ATTACHMENT A

CHAPTER 2.0,
SECTIONS 3.1, 3.2, 3.4, 3.5, 3.9, AND 3.10,
CHAPTER 5.0, AND
APPENDIX B

DIFFERENCES BETWEEN PBAPS UNIT 2 AND UNIT 3 ITS AND BASES
(AS APPLICABLE)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormal operational transients.

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for General Electric (GE) Company, ~~Asea Brown Boveri (ABB) Atom, and Siemens Power Corporation (SPC)~~ fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

for Unit 2 only

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations 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.

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 SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during abnormal operational transients, at least 99.9% of the fuel rods in the core do not experience transition boiling.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10^3 lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of 25% RTP prevents any bundle from exceeding critical power and is a conservative limit when reactor pressure < 785 psig.

for Unit 3 only

Lead Fuel Assemblies (LFAs)

In addition to being applicable to GE fuel, the Fuel Cladding Integrity Safety Limit is also applicable to the Qualification Fuel Bundles (QFBs) manufactured by GE, ABB Atom, and SPC as justified in References 1, 2, and 3 respectively.

for Unit 2 only

for Unit 2 only

for Unit 2 only

for Unit 2 only

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore,

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 4. Reference 4 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis. In addition to being applicable to GE fuel, the MCPR Safety Limit is also applicable to the QFBs manufactured by GE, ABB Atom, and SPC as justified in References 1, 2, and 3 respectively.

for Unit 3 2

for Unit 3 LFAs

for Unit 2 only

for Unit 2 only

for Unit 2 only

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be adequately cooled as long as water level is above ⅔ of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

(continued)

BASES (continued)

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

3 for Unit 3

2.2.2

4 for Unit 3.

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.4

5 for Unit 3

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 7). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

for Unit 3 only

EMF-93-115(P), July 1993.

1. GE Nuclear Energy 23A7188, Revision 1, September 1992.
2. ABB Atom Report BR 90-004, October 1990.
3. ANF-90-133 (P), Revision 2, August 1992.
4. NEDE-24011-P-A-10, February 1991.
5. 10 CFR 50.72.
6. 10 CFR 100.
7. 10 CFR 50.73.

for Unit 2 only

2.
3.
4.
5.

for Unit 3

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	AND		
	D.3	Initiate action to restore one standby gas treatment (SGT) subsystem for Unit 2 3 for Unit 3 to OPERABLE status.	1 hour
	AND		
	D.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.3 Initiate action to restore secondary containment to OPERABLE status. AND E.4 Initiate action to restore one SGT subsystem for Unit 2 3 for Unit 3 to OPERABLE status. AND E.5 Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour 1 hour 1 hour

BASES

ACTIONS

A.1 (continued)

acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval.

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem for Unit 2 is OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability), in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. This may be performed as

3
for Unit 3

(continued)

BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM, e.g., insertion of fuel in the core or the withdrawal of control rods. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at 3
least one SGT subsystem for Unit 2 is OPERABLE; and
secondary containment isolation capability (i.e., at least for Unit 3
one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability), in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactive releases. This may be performed as an administrative check, by examining logs or other

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 10.

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 11. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum of 0.90 (Ref. 11). This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

for Unit 3 only

Lead Fuel Assemblies (LFAs)

for Unit 2 only

In addition to being applicable to the General Electric (GE) fuel, the APLHGR limits are also applicable to the Qualification Fuel Bundles (QFBs) manufactured by GE, Asea (for Unit 2 only) Brown Boveri (ABB) Atom, and Siemens Power Corporation (SPC) as justified in References 12, 13, and 14, respectively. (for Unit 2 only)

The APLHGR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For two

(continued)

BASES

ACTIONS

B.1 (continued)

allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDO-24011-P-A-10, "General Electric Standard Application for Reactor Fuel," February 1991.
2. UFSAR, Chapter 3.
3. UFSAR, Chapter 6.
4. UFSAR, Chapter 14.
5. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single Loop Operation," May 1980.
6. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 1, February 1993.
7. NEDC-32183P, "Power Rerate Safety Analysis Report for Peach Bottom 2 & 3," May 1993.
8. NEDC-32428P, "Peach Bottom Atomic Power Station Unit 2 (for Unit 3: 3) Cycle 11 (10 for Unit 3) ARTS Thermal Limits Analyses," December 1994.
9. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

A

(continued)

BASES

REFERENCES (continued)

10. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

11. NEDC-32163P, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993.

12. GE Nuclear Energy 23A7188, Revision 1, September 1992.

13. ABB Atom Report BR90-004, October 1990.

14. ANF-90-133(P), Revision 2, August 1992.

for Unit 2 only

EMF-93-115(P), July 1993.

for Unit 3 only

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, 8, and 9). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 10) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. 11). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

for Unit 3 only

Lead Fuel Assemblies (LFAs)

In addition, unique MCPR limits have been established for the Qualification Fuel Bundles (QFBs) manufactured by General Electric (GE), Asea Brown Boveri (ABB) Atom, and Siemens Power Corporation (SPC) as discussed in References 12, 13, and 14, respectively. for Unit 2

for Unit 2 only

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and

(continued)

BASES

REFERENCES (continued)

7. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 1, February 1993.

8. NEDC-32183P, "Power Rerate Safety Analysis Report for Peach Bottom 2 & 3," May 1993.

for Unit 3 3

9. NEDC-32428P, "Peach Bottom Atomic Power Station Unit 2 Cycle 11 ARTS Thermal Limits Analyses," December 1994.

10 for Unit 3

10. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

11. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

12. GE Nuclear Energy 23A7188, Revision 1, September 1992.

13. ABB Atom Report BR 90-004, October 1990.

14. ANF-90-133(P), Revision 2, August 1992.

EMF-93-115 (P), July 1993

for Unit 2 only

for Unit 3 only

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including abnormal operational transients. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

for Unit 3

APPLICABLE SAFETY ANALYSES

(and) for Unit 2 only

The analytical methods and assumptions used in evaluating the fuel system ~~design are presented~~ in References 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, and 11. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 12).

10 for Unit 3

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. NEDO-24011-P-A-10, "General Electric Standard Application for Reactor Fuel," February 1991.

2. UFSAR, Chapter 3.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 14.

5. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single-Loop Operation," May 1980.

6. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvements Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 1, February 1993.

for Unit 3 only

EMF-93-115(P), July 1993.

7. NEDC-32183P, "Power Rerate Safety Analysis Report for Peach Bottom 2 & 3," May 1993.

8. NEDC-32163P, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993.

9. GE Nuclear Energy 23A7188, Revision 1, September 1992.

10. ABB Atom Report BR 90-004, October 1990.

11. ANF-90-133(P), Revision 2, August 1992.

for Unit 2 only

12. 10 NUREG-0800, Section 4.2, Subsection II.A.2(g), Revision 2, July 1981.

for Unit 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

This SR ensures the reactor THERMAL POWER and core flow are within appropriate parameter limits to prevent uncontrolled power oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal hydraulic instability. Figure 3.4.1-1 is based on guidance provided in Reference 6, which is used to respond to operation in these conditions. The 24 hour Frequency is based on operating experience and the operators' inherent knowledge of reactor status, including significant changes in THERMAL POWER and core flow.

REFERENCES

1. UFSAR, Section 14.6.3.

2. NEDC-32163P, "PBAPS Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993.

3. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Unit 2 and 3," Revision 1, February 1993.

4. NEDC-32428P, "Peach Bottom Atomic Power Station Unit 2 3 for Unit 3 Cycle 11 10 for Unit 3 ARTS Thermal Limits Analyses," December 1994.

5. NEDO-24229-1, "PBAPS Units 2 and 3 Single-Loop Operation," May 1980.

6. GE Service Information Letter No. 380, "BWR Core Thermal Hydraulic Stability," Revision 1, February 10, 1984.

7. NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors (BWRs)," Supplement 1, December 30, 1988.

8. NRC Generic Letter 86-02, "Technical Resolution of Generic Issue B-19 Thermal Hydraulic Stability," January 22, 1986.

BASES

3 for Unit 3

LCO (continued)

both subsystems. In MODE 4, the RHR cross tie valve (MO-2-10-020) may be opened (per LCO 3.5.2) to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. In addition, the HPSW cross-tie valve may be opened to allow an HPSW pump in one loop to provide cooling to a heat exchanger in the opposite loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 permits both required RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period. Note 2 allows one required RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODE 4, the RHR Shutdown Cooling System must be OPERABLE and shall be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F. Otherwise, a recirculation pump is required to be in operation.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR shutdown cooling isolation pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures above the RHR shutdown cooling isolation pressure is typically accomplished by condensing the steam in the main condenser.

(continued)

BASES

REFERENCES (continued)

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. SASR 88-24 (90-50 for Unit 3), Peach Bottom Atomic Power Station Unit 2 (3 for Unit 3) Vessel Surveillance Materials Testing and Fracture Toughness Analysis, Revision 1, December 1991 (June 1990 for Unit 3).

8. UFSAR, Section 14.5.6.2.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action C.2 and associated Completion Time not met.	D.1	Initiate action to restore secondary containment to OPERABLE status.	Immediately
	AND		
	D.2	Initiate action to restore one standby gas treatment subsystem for Unit 2 3 for Unit 3 to OPERABLE status.	Immediately
	AND		
	D.3	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify, for each required low pressure coolant injection (LPCI) subsystem, the suppression pool water level is ≥ 11.0 ft.	12 hours

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

the ECCS not being able to perform its intended function. The 4 hour Completion Time for restoring the required low pressure ECCS injection/spray subsystem to OPERABLE status is based on engineering judgment that considered the remaining available subsystem and the low probability of a vessel draindown event.

With the inoperable subsystem not restored to OPERABLE status in the required Completion Time, action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

C.1, C.2, D.1, D.2, and D.3

With both of the required ECCS injection/spray subsystems inoperable, all coolant inventory makeup capability may be unavailable. Therefore, actions must immediately be initiated to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. One ECCS injection/spray subsystem must also be restored to OPERABLE status within 4 hours.

If at least one low pressure ECCS injection/spray subsystem is not restored to OPERABLE status within the 4 hour Completion Time, additional actions are required to minimize any potential fission product release to the environment. This includes ensuring secondary containment is OPERABLE; 3 one standby gas treatment subsystem for Unit 2 is OPERABLE; for Unit 3 and secondary containment isolation capability (i.e., one isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability) in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. OPERABILITY may be verified by an administrative check, or by examining logs or other information, to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3	Initiate action to restore one standby (3) gas treatment subsystem for Unit (2) to OPERABLE status.	Immediately 6. Unit 3
	AND		
	B.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately
C. No RHR shutdown cooling subsystem in operation.	C.1	Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND		
	C.2	Monitor reactor coolant temperature.	Once per hour

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Initiate action to restore one standby gas treatment subsystem for Unit 3 2 to OPERABLE status.	Immediately for Unit 3
	AND		
	B.3	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately
C. No RHR shutdown cooling subsystem in operation.	C.1	Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND		
	C.2	Monitor reactor coolant temperature.	Once per hour

BASES

ACTIONS

A.1 (continued)

the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. For example, this may include the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed. The method used to remove the decay heat should be the most prudent choice based on unit conditions.

B.1, B.2, B.3, and B.4

If no RHR shutdown cooling subsystem is OPERABLE and an alternate method of decay heat removal is not available in accordance with Required Action A.1, actions shall be taken immediately to suspend operations involving an increase in reactor decay heat load by suspending loading of irradiated fuel assemblies into the RPV.

3 for Unit 3

Additional actions are required to minimize any potential fission product release to the environment. This includes ensuring secondary containment is OPERABLE; one standby gas treatment subsystem for Unit 2 is OPERABLE; and secondary containment isolation capability (i.e., one secondary containment isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability) in each associated penetration not isolated that is assumed to be isolated to mitigate radioactive releases. This may be performed as an administrative check, by examining logs or other information to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, a surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

(continued)

BASES

ACTIONS

A.1 (continued)

LCO. The 1 hour Ccmpletion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of this alternate method(s) must be reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. For example, this may include the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed. The method used to remove decay heat should be the most prudent choice based on unit conditions.

B.1, B.2, and B.3

3 for Unit 3

With the required decay heat removal subsystem(s) inoperable and the required alternate method(s) of decay heat removal not available in accordance with Required Action A.1, additional actions are required to minimize any potential fission product release to the environment. This includes ensuring secondary containment is OPERABLE; one standby gas treatment subsystem for Unit 2 is OPERABLE; and secondary containment isolation capability (i.e., one secondary containment isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability) in each associated penetration that is assumed to be isolated to mitigate radioactive releases. This may be performed as an administrative check, by examining logs or other information to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"; and

8. PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

90-50 for Unit 3

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed by the NRC in SASR 88-24, Peach Bottom Atomic Power Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis, Revision 1, December 1991, and approved by the NRC in [ ].

3 for Unit 3

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c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

(continued)

APPENDIX B

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

FOR

PEACH BOTTOM ATOMIC POWER STATION UNIT 2

3 for Unit 3