressure is low. Draining the torus will remove the normal source of water for both CS pumps and the

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residual heat removal (RHR) pumps when in the LPCI mode. The Containment Cooling System (Suppression Pool Cooling) will also be deactivated with no water in the torus.

The revised Technical Specifications will require that the vessel head be removed, the spent fuel pool gates be removed, and the entire cavity be flooded to a point at least 21 feet over the top of the fuel in the spent fuel storage racks. Even if some water were to be drained from the vessel, this requirement will provide a significant inventory of water always available to cool the core without the use of pumps and valves. (Because of the height of the spent fuel pool gate threshold, the stored fuel cannot be uncovered by a vessel leak).

We have examined an analogous change on another docket[(2)], and found that a leak to the drywell would flood the torus sufficiently to provide NPSH to the LPCI and CS pumps well before the water level would drop to the upper edge of the vessel.

In reviewing the proposed new Technical Specifications, we have considered the possible ways in which water could be lost from the reactor vessel. These include inadvertent operation of valves or pumps in such a way that water flows from the core and the break of a line connected to the vessel.

The first mechanism for a loss of reactor cooling water is a possible error in which a pump is started or a valve is opened such that there is a decrease in the amount of available water to protect the core. The licensee has addressed this mechanism and has concluded that sufficient controls are in effect to preclude the possibility of this mechanism existing when no work is being done which has the potential for draining the reactor vessel. Thus, no requirements need be placed on the operability of CS and Containment Cooling Systems with the torus drained, unless work is being done which has the potential for draining the reactor vessel.

The second mechanism by which reactor water could be lost is by a pipe break. (Because the system is not pressurized in the refueling mode, the probability of a significant pipe break is very low.) Should a pipe break occur within the primary containment, the extra water inventory above the reactor would mitigate the effects of the break in the short term, and reflood the torus for long-term mitigation, as described above.

Should a break occur in a pipe outside of the primary containment, the spilled water would not run into the torus. Most of these pipes are equipped with check valves to prevent outward flow, and also penetrate the vessel above the top of the core. Thus, a break in these lines cannot cause core uncovery.

However, certain control and instrumentation lines (e.g., jet pump pressure taps and control rod drive hydraulic lines) are exceptions to this. (Most of these are equipped with excess flow check valves to protect against breaks.) These lines are so small that many days would be required to drain the reactor, and considerable time would be available to diagnose, mitigate, and repair the break.

The remaining exception is the reactor water cleanup system (RWCU). This system is normally in operation, even during shutdown, and has an inlet line which does not have check valve isolations. This system is equipped with automatic isolation capabilities (called "group 3 isolation"). The isolation is initiated on many diverse signals, including differential flow mix-match and low reactor water level. These isolations are intended to mitigate the consequences of a RWCU pipe break during operation, not just during shutdown. The differences during the proposed circumstances are (a) the extra water inventory in and above the reactor replaces the flooding capability of low pressure ECCS, and (b) some of the temperature-initiated isolation will not work. However, the temperature-initiated isolations are intended for RWCU protection; they are not necessary for pipe break protection. Therefore, we find protection from breaks in the RWCU system to be adequate.

In sum, neither operator error nor pipe breaks can credibly lead to unacceptable consequences. Therefore, we find the proposed changes to the Technical Specifications acceptable.

3.0 Inoperable Reactor Protection System

When the reactor core has had all of the fuel removed, the Reactor Protection System ("scram" system) serves no protective functions. Therefore, we find the licensee's proposal to allow inoperability of this system with no fuel in the vessel to be acceptable.

4.0 SRM Count Rate

4.1 Discussion

During any core alteration, and especially during core loading, it is necessary to monitor flux levels. In this manner, even in the highly unlikely event of multiple operator errors, there is reasonable assurance that any approach to criticality would be detected in time to halt operations.

The minimum count rate requirement in the Technical Specifications accomplishes three safety functions: (1) it assures the presence of some neutrons in the core, (2) it provides assurance that the analog portion of the SRM channels is operable, and (3) it provides assurance that the SRM detectors are close enough to the array of fuel assemblies to monitor core flux levels.

Unloading and reloading of the entire core leads to some difficulty with this minimum count rate requirement. When only a small number of assemblies are present within the core, the SRM count rate will drop below the minimum due to the small number of neutrons being produced, and due to attenuation of these neutrons in the water (and control blades) separating the fuel from the SRM detectors. Past practice has been to connect temporary "dunking" chambers to the SRM channels in place of the normal detectors, and to locate these detectors near the fuel.

Besides being operationally inconvenient, dunking chambers suffer from signal variations due to their lack of fixed geometry. Moreover, the use of dunking chambers increases the risk of loose objects being dropped into the vessel.

Several licensees have opted for a spiral unloading/reloading pattern. Philadelphia Electric Company has instead proposed a different pattern in which the fuel assemblies immediately adjacent to the SRMs are removed last.

Although there is some freedom in the details of the unloading/reloading pattern, the specific procedure intended is as follows: One quadrant will be unloaded starting at the core periphery and working toward the center. The remaining three quadrants are then, in turn, unloaded in a quadrant spiral centered on the quadrant's SRM.

4.2 Evaluation

4.2.1 Minimum Flux in the Core

A multiplying medium with no neutrons present forms the basis for an accident scenario in which reactivity is gradually but inadvertently added until the medium is highly supercritical. No neutron flux will be evident since there are no neutrons present to be multiplied. The introduction of some neutrons at this point would cause the core to undergo a sudden power burst, rather than a gradual startup, with no warning from the nuclear instrumentation.

This scenario is of great concern when loading fresh fuel, but is of lesser concern for exposed fuel. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photo-fission, and photodisintegration of deuterium naturally present in the moderator. This neutron production in exposed fuel is normally great enough to meet the 3 cps minimum for a full core after a refueling outage with the lumped neutron source removed.

The proposed specification guarantees the presence of neutrons in the core because it requires the first assemblies, which will always be loaded next to the SRMs, to produce 3 cps on the SRM. If the first four assemblies fail to produce 3 cps, the operation must be stopped. (Four assemblies are not sufficient by themselves for criticality, even if all control blades were removed.) Thus, we find this amendment acceptable from the point of view of minimum flux.

4.2.2 SRM Operability

The proposed Technical Specifications relax the 3 cps minimum count rate requirement only when the last four assemblies are removed from around the detectors. Thus, a continuous check of neutron response is available.

In addition, in this type of unloading sequence, the SRMs cannot be mechanically disturbed by fuel handling operations until the defueling operation is nearly complete. Thus, the normal channel checking frequency should be adequate.

4.2.3 Flux Attenuation

In the unloading/reloading sequences as described by the licensee, the SRM detectors are always located within the array of fuel. For such sequences, there is no flux attenuation problem, since the detectors are never displaced from the array of remaining fuel.

However, the licensee's proposed Technical Specifications did not adequately enforce this. We proposed revisions in the wording of Section 3.10.B of the Technical Specifications on "Core Monitoring" as described below. These revisions were discussed with and accepted by the licensee's staff. We proposed that:

(a) Specification 3.10.B.1, p. 227, be modified to read, "Except as specified in 3.10.B.2, 3.10.B.3, 3.10.B.4, and 3.10.B.5, during core alterations two SRMs shall be operable..."

(b) A new Specification 3.10.B.5 be added on p. 228:

"5. During the unloading and reloading of fuel, intermediate arrays of fuel shall always contain at least one SRM."

4.2.4 Subcriticality of the Intermediate Arrays

The major characteristics of the proposed unloading sequences is that they produce water-filled spaces within the fueled arrays. In order to avoid problems with shutdown margin caused by such "flux traps," it is mandatory to keep the control blades inserted into all water-filled control cells located in the interior of the array of remaining fuel. The Peach Bottom site is equipped with sufficient blade guides to do this.

During loading of the core, Specification 3.10.A.2 requires all blades to be in place before any fuel is loaded. During unloading, Specification 3.10.B.2 will allow blades "outside the periphery of the existing fuel matrix" to be withdrawn. "Outside the periphery" in this context means that the control cell in question is not surrounded by fueled cells, i.e., the cell can "see" the core shroud and is not recessed into the fueled area. Such a cell cannot cause a shutdown margin problem.

Therefore, we find this amendment to be acceptable from the point of view of subcriticality of the intermediate array. However, it should be noted that the licensee's proposed sequences depend upon the availability of a full complement of blade guides. Thus, this approach is not generically applicable to other BWRs unless the licensee has a full complement of blade guides.

5.0 Summary

We have reviewed and evaluated the changes to the Technical Specifications proposed by the licensee and the related safety issues. With the minor revision in wording proposed by the staff in Section 4.2.3 of this safety evaluation to be more explicit in what is permitted, we concluded that the proposed changes to the Technical Specifications are acceptable.

6.0 Environmental Consideration

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

7.0 Conclusion

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

Dated: March 26, 1980

References

1. Letter, E. J. Bradley, (PECo) to H. R. Denton (NRC), dated January 14, 1980.

2. Letter, T. A. Ippolito (NRC) to G. C. Andognini (Boston Edison), dated January 8, 1980.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-277 AND 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 65 and 64 to Facility Operating License Nos. DPR-44 and DPR-56, issued to the Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station, Units Nos. 2 and 3 (the facility) located in York County, Pennsylvania. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications relating to core monitoring, core flooding and cooling system operability requirements. The effect of the revised Technical Specifications would be (1) to allow dewatering of the suppression chambers (torus) during refueling outages, provided adequate core flooding inventory is available and no operations are being performed which have a potential for draining the reactor vessel, (2) to allow unloading and reloading the core under special conditions without having 3 counts per second (cps) on the source range monitors (SRMs) and (3) to allow the reactor protection system instrumentation to be inoperable when the reactor is completely defueled.

The application for the amendments 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendment dated January 14, 1980, (2) Amendment Nos. 65 and 64 to License Nos. DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director: Division of Operating Reactors.

Dated at Bethesda, Maryland this 26th day of March 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

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