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. 129
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 February 12, 1987 as supplemented on October 20, 1987, 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 or 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. 129 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RMartin:mr	SETurk	WButler
2/24/88	01/05/88	2/25/88	3/3/87

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 129 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines. Asterisk page is provided for document completeness.

Remove	Insert
37	37
38	38
47	47*
48	48
61	61
62	62
63	63
90	90

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
1	1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		X	X	X	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(0.58W+62-0.58ΔW) W			X	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	X	X	X	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	X	X		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	X(9)	X	X	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	X	X	X	4 Instrument Channels	A

Table 3.1.1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
12	2	High Water Level in Scram Discharge Instrument Volume	≤50 Gallons	X(2)	X	X	4 Instrument Channels	A
13	2	Turbine Condenser Low Vacuum	≥23 in. Hg. Vacuum			X	4 Instrument Channels	A or C
14	2	Main Steam Line High Radiation	≤15 X Normal Full Power Background	X	X	X	4 Instrument Channels	A
15	4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure			X(6)	8 Instrument Channels	A
16	2	Turbine Control Valve Fast Closure	500<P<850 psig Control Oil Pressure Between Fast Closure Solenoid and Disc Dump Valve			X(4)	4 Instrument Channels	A or D
17	4	Turbine Stop Valve Closure	≤10% Valve Closure			X(4)	8 Instrument Channels	A or D

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve t.e 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 prevent inadvertant criticality.

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

3.0 BASES (Cont'd)

the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation sensing drywell pressure is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds fifteen times normal background. The purpose of this scram is to limit fission product release so that 10 CFR Part 100 guidelines are not exceeded. Discharge of excessive amounts of radioactivity to the site environs is prevented by the off-gas treatment system, which provides sufficient delay time to reduce fission product release rates to well below 10 CFR 20 guidelines.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns

TABLE 3.2.A   PBAPS Unit 2

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
1	2 (6)	Reactor Low Water Level	≥ 0" Indicated Level (3)	4 Inst. Channels	A
2	1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
3	2	Reactor Low-Low-Low Water Level	at or above -160" indicated level (4)	4 Inst. Channels	A
4	2 (6)	High Drywell Pressure	≤ 2 psig	4 Inst. Channels	A
5	2	High Radiation Main Steam Line Tunnel	≤15 X Normal Rated Full (8) Power Background	4 Inst. Channels	B
6	2	Low Pressure Main Steam Line	≥ 850 psig (7)	4 Inst. Channels	B
7	2 (5)	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
8	2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤ 200 deg. F (9)	4 Inst. Channels	B

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
9	2	Main Steam Line Leak Detection High Temperature	≤ 200 Deg. F	4 Inst. Channels	B
10	1	Reactor Cleanup System High Flow	≤ 300% of Rated Flow	2 Inst. Channels	C
11	1	Reactor Cleanup System High Temperature	≤ 200 Deg. F.	1 Inst. Channels	E
12	2	Reactor Pressure (Feedwater Flush System Interlock)	≤ 600 psig	4 Inst. Channels	F

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken:

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup Syrtem.

   D. Isolate Shutdown Cooling.

   E. Isolate Reactor Water Cleanup Filter Demineralizers unless the following provision is satisifed. The RWCU Filter Demineralizer may be used (the isolation overridden) to route the reactor water to the main condenser or waste surge tank, with the high temperature trip inoperable for up to 48 hours, provided the water inlet temperature is monitored once per hour and confirmed to be below 180 degrees F.

   F. Isolate Feedwater Flush System

3. Instrument setpoint corresponds to 538 inches above vessel zero.

4. Instrument setpoint corresponds to 378 inches above vessel zero.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

8. An alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level.

3.2 BASES (Cont'd)

the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Spec. 3.7 for Isolation Valve Closure Group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures peak at approximately 1000 degrees F and release of radioactivity to the environs is below CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel exhaust duct and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200 degrees F for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 15 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR.

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

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
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 February 12, 1987 as supplemented on October 20, 1987, 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 or 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. 132 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 3, 1988

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RMartin:mr	SETurk	WButler
2/24/87 88	01/05/88	2/25/88	3/3/87

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 132 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines. Asterisk page is provided for document completeness.

Remove	Insert
37	37
38	38
40	40
47	47*
48	48
61	61
62	62
63	63
90	90

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
1	1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		X	X	X	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(.66W+54-0.66 ΔW) FRP/MFLPD (12) (13)			X	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	X	X	X	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	X	X		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	X(9)	X	X	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	X	X	X	4 Instrument Channels	A

Table 3.1.1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
12	2	High Water Level in Scram Discharge Instrument Volume	≤50 Gallons	X(2)	X	X	4 Instrument Channels	A
13	2	Turbine Condenser Low Vacuum	≥23 in. Hg. Vacuum			X	4 Instrument Channels	A or C
14	2	Main Steam Line High Radiation	≤15 X Normal Full Power Background	X	X	X	4 Instrument Channels	A
15	4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure			X(8)	8 Instrument Channels	A
16	2	Turbine Control Valve Fast Closure	500<P<850 psig Control Oil Pressure Between Fast Closure Solenoid and Disc Dump Valve			X(4)	4 Instrument Channels	A or D
17	4	Turbine Stop Valve Closure	≤10% Valve Closure			X(4)	8 Instrument Channels	A or D

NOTES FOR TABLE 3.1.1 (Cont'd)

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

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

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

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

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

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

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

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

13. See Section 2.1.A.1.

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 prevent inadvertant criticality.

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

3.0 BASES (Cont'd)

the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation sensing drywell pressure is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds fifteen times normal background. The purpose of this scram is to limit fission product release so that 10 CFR Part 100 guidelines are not exceeded. Discharge of excessive amounts of radioactivity to the site environs is prevented by the off-gas treatment system, which provides sufficient delay time to reduce fission product release rates to well below 10 CFR 20 guidelines.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns

TABLE 3.2.A PBAPS Unit 3

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
1	2 (6)	Reactor Low Water Level	≥ 0" Indicated Level (3)	4 Inst. Channels	A
2	1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
3	2	Reactor Low-Low-Low Water Level	at or above -160" indicated level (4)	4 Inst. Channels	A
4	2 (6)	High Drywell Pressure	≤ 2 psig	4 Inst. Channels	A
5	2	High Radiation Main Steam Line Tunnel	≤15 X Normal Rated Full (8) Power Background	4 Inst. Channels	B
6	2	Low Pressure Main Steam Line	≥ 850 psig (7)	4 Inst. Channels	B
7	2 (5)	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
8	2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤ 200 deg. F (9)	4 Inst. Channels	B

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
9	2	Main Steam Line Leak Detection High Temperature	≤ 200 Deg. F	4 Inst. Channels	B
10	1	Reactor Cleanup System High Flow	< 300% of Rated Flow	2 Inst. Channels	C
11	1	Reactor Cleanup System High Temperature	≤ 200 Deg. F.	1 Inst. Channels	E
12	2	Reactor Pressure (Feedwater Flush System Interlock)	≤ 600 psig	4 Inst. Channels	F

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken:

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

   E. Isolate Reactor Water Cleanup Filter Demineralizers unless the following provision is satisifed. The RWCU Filter Demineralizer may be used (the isolation overridden) to route the reactor water to the main condenser or waste surge tank, with the high temperature trip inoperable for up to 48 hours, provided the water inlet temperature is monitored once per hour and confirmed to be below 180 degrees F.

   F. Isolate Feedwater Flush System

3. Instrument setpoint corresponds to 538 inches above vessel zero.

4. Instrument setpoint corresponds to 378 inches above vessel zero.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

8. An alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level.

3.2 BASES (Cont'd)

the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Spec. 3.7 for Isolation Valve Closure Group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures peak at approximately 1000 degrees F and release of radioactivity to the environs is below CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel exhaust duct and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200 degrees F for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 15 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR.