3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSIV(s) inoperable in MODE 1 or 2.	A.1 Restore MSIV(s) to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. -------------NOTE------------- Separate Condition entry is allowed for each MSIV. --------------------------------- One or more MSIV(s) inoperable in MODE 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	48 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-----NOTE----- Only required to be performed in MODES 1 and 2. ----- Verify isolation time of each MSIV is within the limits specified in the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.7.2.2	-----NOTE----- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. ----- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves

LCO 3.7.3 All MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MFIV in one or more flow paths inoperable	A.1 Close or isolate MFIV.	72 hours
	<u>AND</u>	
	A.2 Verify MFIV is closed or isolated.	Once per 7 day
B. One Main Feedwater Block Valve in one or more flow paths inoperable	B.1 Close or isolate Main Feedwater Block Valve.	72 hours
	<u>AND</u>	
	B.2 Verify Main Feedwater Block Valve is closed or isolated.	Once per 7 days
C. One Low Load Feedwater Control Valve in one or more flow paths inoperable.	C.1 Close or isolate Low Load Feedwater Control Valve.	72 hours
	<u>AND</u>	
	C.2 Verify Low Load Feedwater Control Valve is closed or isolated.	Once per 7 days

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One Startup Feedwater Control Valve in one or more flow paths inoperable.	D.1	Close or isolate Startup Feedwater Control Valve.	72 hours
	AND		
	D.2	Verify Startup Feedwater Control Valve is closed or isolated.	Once per 7 days
E. Two valves in the same flow path inoperable for one or more flow paths.	E.1	Isolate affected flow path.	8 hours
F. Required Action and associated Completion Time not met.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify the isolation time of each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve is within the limits provided in the Inservice Testing Program.	In accordance with the Inservice Testing Program

SURVEILLANCE		FREQUENCY
SR 3.7.3.2	------------------------------NOTES------------------------------ 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. --- Verify that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be $\leq$ 0.17 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant is $\leq$ 0.17 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

--NOTE--
Only one EFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven EFW pump inoperable. <u>OR</u> -------------NOTE------------- Only applicable if MODE 2 has not been entered following refueling. --------------------------------- Turbine driven EFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One EFW train inoperable for reasons other than Condition A in MODE 1, 2, or 3.	B.1 Restore EFW train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 18 hours
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 --------------NOTE-------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ----------------------------------- Initiate action to restore one EFW train to OPERABLE status.	 Immediately
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	-----------------------------NOTE----------------------------- Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching $\geq$ 750 psig in the steam generators. --- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	 In accordance with the Inservice Testing Program

SURVEILLANCE		FREQUENCY
SR 3.7.5.3	------------------------------NOTE------------------------------ Not required to be met in MODE 4 when steam generator is relied upon for heat removal. --- Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4	------------------------------NOTE------------------------------ Not required to be met in MODE 4 when steam generator is relied upon for heat removal. --- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying manual valve alignment from the "Q" condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days
SR 3.7.5.6	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	18 months

3.7 PLANT SYSTEMS

3.7.6 Q Condensate Storage Tank (QCST)

LCO 3.7.6 The QCST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The QCST inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore QCST to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4 without reliance on steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify QCST volume is $\geq$ 267,000 gallons when required for both units and $\geq$ 107,000 gallons when only required for Unit 1.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Service Water System (SWS)

LCO 3.7.7 Two SWS loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SWS loop inoperable.	A.1	-------------NOTES------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by SWS. 2. Enter Applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal made inoperable by SWS. -------------------------------- Restore SWS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----------------------------NOTE----------------------------- Isolation of SWS flow to individual components does not render the SWS inoperable. --- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.7.3	Verify each required SWS pump starts automatically on an actual or simulated signal.	18 months

3.7 PLANT SYSTEMS

3.7.8 Emergency Cooling Pond (ECP)

LCO 3.7.8 The ECP shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ECP inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify water level of ECP is $\geq$ 5 ft.	24 hours
SR 3.7.8.2	-----NOTE----- Only required to be performed from June 1 through September 30. ---------- Verify average water temperature is $\leq$ 100°F.	24 hours
SR 3.7.8.3	Verify contained water volume of ECP $\geq$ 70 acre-ft at water level of 5 ft.	12 months

SURVEILLANCE		FREQUENCY
SR 3.7.8.4	Verify earth portions of stone covered embankments and spillway of ECP: a. Have not been eroded or undercut by wave action, and b. Do not show apparent changes in visual appearance or other abnormal degradation from as-built condition.	12 months

3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.9 Two CREVS trains shall be OPERABLE.

-----NOTES-----

1. The control room boundary may be opened intermittently under administrative controls.
2. One CREVS train shall be capable of automatic actuation.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVS train inoperable.	A.1 Restore CREVS train to OPERABLE status.	7 days
B. Two CREVS trains inoperable due to inoperable control room boundary in MODES 1, 2, 3, and 4.	B.1 Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 Place OPERABLE CREVS train in emergency recirculation mode. <u>OR</u> D.2. Suspend movement of irradiated fuel assemblies.	Immediately Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two CREVS trains inoperable during movement of irradiated fuel assemblies.	E.1	Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREVS trains inoperable during MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each CREVS train for $\geq$ 15 minutes.	31 days
SR 3.7.9.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify the CREVS automatically isolates the Control Room and switches into a recirculation mode of operation on an actual or simulated actuation signal.	18 months
SR 3.7.9.4	Verify the system makeup flow rate is $\geq$ 300 and $\leq$ 366 cfm when supplying the control room with outside air.	18 months

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Air Conditioning System (CREACS)

LCO 3.7.10 Two CREACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREACS train inoperable.	A.1 Restore CREACS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	C.1 Place OPERABLE CREACS train in operation. OR C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
D. Two CREACS trains inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREACS trains inoperable during MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify each CREACS train starts, operates for at least 1 hour, and maintains control room air temperature $\leq$ 84°F D. B.	31 days
SR 3.7.10.2	Verify system flow rate of 9900 cfm $\pm$ 10%.	18 months

3.7 PLANT SYSTEMS

3.7.11 Penetration Room Ventilation System (PRVS)

LCO 3.7.11 Two PRVS trains shall be OPERABLE.

-----------------------------------NOTE-----------------------------------
The penetration room negative pressure boundary may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PRVS train inoperable.	A.1 Restore PRVS train to OPERABLE status.	7 days
B. Two PRVS trains inoperable due to inoperable penetration room negative pressure boundary.	B.1 Restore penetration room negative pressure boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met. OR Both PRVS trains inoperable for reasons other than Condition B.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each PRVS train for $\geq$ 15 minutes.	31 days
SR 3.7.11.2	Perform required PRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

SURVEILLANCE		FREQUENCY
SR 3.7.11.3	Verify each PRVS train actuates on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.12 Fuel Handling Area Ventilation System (FHAVS)

LCO 3.7.12 The FHAVS shall be OPERABLE and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling area.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. FHAVS inoperable or not in operation.	A.1	Suspend movement of irradiated fuel assemblies in the fuel handling area.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify FHAVS in operation.	12 hours
SR 3.7.12.2	Perform required FHAVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFPT

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Water Level

LCO 3.7.13 The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 -------------NOTE------------- LCO 3.0.3 is not applicable. ----------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Boron Concentration

LCO 3.7.14 The spent fuel pool boron concentration shall be ≥ 1600 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	-------------------NOTE------------------- LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the spent fuel pool boron concentration is ≥ 1600 ppm.	7 days

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 2 shall be within the acceptable range of Figure 3.7.15-1 or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

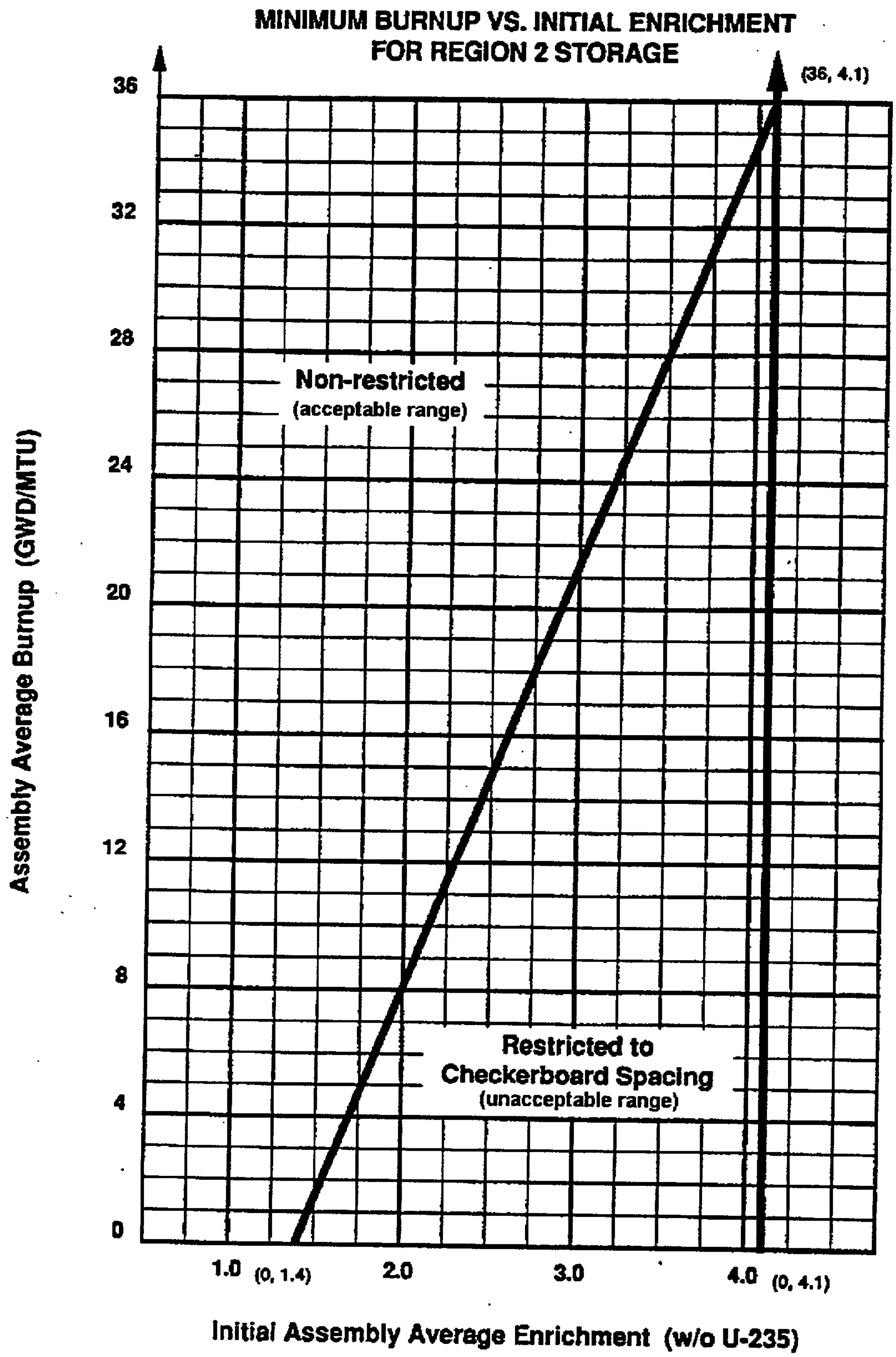
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ----------------------------------- Initiate action to move the noncomplying fuel assembly from Region 2.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 or Specification 4.3.1.1.	Prior to storing the fuel assembly in Region 2

Figure 3.7.15-1
Burnup versus Enrichment Curve for
Spent Fuel Storage Racks



3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour AND Once per 12 hours thereafter
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		

CONDITION	REQUIRED ACTION		COMPLETION TIME
A (continued)	A.3	-------------NOTE------------- Startup Transformer No. 2 may be removed from service for up to 30 days for preplanned preventative maintenance. This 30 day Completion Time may be applied not more than once in any 10 year period. The provisions of LCO 3.0.4 are not applicable to Startup Transformer No. 2 during this 30 day preventative maintenance period. ----------------------------------- Restore required offsite circuit to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 12 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	OR		

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	AND		
	B.4	Restore DG to OPERABLE status.	7 days AND 10 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours
D. One required offsite circuit inoperable. AND One DG inoperable.	--------------------NOTE-------------------- Enter applicable Conditions and Required Actions of LCO 3.8.6, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any train. ---		
	D.1	Restore required offsite circuit to OPERABLE status.	12 hours
	OR		
	D.2	Restore DG to OPERABLE status.	12 hours
E. Two DGs inoperable.	E.1	Restore one DG to OPERABLE status.	2 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	-----------------------------NOTE----------------------------- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. --- Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves "ready-to-load" conditions.	31 days
SR 3.8.1.3	-----------------------------NOTES----------------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and follow, without shutdown, a successful performance of SR 3.8.1.2. --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.	31 days

SURVEILLANCE		FREQUENCY
SR 3.8.1.4	Verify each day tank contains ≥ 160 gallons of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage tanks to the day tank.	31 days
SR 3.8.1.7	------------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. --- Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.	18 months
SR 3.8.1.8	------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. --- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes.	18 months

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. -- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected emergency loads through load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown"; and

b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. -----	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	A.2.3	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND		
	A.2.4	Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	B.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	B.3	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND		
	B.4	Initiate action to restore required DG to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	-----------------------------NOTES----------------------------- 1. SR 3.8.1.3 is not required to be performed. 2. The 15 second acceptance criteria of SR 3.8.1.2 is not applicable. --- For AC Sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources – Operating," except SR 3.8.1.4, SR 3.8.1.7, SR 3.8.1.8, and SR 3.8.1.9, are applicable.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil and Starting Air

LCO 3.8.3 The stored diesel fuel oil and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DG fuel oil storage tank(s) with fuel volume < 20,000 gallons and > 17,140 gallons.	A.1 Restore fuel oil volume to within limits.	48 hours
B. One or more DGs with stored fuel oil total particulates not within limit.	B.1 Restore fuel oil total particulates to within limits.	7 days
C. One or more DGs with new fuel oil properties not within limits.	C.1 Restore stored fuel oil properties to within limits.	30 days
D. One or more DGs with required starting air receiver pressure < 175 psig and ≥ 158 psig.	D.1 Restore required starting air receiver pressure to within limits.	48 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met. <u>OR</u> One or more DGs with diesel fuel oil or required starting air subsystem not within limits for reasons other than Condition A, B, C, or D.	E.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains $\geq$ 20,000 gallons of fuel.	31 days
SR 3.8.3.2	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.3	Verify each DG required air start receiver pressure is $\geq$ 175 psig.	31 days
SR 3.8.3.4	Check for and remove accumulated water from each fuel oil storage tank.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Both DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	8 hours
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 124.7 V on float charge.	7 days
SR 3.8.4.2	Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	18 months

SURVEILLANCE		FREQUENCY
SR 3.8.4.3	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 The DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.1.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.1.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	.<u>AND</u>	
	A.1.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately
	<u>AND</u>	

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1.5 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by Condition A.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	------------------------------NOTE------------------------------ The following SRs are not required to be performed: SR 3.8.4.2 and SR 3.8.4.3. -- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, and SR 3.8.4.3.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters shall be within limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category A or B limits.	A.1	Verify pilot cell electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	<u>AND</u>		
	A.2	Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
	A.3	Restore battery cell parameters to Table 3.8.6-1 Category A and B limits.	31 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with pilot cell or average electrolyte temperature of the representative cells < 60°F. <u>OR</u> One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days
SR 3.8.6.2	Verify electrolyte temperature of the pilot cell is ≥ 60°F.	31 days

SURVEILLANCE		FREQUENCY
SR 3.8.6.3	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days <u>AND</u> Once within 24 hours after a battery discharge < 110 V <u>AND</u> Once within 24 hours after a battery overcharge > 145 V
SR 3.8.6.4	Verify average electrolyte temperature of representative cells is ≥ 60°F.	92 days

Table 3.8.6-1
Battery Cell Surveillance Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[(b)(c)]	≥ 1.195	≥ 1.190 AND Average of all connected cells > 1.195	Not more than 0.020 below average connected cells AND Average of all connected cells ≥ 1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 The following inverters shall be OPERABLE.

a. Two Red Train inverters (Y11 and Y13, Y11 and Y15, or Y13 and Y15),

b. Two Green Train inverters (Y22 and Y24, Y22 and Y25, or Y24 and Y25), and

c. Inverter Y28

---------------------------------------NOTE---------------------------------------
One of the four inverters requried by LCO 3.8.7.a and LCO 3.8.7.b may be disconnected from its associated DC bus for ≤ 2 hours to perform load transfer to or from the swing inverter, provided:

a. The associated 120 VAC bus is energized from its alternate AC source; and

b. The other three 120 VAC buses are energized from their associated OPERABLE inverters.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	A.1	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any of the 120 VAC buses RS1, RS2, RS3, or RS4 de-energized. ------------------------------------- Restore inverter to OPERABLE status.	 24 hours <u>AND</u> 96 hours from discovery of failure to meet LCO

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Inverter Y28 inoperable.	B.1	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with 120 VAC bus C540 de-energized. -------------------------------- Restore inverter to OPERABLE status.	72 hours AND 96 hours from discovery of failure to meet LCO
C. Inverter Y28 inoperable. AND One of the two Red Train inverters required by LCO 3.8.7.a inoperable.	C.1	Restore one inverter to OPERABLE status.	2 hours
D. Required Action and associated Completion Time not met. OR Two or more of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	D.1 AND D.2	Be in MODE 3. Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to associated 120 VAC buses RS1, RS2, RS3, RS4, and C540.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 Inverters shall be OPERABLE to support the onsite Class 1E AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately
	<u>AND</u>	

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.5	Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by AC vital bus inverter(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage and alignments to required 120 VAC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Two AC, DC, and 120 VAC electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystem(s) inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
B. One or more 120 VAC electrical power distribution subsystem(s) (RS1, RS2, RS3, RS4) inoperable.	B.1 Restore 120 VAC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
C. 120 VAC electrical power distribution subsystem C540 inoperable.	C.1 Enter applicable Conditions and Required Actions of LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation," LCO 3.3.15, "Post Accident Monitoring (PAM) Instrumentation," and LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage."	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more DC electrical power distribution subsystem(s) inoperable.	D.1	Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	E.2	Be in MODE 5.	36 hours
F. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE by the following specifications:

LCO 3.3.9, "Source Range Neutron Flux,"
LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System,"
LCO 3.7.9, "Control Room Emergency Ventilation System (CREVS),"
LCO 3.7.10, "Control Room Emergency Air Conditioning System (CREACS),"
LCO 3.7.12, "Fuel Handling Area Ventilation System (FHAVS),"
LCO 3.9.2, "Nuclear Instrumentation," for one monitor,
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or 120 VAC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.4 Initiate actions to restore required AC, DC, and 120 VAC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.5 Declare associated required decay heat removal subsystem(s) inoperable.	Immediately
	AND	
	A.2.6 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by Electrical Power Distribution System.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	7 days

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System and the refueling canal shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

------------------------------------NOTE------------------------------------
Only applicable to the refueling canal when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2

a. One source range neutron flux monitor shall be OPERABLE, and

b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable during CORE ALTERATIONS.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. No OPERABLE source range neutron flux monitor.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	<u>AND</u>	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.3 Reactor Building Penetrations

LCO 3.9.3 The reactor building penetrations shall be in the following status:

a. The equipment hatch is capable of being closed;

b. One door in each air lock is capable of being closed; and

c. Each penetration providing direct access from the reactor building atmosphere to the outside atmosphere either:

   1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. capable of being closed by an OPERABLE reactor building isolation valve, except reactor building purge isolation valves, or

   3. capable of being closed by an OPERABLE reactor building purge isolation valve with the purge exhaust radiation monitoring channel OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more reactor building penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required reactor building penetration is in the required status.	7 days
SR 3.9.3.2	-----------------------------NOTE----------------------------- Not required to be met for reactor building isolation valves and reactor building purge isolation valves in penetrations closed to comply with LCO c.1. -- Verify each required reactor building isolation valve and each reactor building purge isolation valve actuates to the isolation position.	18 months
SR 3.9.3.3	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor.	18 months

3.9 REFUELING OPERATIONS

3.9.4 Decay Heat Removal (DHR) and Coolant Circulation - High Water Level

LCO 3.9.4 One DHR loop shall be OPERABLE and in operation.

-----NOTE-----

The required DHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System, coolant with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of the irradiated fuel seated in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DHR loop requirements not met.	A.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>		
	A.3	Initiate action to satisfy DHR loop requirements.	Immediately
	<u>AND</u>		
	A.4	Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one DHR loop is in operation.	12 hours

3.9 REFUELING OPERATIONS

3.9.5 Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level

LCO 3.9.5 Two DHR loops shall be OPERABLE, and one DHR loop shall be in operation.

---NOTE---

1. All DHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10 degrees F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration; and

   c. No draining operations to further reduce RCS water volume are permitted.

2. One required DHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other DHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 6 with the water level < 23 feet above the top of the irradiated fuel seated in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Less than required number of DHR loops OPERABLE.	A.1	Initiate action to restore DHR loop to OPERABLE status.	Immediately
	<u>OR</u>		
	A.2	Initiate action to establish ≥ 23 feet of water above the top of the irradiated fuel seated in the reactor pressure vessel.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No DHR loop OPERABLE or in operation.	B.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND		
	B.2	Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately
	AND		
	B.3	Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one DHR loop is in operation.	12 hours
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling canal water level is ≥ 23 feet above the top of irradiated fuel assemblies seated within the reactor pressure vessel.	24 hours

4.0 DESIGN FEATURES

4.1 Site Location

The site for Arkansas Nuclear One is located in Pope County, Arkansas on the north bank of the Dardanelle Reservoir (Arkansas River), approximately 6 miles west-northwest of Russellville, AR. The exclusion area boundary shall have a radius of 0.65 statute miles from the Unit 1 reactor building.

4.0 DESIGN FEATURES

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Assemblies

The reactor core shall contain 60 safety and regulating CONTROL ROD assemblies and 8 APSR assemblies. The CONTROL ROD assembly control material shall be a silver-indium-cadmium alloy and the APSR assembly control material shall be an Inconel alloy, as approved by the NRC.

DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.6.2.4.3 of the SAR;

c. A nominal 10.65 inch center to center distance between fuel assemblies placed in the storage racks;

d. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure 3.7.15-1 allowed unrestricted storage in either fuel storage rack Region 1 or Region 2; and

e. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure 3.7.15-1 stored in Region 1, or in checkerboard configuration in Region 2.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent;

b. $k_{eff} \leq 0.95$ under normal conditions, which includes an allowance for uncertainties as described in Section 9.6.2.4.3 of the SAR;

c. $k_{eff} \leq 0.98$ with optimum moderation, which includes an allowance for uncertainties as described in Section 9.6.2.4.3 of the SAR;

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks; and

e. Ten interior storage cells, as shown in Figure 4.3.1.2-1, precluded from use during fuel storage.

DESIGN FEATURES

4.3 Fuel Storage

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 397 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 968 fuel assemblies.

Figure 4.3.1.2-1

Fresh Fuel Storage Rack
Loading Pattern

<-----NORTH

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

"NO" Indicates a location in which fuel loading is prohibited.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The ANO-1 plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is in MODE 1, 2, 3, or 4. With the unit not in MODES 1, 2, 3, or 4, an individual with an active SRO or Reactor Operator license shall be designated as responsible for the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power unit.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the unit specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Safety Analysis Report (SAR);

b. The ANO-1 plant manager shall be responsible for overall safe operation of the unit and shall have control over those onsite activities necessary for safe operation and maintenance of the unit;

c. A specified corporate executive shall have corporate responsibility for overall unit nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the unit to ensure nuclear safety. The specified corporate executive shall be identified in the SAR; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

a. A non-licensed operator shall be on site when fuel is in the reactor and an additional non-licensed operator shall be on site when the reactor is in MODES 1, 2, 3, or 4.

b. The minimum shift crew composition for licensed operators shall meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) for one unit, one control room, and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

f. The operations manager or assistant operations manager shall hold an SRO license.

g. In MODES 1, 2, 3, or 4, an individual shall provide advisory technical support for the operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI ANS 3.1 - 1978 for comparable positions, except for the designated radiation protection manager, who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the ANO general manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 18 months. The provisions of SR 3.0.2 are applicable.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodine, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001 – 20.2402;

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.5 (Not Used).

5.5.6 (Not Used).

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Code terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Monthly	At least once per 31 days
Every 6 weeks	At least once per 42 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls to ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Tube Surveillance Program inspection frequencies.

a. The first steam generator tubing inspection performed in accordance with 5.5.9.b and 5.5.9.c.1 shall be considered as constituting the baseline condition for subsequent inspections.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

b. Examination Methods:

1. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end.

2. For examination of the sleeved steam generator tubing at the lower sleeve end, the indications will be compared to those obtained during the baseline sleeved tube inspection. Significant deviations between these indications will be considered sufficient evidence to warrant designation as a degraded tube. Direct quantification of the 40 percent through-wall plugging limit is available with eddy-current testing.

c. Selection and Testing:

The steam generator sample size is specified in Table 5.5.9-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies as specified in 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:

1. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

   i. All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and

   ii. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per 5.5.9.c.1.iii.

   A tube inspection (pursuant to 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

iii. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

(2) Group A-2: Unplugged tubes with sleeves installed.

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure 5.5.9-1.

iv. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to 5.5.9.d. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category of the OTSG.

v. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to 5.5.9.d. Tubes with ODIGA identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with ANO Engineering Report No. 00-R-1005-01.

2. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

3. The second and third sample inspections during each inservice inspection as required by Table 5.5.9-2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

4. The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected, are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes whose degradations have not been spanned by a sleeve must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.9.c.1.iii, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

(3) Where special inspections are performed pursuant to 5.5.9.c.2, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

d. The above-required inservice inspections of steam generator tubes shall be performed at the following frequencies:

1. The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of 40 months.

2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.9-2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 5.5.9.d.1 and the interval can be extended to 40 months.

3. Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   i. Primary-to-secondary leakage in excess of the limits of Specification 3.4.13 (inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification 5.5.9.c.1.iii, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table 5.5.9-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.

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*A group of tubes means:
(a) All tubes inspected pursuant to 5.5.9.c.1.iii, or
(b) All tubes in a steam generator less those inspected pursuant to 5.5.9.c.1.iii.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table 5.5.9-2.

ii. A seismic occurrence greater than the Operating Basis Earthquake,

iii A loss-of-coolant accident requiring actuation of the engineered safeguards, or

iv. A main steam line or feedwater line break.

e. Acceptance Criteria:

1. Terms as used in this program:

i. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

ii. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

iii. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

iv. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve or repaired by a rerolled joint.

The reroll repair process will be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.

v. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

vi. Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

vii. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with ANO Engineering Report No. 00-R-1005-01, Rev. 1.

Axially-oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.

viii. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.d.3.

ix. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the tubesheets, that portion of the tube outboard of the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table 5.5.9-2.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

TABLE 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspection	One[1]

Table Notation:

[1] The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION [2,3]

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug, reroll, or sleeve Defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug, reroll, or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes.	N/A	N/A

NOTES:

1. $S = \frac{3N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

2. For tubes inspected pursuant to 5.5.9.c.1.iii: No action is required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a report to NRC pursuant to 5.6.7.

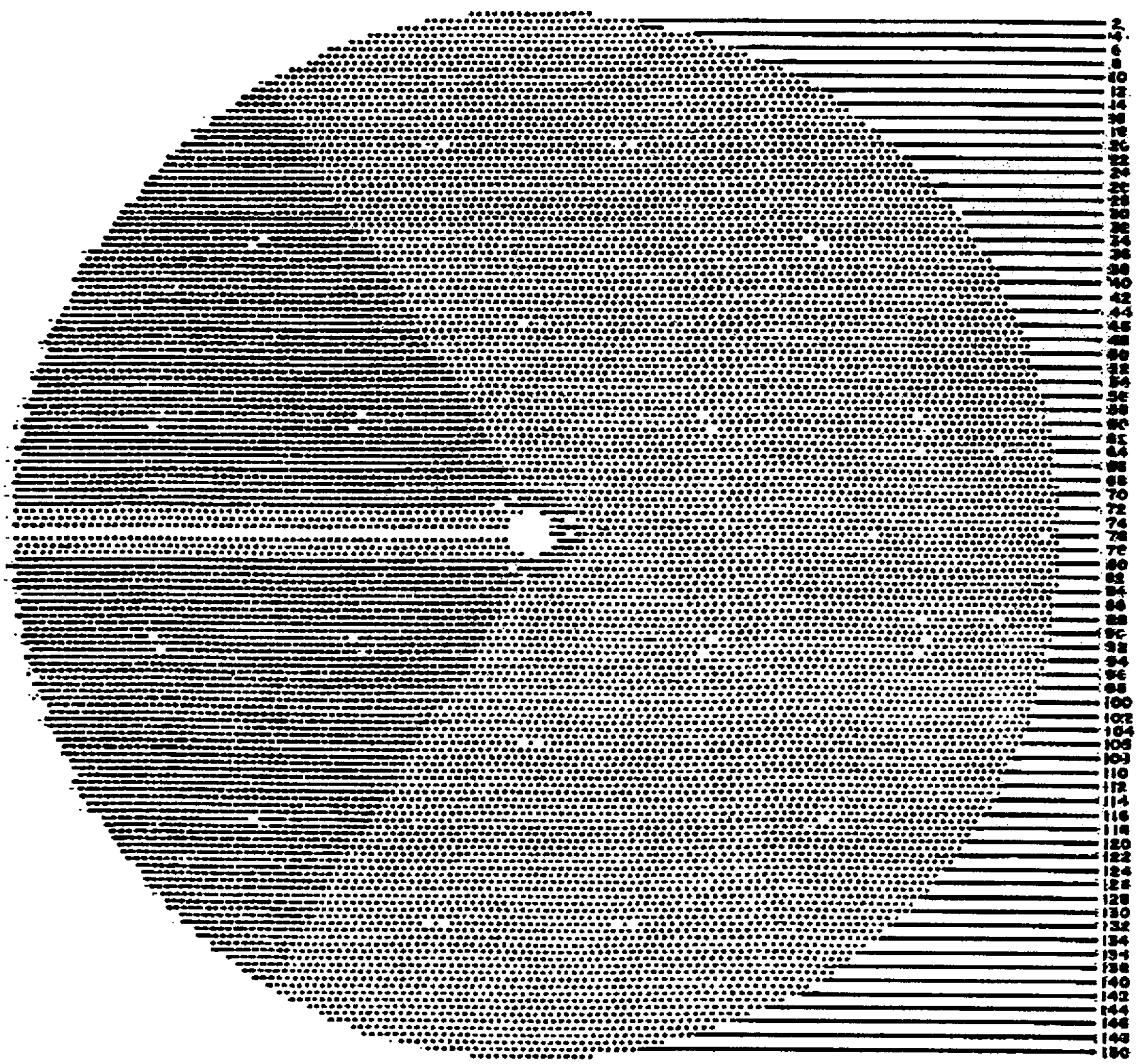
3. No more than ten thousand (10,000) sleeves may be installed in both ANO-1 steam generators combined.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

FIGURE 5.5.9-1

Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per 5.5.9.c.1.iii



DESCRIPTION	TUBE COUNT
Group A-1: Lane region tubes as defined in 5.5.9.c.1.iii(1)	382
Group A-3: Wedge shaped group depicted by darkened region of figure	4880

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safeguards (ES) ventilation systems filters at the frequencies specified in Regulatory Guide 1.52, Revision 2. The VFTP is applicable to the Penetration Room Ventilation System (PRVS), the Fuel Handling Area Ventilation System (FHAVS), and the Control Room Emergency Ventilation System (CREVS).

a. Demonstrate that an inplace cold DOP test of the high efficiency particulate (HEPA) filters shows:

   1. $\geq$ 99% DOP removal for the PRVS when tested at the system design flowrate of 1800 scfm $\pm$ 10% and the FHAVS when tested at the system design flowrate of 39000 cfm $\pm$ 10%; and

   2. $\geq$ 99.95% DOP removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm $\pm$ 10%.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

b. Demonstrate that an inplace halogenated hydrocarbon test of the charcoal adsorbers shows:

   1. ≥ 99% halogenated hydrocarbon removal for the PRVS when tested at the system design flowrate of 1800 cfm ± 10% and FHAVS when tested at the system design flowrate of 39000 cfm ± 10%; and

   2. ≥ 99.95% halogenated hydrocarbon removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%.

c. Demonstrate that a laboratory test of a sample of the charcoal adsorber meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

   1. < 5% for the PRVS;

   2. < 5% for the FHAVS; and

   3. when obtained as described in Regulatory Guide 1.52, Revision 2, for CREVS

      i. ≤ 2.5% for 2 inch charcoal adsorber beds; and

      ii. ≤ 0.5% for 4 inch charcoal adsorber beds.

d. Demonstrate for the PRVS, FHAVS, and CREVS, that the pressure drop across the combined HEPA filters, other filters in the system, and the charcoal adsorbers is < 6 inches of water when tested at the following system design flowrates ± 10%:

PRVS	1800 cfm
FHAVS	39000 cfm
CREVS	2000 cfm

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected temporary outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with the ODCM.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents;

c. A surveillance program to ensure that the quantity of radioactivity contained in all temporary outdoor liquid radwaste tanks: 1) that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents; and 2) that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations equal to the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

1. an API gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. water and sediment within limits;

b. Within 31 days following addition of new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days based on ASTM D-2276, Method A-2 or A-3; and

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance Frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

Proposed changes that do not meet these criteria shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the SAR.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.16 Reactor Building Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

In addition, the reactor building purge supply and exhaust isolation valves shall be leakage rate tested once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

Reactor Building leakage rate acceptance criteria is $\leq 1.0L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $< 0.60L_a$ for the Type B and Type C tests and $< 0.75L_a$ for Type A tests.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.1 Occupational Radiation Exposure Report

------------------------------NOTE------------------------------
A single submittal may be made for ANO. The submittal should combine sections common to both units.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 <u>Annual Radiological Environmental Operating Report</u>

------------------------------NOTE------------------------------
A single submittal may be made for ANO. The submittal should combine sections common to both units.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for ANO. The submittal shall combine sections common to both units . The submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1 Variable Low RCS Pressure – Temperature Protective Limits
3.1.1 SHUTDOWN MARGIN (SDM)
3.1.8 PHYSICS TESTS Exceptions – MODE 1
3.1.9 PHYSICS TEST Exceptions - MODE 2
3.2.1 Regulating Rod Insertion Limits
3.2.2 AXIAL POWER SHAPING RODS (APSR) Insertion Limits
3.2.3 AXIAL POWER IMBALANCE Operating Limits
3.2.4 QUADRANT POWER TILT (QPT)
3.2.5 Power Peaking
3.3.1 Reactor Protection System (RPS) Instrumentation
3.4.1 RCS Pressure, Temperature, and Flow DNB limits
3.4.4 RCS Loops – MODES 1 and 2
3.9.1 Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

Babcock & Wilcox Topical Report BAW-10179-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the COLR.

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 System (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.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.6 Reactor Building Inspection Report

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

5.6.7 Steam Generator Tube Surveillance Reports

a. Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

   1. Number and extent of tubes inspected;

   2. Location and percent of wall-thickness penetration for each indication of an imperfection;

   3. Identification of tubes plugged and tubes sleeved;

   4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

   5. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

   6. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

b. In addition, the Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 5.5.9-2 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP, or equivalent, while in the area by means of closed circuit television, or personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area and with the means to communicate with individuals in the area who are covered by such surveillance.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

B 2.0 SAFETY LIMITS (SLS)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormalities. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Ref. 2) and BWC (Ref. 3) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

The 95 percent confidence level that DNB will not occur is preserved by ensuring that the DNBR remains greater than the DNBR design limit based on the applicable CHF correlation for the core design. In the development of the applicable DNBR design limit, uncertainties in the core state variables, power peaking factors, manufacturing-related parameters, and the CHF correlation may be statistically combined to determine a statistical DNBR design limit. This statistical design limit protects the respective CHF design limit. Additional retained thermal margin may also be applied to the statistical DNBR design limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNB design methods preserves a 95 percent probability at a 95 percent confidence level that DNB will not occur (Ref. 4).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. The maximum fuel centerline temperatures are given by the relationships defined in SL 2.1.1.1 for the respective fuel designs and are dependent on whether the TACO2 (Ref. 5) or TACO3 (Ref. 6) analysis was utilized. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

BACKGROUND (continued)

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding. The oxidized cladding then exists in a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities (Ref. 7).

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip (also known as Pressure Temperature Trip);

e. Reactor Coolant Pump to Power trip;

f. Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE trip; and

g. RCS High Temperature trip.

APPLICABLE SAFETY ANALYSES (continued)

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation. In addition, the COLR identifies the pressure/temperature operating region that keeps the reactor from reaching an SL when operating up to design power.

The COLR presents the most limiting condition of pressure/temperature combinations for all possible reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed which bound the three pump and two pump (one pump in each loop) allowed operating conditions based on the expected minimum flow rates and maximum ALLOWABLE THERMAL POWER for these operating conditions.

The SLs are preserved by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE protective limits are preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," and are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. Automatic protection actions serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1 AND 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 8).

REFERENCES

1. SAR, Section 1.4, GDC 10.
2. BAW-10000A, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox, Lynchburg, VA, May 1976 .
3. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," Babcock & Wilcox, Lynchburg, VA, April 1985.
4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2, Babcock & Wilcox, Lynchburg, VA, October 1997.
5. BAW-10141P-A, Rev. 1, "TACO2 Fuel Pin Performance Analysis," Babcock & Wilcox, Lynchburg, VA, June 1983.
5. BAW-10162P-A, "TACO3 Fuel Pin Thermal Analysis Code," Babcock & Wilcox, Lynchburg, VA, October 1989.
7. SAR, Chapters 3 & 14.
8. 10 CFR 50.72.

B 2.0 SAFETY LIMITS (SLS)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

In SAR, Section 1.4 (Ref. 1), GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," and GDC 15, "Reactor Coolant System Design", address RCPB design and protection, respectively. The ANO-1 discussion regarding how GDC 15 is accomplished states that analysis and evaluation of all normal and abnormal operating conditions and transients are integrally related to all RCS and associated systems design. SAR Chapter 14 (Ref. 2) lists these abnormal operating conditions and transients and terms them "abnormalities". In addition, GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and abnormalities, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with the design codes (Ref. 3 and 4). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure prior to initial operation, according to the design code requirements. Inservice leak testing at not less than 2155 psig is also required, prior to MODE 2, following any opening of the reactor coolant system in accordance with ASME code, Section XI; IWA-5000. When performed at the end of refueling outages, this leak test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components (Ref. 5).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME code for Nuclear Power Plant Components (Ref. 3). The design basis transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal event from low power.

The startup event analysis (rod withdrawal at low power) (Ref. 2) is performed using conservative assumptions relative to pressure control devices.

APPLICABLE SAFETY ANALYSES (continued)

More specifically, no credit is taken for operation of the following:

a. Electromatic relief valve (ERV);

b. Steam line turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS B31.7 (Ref. 4), is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2750 psig.

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized significantly.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.3

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6).

SAFETY LIMIT VIOLATIONS (continued)

The allowed Completion Time of 1 hour is based on the importance of reducing power level to a MODE where the potential for challenges to safety systems is minimized.

2.2.4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

REFERENCES

1. SAR, Section 1.4, GDC 14, GDC 15, and GDC 28, 1988.
2. SAR, Chapter 14.
3. ASME Boiler and Pressure Vessel Code, Section III, 1965-S67, Article NB-7000.
4. USAS B31.7, Nuclear Power Piping, 1969.
5. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
6. 10 CFR 100.
7. 10 CFR 50.72.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specification.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. Reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.3 (continued)

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.12, "Fuel Handling Area Ventilation System." LCO 3.7.12 has an Applicability of "During movement of irradiated fuel assemblies in the fuel handling area." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.12 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.12 of "Suspend movement of irradiated fuel assemblies in the fuel handling area" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.4 (continued)

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability associated with operation in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate

BASES

LCO APPLICABILITY (continued)

LCO 3.0.5 (continued)

OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.6 (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry in Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the remaining OPERABLE support systems are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump

BASES

LCO APPLICABILITY (continued)

LCO 3.0.6 (continued)

suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Test Exception (STE) LCO are only applicable when the STE LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR APPLICABILITY (continued)

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the EFW pump testing.

b. High pressure injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers unit operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply.

BASES

SR APPLICABILITY (continued)

SR 3.0.2 (continued)

These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Reactor Building Leakage Rate Testing Program.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay

BASES

SR APPLICABILITY (continued)

SR 3.0.3 (conntinued)

period of 24 hours to perform the Surveillance. SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Satisfactory completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified

BASES

SR APPLICABILITY (continued)

SR 3.0.4 (continued)

conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability associated with operation in MODES 1, 2, 3 or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions per GDC 26 (Ref. 1). In MODES 3, 4, and 5, SDM requirements provide sufficient reactivity margin to maintain the core subcritical during these conditions.

In MODES 1 and 2 while critical, SDM requirements are met by the worth of the withdrawn CONTROL RODS which provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and abnormalities. In MODE 2 while subcritical and in MODE 3, with all safety rods withdrawn and the RPS not in Shutdown Bypass, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all CONTROL RODS , assuming the single CONTROL ROD of highest reactivity worth is fully withdrawn. In MODES 3, 4, or 5, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, the SDM defines the degree of subcriticality required to be maintained, assuming the CONTROL ROD of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of CONTROL RODS and soluble boric acid in the Reactor Coolant System (RCS). In MODES 1 and 2, the CONTROL RODS can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, for analyzed events initiated in MODES 1 and 2, the CONTROL RODS, together with the Chemical Addition and Makeup and Purification System, provide SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn (Ref. 1).

The Chemical Addition and Makeup and Purification System can compensate for fuel depletion, during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions (Ref. 1).

During operation in MODES 1 and 2, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5, "Safety Rod Insertion Limits") and the regulating rods within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits." In MODE 3, consideration must be given to the position of the safety rods and whether the RPS is in Shutdown Bypass in determining the required SDM. When the unit is

BACKGROUND (continued)

in MODES 3, 4, and 5, the SDM requirements are met by means of adjustments to the RCS boron concentration. Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable CONTROL ROD prior to reactor shutdown.

APPLICABLE SAFETY ANALYSES

For analyzed events in MODES 1 and 2 while critical, the minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and abnormalities, with assumption of the highest worth rod stuck out following a reactor trip.

In MODES 1 and 2 while critical, the acceptance criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events; and

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for abnormalities, and $\leq$ 280 cal/gm energy deposition for the rod ejection accident).

In MODES 3, 4, and 5, the SDM requirements must ensure that the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

In MODES 1 and 2 while critical, SDM satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical and in MODES 3, 4, and 5, SDM satisfies Criterion 4 of 10 CFR 50.36.

LCO

In MODES 1 and 2, and in MODE 3 when all safety rods are fully withdrawn and the RPS is not in Shutdown Bypass, SDM is a core design condition that can be ensured through CONTROL ROD positioning (regulating and safety groups) and through the soluble boron concentration.

In MODE 3, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, and in MODES 4 and 5, SDM represents a required degree of subcriticality that assumes the highest reactivity worth CONTROL ROD is fully withdrawn.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to ensure that the reactor remains subcritical.

In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5 and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron source concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid addition tank (BAAT) or the borated water storage tank (BWST). The operator should borate with the best source available for the unit conditions.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation. The reactivity effects that are considered in the reactivity balance are dependent upon the operational MODE of the unit. In general, the reactivity balance includes the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.1.1 (continued)

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC);

h. Moderator temperature coefficient (MTC); and

i. Doppler defect.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the point of adding heat (POAH), and the fuel temperature will be changing at the same rate as the RCS.

Using the MTC and Doppler defect accounts for the reactivity effects of power operation above the POAH.

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which may include performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. SAR, Section 1.4, GDC 26.
2. SAR, Chapter 3.
3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and abnormalities. Therefore, the reactivity balance is used as a measure of the agreement between the predicted core reactivity and the actual core reactivity during power operation. The periodic confirmation of the predicted core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CONTROL ROD, or burnable poison worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity. These could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing the predicted core reactivity with the actual core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations in ensuring the reactor can be brought safely to cold, subcritical conditions. The difference between the actual and predicted core reactivity is commonly referred to as a reactivity anomaly.

When the reactor is critical in MODE 1 or 2, a reactivity balance exists where the net reactivity is zero (referred to as the actual core reactivity state). A comparison of predicted core reactivity and the actual core reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions and the net reactivity is known to be zero. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as soluble boron and burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel remaining from the previous cycle provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical, the excess positive reactivity of the fuel is compensated by burnable absorbers, CONTROL RODS, APSRs, thermal feedback from the fuel and moderator, fission product poisons (mainly xenon and samarium), epithermal energy neutron absorbers, neutron leakage and the reactor coolant system (RCS) boron concentration. During cycle operation, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the primary method of compensating for the reduction in excess reactivity is through a reduction in the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are the establishment of the reactivity balance limit to ensure that unit operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation is, therefore, dependent upon an accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CONTROL ROD withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity (Ref. 2). These accident analysis evaluations rely on computer codes which have been qualified against available test data, operating unit data, and analytical benchmarks. Monitoring the core reactivity balance ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the requirements for reactivity control during the operating cycle.

The comparison between the actual reactivity condition of the critical reactor and the predicted initial core reactivity provides an opportunity for the normalization of the calculational models used to predict core reactivity. If the predicted core reactivity and the actual core reactivity at reference core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict reactivity requirements may not be accurate. If reasonable agreement between the actual and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the predicted reactivity condition from the actual reactivity condition during the operating cycle may be an indication that the calculational model is not adequate for the operating cycle or that an unexpected change in core conditions has occurred.

The normalization of the predicted reactivity parameters to the actual reactivity value is typically performed after reaching RTP following startup from a refueling outage, with the RCS temperature, CONTROL RODS, and APSRs in their reference positions and fission product poisons at their expected equilibrium concentrations. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated, as core conditions change during the cycle.

Reactivity balance satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled, once the core design is fixed. During operation, therefore, the conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the accident analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity of ± 1% $\Delta k/k$ has been established, based on engineering judgment. A ± 1% $\Delta k/k$ deviation in the predicted reactivity from the actual reactivity condition of the reactor is larger than expected for normal operation and should therefore be evaluated.

When the predicted core reactivity is within 1% $\Delta k/k$ of the actual reactivity value at steady state thermal conditions, the core is considered to be operating within acceptable design limits.

APPLICABILITY

In MODES 1 and 2, the limits on the core reactivity balance must be maintained to ensure an acceptable SDM and continued adherence to the assumptions used in the accident analyses. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed.

This Specification does not apply in MODES 3, 4, and 5, because the reactor is shutdown and the net reactivity condition of the reactor can not be easily determined and changes to core reactivity due to fuel depletion cannot occur.

In MODE 6, boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within acceptable bounds.

ACTIONS

A.1 and A.2

Should an anomaly develop between the actual core reactivity and the predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with the input assumptions used in the core design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of an abnormality or

ACTIONS (continued)

A.1 and A.2 (continued)

accident occurring during this period, and allows sufficient time to assess the physical condition of the core and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core reference conditions at the time of the reactivity balance, then a recalculation of the reactivity balance may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the appropriate reactivity parameter may be renormalized, and operation in MODE 1 may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing operating restrictions or surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity balance cannot be restored to within the ± 1% Δk/k limit, the unit must be brought to a MODE in which the LCO does not apply. As a conservative measure, the unit must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Required Action A.1 of LCO 3.1.1 would occur. The allowed Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by a periodic reactivity balance calculation that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition). The comparison is made considering that core conditions are fixed or stable, including CONTROL ROD and APSR positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed once prior to entering MODE 1 after each fuel loading as an initial check on core reactivity conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.2.1 (continued)

core reactivity to the measured value may take place within the first 60 effective full power days (EFPD) after each fuel loading. The required Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1 is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly. The 60 EFPD after entering MODE 1 allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

REFERENCES

1. SAR, Section 1.4, GDC 26, GDC 28, and GDC 29.
2. SAR, Chapter 3A and 14.
3. 10 CFR 50.36

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and associated Reactor Coolant System (RCS) shall be designed so that in the power operating range the net affect of the prompt inherent nuclear feedback characteristic tends to compensate for a rapid increase in reactivity.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). Therefore, with a negative MTC a coolant temperature increase will cause a reactivity decrease. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is 95% RTP or greater. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure the MTC does not become more negative than the value assumed in the safety analyses.

APPLICABLE SAFETY ANALYSES

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are initial conditions in the safety analyses, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations for overheating events, to ensure the accident results are bounding.

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis; and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

APPLICABLE SAFETY ANALYSES (continued)

Accidents that cause core overheating (either decreased heat removal or increased power production) must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the CONTROL ROD withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to positive MTC is the startup accident.

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction, combined with the large negative MTC, may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power may be produced with all CONTROL ROD assemblies inserted, except the most reactive one. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC.

In MODES 1 and 2 while critical, MTC satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, MTC satisfies Criterion 4 of 10 CFR 50.36.

LCO

LCO 3.1.3 requires the MTC to be within specified limits to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit of +0.9E-4 $\Delta k/k/°F$ (corrected to 95% RTP) on positive MTC, when THERMAL POWER is < 95% RTP, ensures that core overheating accidents will not violate the accident analysis assumptions. The requirement for a non-positive MTC, when THERMAL POWER is ≥ 95% RTP, ensures that core operation will be stable.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be controlled directly once the core design is fixed during operation, therefore, the LCO can only be ensured through measurement. The surveillance check at BOC on MTC provides confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from power operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis, is accepted as valid once the BOC measurement is used for normalization.

ACTIONS

A.1

MTC is a core physics parameter determined by the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis assumptions. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, for reaching MODE 3 conditions from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The SR for measurement of the MTC at the beginning of each fuel cycle provides for confirmation of the limiting MTC values. The MTC changes slowly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced with fuel depletion.

The requirement for measurement, prior to initial operation in MODE 1, satisfies the confirmatory check on the most positive (least negative) MTC value. MTC values are extrapolated and compensated to permit direct comparison to the specified MTC limits.

REFERENCES

1. SAR, Section 1.4, GDC 11.
2. SAR, Chapter 3A and 14.
3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of SDM.

The applicable criteria for these design requirements are GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy-and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CONTROL ROD to become inoperable or to become misaligned from its group. CONTROL ROD inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD alignment and OPERABILITY are related to core operation within design power peaking limits and the core design requirement of a minimum SDM.

Limits on CONTROL ROD alignment and OPERABILITY have been established, and all CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CONTROL RODS are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The CONTROL RODS provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity control during normal operation and transients, and their movement is normally controlled in automatic by a rod control system.

The axial position of the CONTROL RODS is indicated by three independent systems, which are the relative position indicators, the absolute position indicators, and the zone reference indicators (see LCO 3.1.7, "Position Indicator Channels").

BACKGROUND (continued)

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the CRDCS. There is one counter for each CONTROL ROD drive. Individual rods in a group, when aligned to the same power supply, all receive the same signal to move; therefore, the counters for all rods in a group should normally indicate the same position. The Relative Position Indicator System is considered highly precise. However, if a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod.

The Absolute Position Indicator System provides a highly accurate indication of actual CONTROL ROD position, but at a lower precision than the relative position indicators. This system is based on the signals from a series of reed switches spaced along a tube.

Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators.

APPLICABLE SAFETY ANALYSES

CONTROL ROD misalignment and inoperability accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing CONTROL ROD inoperability or misalignment are that:

a. There shall be no violations of:

    1. specified acceptable fuel design limits, or

    2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after an abnormality or accident.

Two types of misalignment are distinguished during MODES 1 and 2. During movement of a CONTROL ROD group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs when one CONTROL ROD drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

The accident analysis and reload safety evaluations define regulating rod insertion limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 3). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted if the increase in local LHR is within

APPLICABLE SAFETY ANALYSES (continued)

the design limits. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 3).

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the local core LHRs are verified to be within their limits in the COLR. When a CONTROL ROD is misaligned, the assumptions that are used to determine the regulating rod insertion limits, APSR insertion limits, AXIAL POWER IMBALANCE limits, and QPT limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and local core LHRs must be verified directly by incore mapping. Bases Section 3.2, "Power Distribution Limits," contains a more complete discussion of the relation of LHR to the operating limits.

In MODES 1 and 2 while critical, the CONTROL ROD group alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODE 2 while subcritical, the CONTROL ROD group alignment limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The limits on CONTROL ROD group alignment, safety rod insertion, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Actions in these LCOs ensure that deviations from the alignment limits will either be corrected or that THERMAL POWER will be adjusted, so that excessive local LHRs will not occur and the requirements on SDM and ejected rod worth are preserved.

The limit for individual CONTROL ROD misalignment is 6.5% (approximately 9 inches) deviation from the group average position. This value is established, based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures.

For the purpose of complying with this LCO, the position of a misaligned rod is not included in the calculation of the rod group average position. A CONTROL ROD is not considered to be inoperable due solely to misalignment. A CONTROL ROD is considered to be inoperable if it is not free to insert into the core within the required insertion time, or as directed by LCO 3.1.7, "Position Indicator Channels."

Failure to meet the requirements of this LCO may produce unacceptable LHRs, or unacceptable SDM or ejected rod worth, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and resultant local power peaking would not exceed fuel design limits. In MODES 3, 4, 5, and 6, the OPERABILITY of the CONTROL RODS has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

ACTIONS

A.1.1

Compliance with Required Actions of Condition A allows for continued power operation with one CONTROL ROD inoperable, or misaligned from its group average position, or both. Since the rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement established in the COLR within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

A.1.2

If the SDM is less than the limit specified in the COLR, then the restoration of the required SDM requires increasing the RCS boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

A.2.1

Alignment of the inoperable or misaligned CONTROL ROD may be accomplished by either moving the single CONTROL ROD to the group average position, or by moving the remainder of the group to the position of the single inoperable or misaligned CONTROL ROD. Either action can be used to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD group sequence, overlap, and insertion limits of LCO 3.2.1, "Regulating Rod Insertion Limits," given in the COLR. THERMAL POWER must

ACTIONS (continued)

A.2.1 (continued)

also be restricted, as necessary, to the value allowed by the insertion limits of LCO 3.2.1. The required Completion Time of 2 hours is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR. This option of inserting the group to the position of the misaligned rod is not available if a safety rod is misaligned, since the limits of LCO 3.1.5, "Safety Rod Insertion Limits," would be violated. If realignment of the CONTROL ROD to the group average or alignment of the group to the misaligned CONTROL ROD is not completed within 1 hour, the rod shall be considered inoperable.

A.2.2.1

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

A.2.2.2

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% $\Delta k/k$ at RTP or 1.00% $\Delta k/k$ at zero power (Ref. 3). This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

A.2.2.3

Performance of SR 3.2.5.1 provides a determination of the local core LHRs using the Incore Detector System. Verification of the local core LHRs from an incore power distribution map is necessary to ensure that excessive local LHRs will not occur due to CONTROL ROD misalignment. This is necessary because the assumption that all CONTROL RODS are aligned (used to determine the regulating rod insertion, AXIAL POWER IMBALANCE, and QPT limits) is not valid when the CONTROL RODS are not aligned. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and adequate time is allowed to obtain an incore power distribution map.

ACTIONS (continued)

<u>A.2.2.3</u> (continued)

Required Action A.2.2.3 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

<u>B.1</u>

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

<u>C.1.1</u>

More than one CONTROL ROD becoming inoperable or misaligned from their group average position, or both, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

<u>C.1.2</u>

If the SDM is less than the limit specified in the COLR, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

<u>C.2</u>

If more than one CONTROL ROD is inoperable or misaligned from their group average position, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

ACTIONS (continued)

C.2 (continued)

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL ROD motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by approximately 1.5% (approximately 2 inches) will not cause radial or axial power tilts, or oscillations, to occur. No additional allowances for instrument uncertainty are required to be incorporated in the implementing procedures for this parameter. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between typical performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is otherwise determined to be capable of being fully inserted, the CONTROL ROD(S) may continue to be considered OPERABLE unless inoperable for some other reason. At any time, if a CONTROL ROD(S) is immovable, a determination of the capability to fully insert (OPERABILITY) the CONTROL ROD(S) must be made, and appropriate action taken.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The CONTROL ROD drop time given in the safety analysis is 1.66 seconds to 3/4 position insertion (Ref. 5). This 1.66 seconds includes 0.14 seconds delay time for opening of the CRD breakers and for CRDM unlatch. Using the CONTROL ROD position versus time and time versus reactivity insertion curves gives a value of 1.4 seconds to 2/3 reactivity insertion upon which the accident analysis is based (Ref. 3). The former value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at 3/4 insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. The CONTROL ROD drop time is the total elapsed time from the loss of power to the control rod drive (CRD) breaker under voltage coils until the CONTROL ROD has completed approximately 104 inches of travel from the fully withdrawn position. The safety analysis has included a CRD breaker time delay of 0.080 seconds in SAR Chapter 14 (Ref. 3). If the trip test measurement is begun with the opening of the CRD breakers, the required trip insertion time shall be reduced to 1.58 seconds and the CRD breaker time delay shall be verified to be less than or equal to 0.080 seconds.

Measuring CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or CONTROL ROD drop time. This Surveillance is performed during a unit outage, due to the unit conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

This testing is normally performed with all reactor coolant pumps operating and average moderator temperature ≥ 525°F to simulate a reactor trip under actual conditions. However, if the CONTROL ROD drop times are determined with less than four reactor coolant pumps operating, a Note allows operation to continue, provided operation is restricted to the pump combination utilized during the CONTROL ROD drop time determination or pump combinations providing less total reactor coolant flow.

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. SAR, Chapter 3A and 14.
4. 10 CFR 50.36.
5. SAR, Chapter 3.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

BACKGROUND

The insertion limits of the CONTROL RODS are initial condition assumptions in all safety analyses that assume CONTROL ROD insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate.

The applicable criteria for the reactivity and power distribution design requirements are GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on safety rod insertion have been established, and all CONTROL ROD positions are monitored and controlled during operation in MODES 1 and 2 to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The regulating groups are used for precise reactivity control of the reactor. The positions of the regulating groups are normally automatically controlled by the automatic control system, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). In MODES 1 and 2, the regulating groups must be maintained above designated insertion limits and are typically near the fully withdrawn position during normal operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature and fuel burnup.

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. Prior to entry into MODE 2 from MODE 3, the safety groups must be fully withdrawn. The safety groups must be completely withdrawn from the core prior to withdrawing any regulating groups during an approach to criticality. The safety groups remain in the fully withdrawn position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all CONTROL RODS, except the most reactive rod, are assumed to insert into the core. The safety groups shall be at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating groups may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Insertion Limits." The safety group and regulating rod group insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from RTP. The combination of regulating groups and safety groups (less the most reactive rod, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to achieve the required SDM at rated no load temperature (Ref. 3).

The acceptance criteria for addressing safety and regulating rod group insertion limits and inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or
   2. RCS pressure boundary integrity; and

b. The core must remain subcritical after an abnormality. Although the SAR does not state this as an acceptance criteria for the main steam line break event, B & W has placed a design objective on this event that the core remains subcritical throughout the event (Ref. 4).

In MODES 1 and 2 while critical, the safety rod insertion limits satisfy Criteria 2 and 3 of 10 CFR 50.36 (Ref. 5). In MODE 2 while subcritical, the safety rod insertion limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The safety groups must be fully withdrawn any time the reactor is in MODE 1 or 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and achieve the required SDM following a reactor trip.

This LCO has been modified by a Note indicating the LCO requirement is suspended for those safety rods which are inserted solely due to testing in accordance with SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the safety group to move below the LCO limits, which would normally violate the LCO.

APPLICABILITY

The safety groups must be within their insertion limits with the reactor in MODES 1 and 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and achieve the required SDM following a reactor trip. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

ACTIONS

A.1.1, A.1.2, and A.2

The safety rod must be declared inoperable within a 1 hour time frame. This requires entry into LCO 3.1.4, "CONTROL ROD Group Alignment Limits." In addition, since the safety rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

Restoration of the required SDM, if necessary, requires increasing the boron concentration, since the safety rod may remain misaligned and not be providing its normal negative reactivity on tripping. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

The allowed Completion Time of 1 hour provides an acceptable time for evaluating and repairing minor problems without allowing the unit to remain in an unacceptable condition for an extended period of time.

B.1.1 and B.1.2

When more than one safety rod is not fully withdrawn, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

In this situation, SDM verification must include the worth of any rod not capable of being fully inserted as well as the CONTROL ROD of maximum worth.

ACTIONS (continued)

B.2

If more than one safety rod is not fully withdrawn, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn ensures the safety rods are available to provide reactor shutdown capability.

Verification that individual safety rod positions are fully withdrawn at a 12 hour Frequency allows the operator to detect a safety rod beginning to deviate from its expected position. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of the safety rods.

REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. SAR, Chapters 3 and 4.
4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2.
5. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and APSR alignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are GDC 10, "Reactor Design," and GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause an APSR to become inoperable or to become misaligned from its group. APSR inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution. Therefore, APSR alignment and OPERABILITY are related to core operation within design power peaking limits.

Limits on APSR alignment and OPERABILITY have been established, and all APSR and CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution limits defined by the design peaking limits are preserved.

APSRs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

The APSRs are arranged into groups that are radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs, which are used to assist in control of the axial power distribution, are positioned manually and do not trip.

LCO 3.1.6 is conservatively based on use of black (Ag-In-Cd) APSRs and bounds use of gray (Inconel) APSRs. The reactivity worth of black APSRs is greater than that of gray APSRs; thus the impact of black APSR misalignment on the core power distribution is greater.

APPLICABLE SAFETY ANALYSES

There are no explicit safety analyses associated with misaligned APSRs. However, alignment of the APSRs is required to prevent inducing a QUADRANT POWER TILT. The LCOs governing APSR alignment are provided because the power distribution analysis supporting LCO 3.2.1, LCO 3.2.3 and LCO 3.2.4 assumes the APSRs are aligned.

During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. Continued operation of the reactor with a misaligned APSR is allowed if Section 3.2, "Power Distribution Limits," are preserved.

Because ANO-1 uses gray APSRs, the APSR alignment limits satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

The limits on CONTROL ROD group alignment, safety rod withdrawal, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Action in this LCO ensures deviations from the alignment limits will be adjusted so that excessive local LHRs will not occur.

The limit for individual APSR misalignment is 6.5% (approximately 9 inches) deviation from the group average position. This value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures. The position of an inoperable APSR is not included in the calculation of the APSR group's average position.

Failure to meet the requirements of this LCO may produce unacceptable LHRs, which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of APSRs have the potential to affect the safety of the unit. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down, and excessive local LHRs cannot occur from APSR misalignment.

ACTIONS

A.1

The ACTIONS described below are required if one APSR is inoperable. The unit is not allowed to operate with more than one inoperable APSR. This would require the reactor to be placed in MODE 3, in accordance with LCO 3.0.3.

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR, while maintaining APSR insertion, in accordance with the limits in the COLR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. This alternative assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason, APSR group movement is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

The reactor may continue in operation with the APSR misaligned if the limits on power peaking are surveilled within 2 hours to determine if power peaking is still within limits. Also, since any additional movement of the APSRs may result in additional imbalance, Required Action A.1 also requires the power peaking surveillance to be performed again within 2 hours after each APSR movement.

B.1

The unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems. In MODE 3, APSR group alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within 6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. In addition, APSR position is continuously available to the operator in the control room so that during actual APSR motion, deviations can immediately be detected.

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 28.

2. 10 CFR 50.46.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

According to the SAR discussion of GDC 13 (Ref. 1), adequate instrumentation and controls are provided to maintain operating variables within prescribed ranges for normal operation and monitor accident conditions as appropriate to assure adequate safety. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD and APSR alignment and insertion limits.

The OPERABILITY, including position indication, of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the CONTROL RODS and APSRs is assumed in the safety analysis, which directly affect core power distributions and assumptions of available SDM.

Mechanical or electrical failures may cause a CONTROL ROD or APSR to become misaligned from its group. CONTROL ROD or APSR misalignment may cause increased local linear heat rates (LHRs), due to the asymmetric reactivity distribution, and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD and APSR alignment are related to core operation within design LHR limits and the core design requirement of a minimum SDM. CONTROL ROD and APSR position indication is needed to assess OPERABILITY and alignment.

Limits on CONTROL ROD and APSR alignment, and CONTROL ROD and APSR group position have been established, and all CONTROL ROD and APSR positions are monitored and controlled during operation to ensure that the power distribution and reactivity limits defined by the design LHR and SDM limits are preserved.

Three methods of CONTROL ROD and APSR position indication are provided in the Control Rod Drive Control System. The three means are by absolute position indicator, relative position indicator transducers, and zone reference indicators. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the control rod drive mechanism (CRDM) motor tube extension. Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD or APSR assembly leadscrew extension comes near. As the leadscrew and CONTROL ROD or APSR move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications, and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven

BACKGROUND (continued)

indicators are called zone reference indicators. The relative position indicator transducer is a potentiometer, driven by a step motor that produces a signal proportional to CONTROL ROD or APSR position, based on the electrical pulse steps that drive the CRDM.

CONTROL ROD and APSR position indicating readout devices located in the control room consist of single rod position meters on a position indication panel and group average position meters. A selector switch permits either relative or absolute position indication to be displayed on all of the individual position indication meters. Indicator lights are provided on the individual position indication panel to indicate when each CONTROL ROD or APSR is fully withdrawn, fully inserted, enabled, or transferred, and whether a rod position asymmetry alarm condition is present. Additional indicators show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD and APSR group. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD and APSR position. Therefore, the potential for operation in violation of design LHR or SDM limits is increased.

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. LHR, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. CONTROL ROD and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design LHRs, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Insertion Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; and LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"). The CONTROL ROD and APSR positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the unit is operating within the bounds of the accident analysis assumptions.

In MODES 1 and 2 while critical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 4 of 10 CFR 50.36.

LCO

LCO 3.1.7 specifies that one position indicator channel be OPERABLE for each CONTROL ROD and APSR.

This requirement ensures that CONTROL ROD and APSR position indication during MODES 1 and 2 and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channel ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, LHR and SDM can be controlled within acceptable limits.

APPLICABILITY

In MODES 1 and 2, OPERABILITY of the position indicator channel is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required because the reactor is shut down with the required minimum SDM and is not generating significant THERMAL POWER.

ACTIONS

A.1

If the required position indicator channel is inoperable for one or more rods, the position of the CONTROL ROD or APSR is not known with certainty. Therefore, each affected CONTROL ROD or APSR must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, this CHANNEL CHECK will be used to detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.1 (continued)

When compared to other channels, the agreement criteria between channels is determined by the unit staff. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required position indicator channel.

The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred.

SR 3.1.7.2

A CHANNEL CALIBRATION of the required position indication channel verifies that the channel responds within the necessary range and accuracy.

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. SAR, Section 1.4, GDC 13.
2. SAR, Chapter 14.
3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions Systems - MODE 1

BASES

BACKGROUND

The purpose of this LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.8 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed execution of testing required to ensure the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still in effect and by the SRs. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on linear heat rate (LHR), ejected rod worth, and shutdown capability are maintained during the PHYSICS TESTS.

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, one or more LCOs must sometimes be suspended to make completion of PHYSICS TESTS possible or practical.

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in:

LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," for the restricted operation region only;
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and
LCO 3.2.4, "QUADRANT POWER TILT (QPT)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the LHR (in MODE 1 PHYSICS TESTS) within limits, maintaining ejected rod worth within limits by restricting regulating rod insertion to within the acceptable operating region or the restricted operating region, by limiting maximum THERMAL POWER and by maintaining SDM within the limit provided in the COLR. Therefore, surveillance of the LHR and SDM is required to verify that their limits are not exceeded. The limits for the LHR are specified in the COLR. Refer to the Bases for LCO 3.2.5 for a complete discussion of LHR. During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR limits may be suspended. However, the results of the safety analysis are not adversely impacted if verification that core LHRs are within their limits is obtained, while one or more of the LCOs is suspended. Therefore, SRs are placed on LHR during MODE 1 PHYSICS TESTS when THERMAL POWER exceeds 20% RTP to verify that the core LHRs remain within their limits. Periodic verification of these factors allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables. Among the process variables involved are AXIAL POWER IMBALANCE and QPT, which represent initial condition input (power peaking) for the accident analysis. Also involved are the movable control components, i.e., the regulating rods and the APSRs, which affect power peaking. The limits for these variables are specified for each fuel cycle in the COLR.

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion for the other LCOs is provided in their respective Bases.

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups, and permits AXIAL POWER IMBALANCE and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1 (for the restricted operation region only, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP;

c. LHR is maintained within limits specified in the COLR while operating at greater than 20% RTP; and

d. SDM is verified to be within the limit provided in the COLR.

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. Eighty-five percent RTP is consistent with the maximum power level for conducting the intermediate core power distribution test specified in Reference 3. The nuclear overpower trip setpoint is reduced so that a similar margin exists between the steady state condition and trip setpoint as exists during normal operation at RTP.

LCO provision c is modified by a Note that requires the adherence to LHR requirements only when THERMAL POWER is greater than 20% RTP. This establishes an LCO provision that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

APPLICABILITY

This LCO is applicable in MODE 1, when the reactor has completed low power testing and is in power ascension, or during power operation with THERMAL POWER > 5% RTP but ≤ 85% RTP. This LCO is applicable for power ascension testing, as described in SAR Section 3A.9 (Ref. 3). In MODE 2, Applicability of this LCO is not required because LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 2," addresses PHYSICS TESTS exceptions initiated in MODE 2. In MODES 3, 4, 5, and 6, Applicability is not required because PHYSICS TESTS are not performed in these MODES.

ACTIONS

A.1 and A.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

B.1

If THERMAL POWER exceeds 85% RTP, then 1 hour is allowed for the operator to reduce THERMAL POWER to within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by PHYSICS TESTS exceptions.

If the nuclear overpower trip setpoint is not within the specified limits, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by these PHYSICS TESTS exceptions.

ACTIONS (continued)

B.1 (continued)

If the results of the incore flux map indicate that LHR has exceeded its limit, then PHYSICS TESTS are suspended. This action is required because of direct indication that the core LHR, which is a fundamental initial condition for the safety analysis, is excessive. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

This Condition is modified by a Note that requires performance of the Required Action only when THERMAL POWER is greater than 20% RTP. This establishes an ACTIONS entry Condition that is consistent with LCO provision c and the Applicability of LCO 3.2.5, "Power Peaking."

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that THERMAL POWER is ≤ 85% RTP ensures that the required additional thermal margin has been established prior to and during PHYSICS TESTS. The required Frequency of once per hour allows the operator adequate time to determine any degradation of the established thermal margin during PHYSICS TESTS.

SR 3.1.8.2

Verification that core LHRs are within their limits ensures that core LHR and departure from nucleate boiling ratio will remain within their limits, while one or more of the LCOs that normally control these design limits are out of specification. The required Frequency of 2 hours allows the operator adequate time for collecting a flux map and for performing the LHR verification, based on operating experience. If SR 3.2.5.1 is not met, PHYSICS TESTS are suspended and LCO 3.2.5 applies. This Frequency is more conservative than the Completion Time for restoration of the individual LCOs that preserve the LHR limits.

This SR is modified by a Note that requires performance only when THERMAL POWER is greater than 20% RTP. This establishes a performance requirement that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

SR 3.1.8.3

Verification that the nuclear overpower trip setpoint is within the limit specified for each PHYSICS TEST ensures that core protection at the reduced power level is established during the PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS at each testing plateau allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. CONTROL ROD position;

c. Doppler defect;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. Xenon concentration; and

g. Moderator defect.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. SAR, Section 3A.9.

4. SAR, Section 13.3, 13.4 and 13.6.

5. SAR, Section 13.4, Table 13-2.

6. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The purpose of this MODE 2 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

BACKGROUND (continued)

Examples of MODE 2 PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worth, and reactivity coefficients.

APPLICABLE SAFETY ANALYSES

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more of the LCOs must be suspended to make completion of PHYSICS TESTS possible or practical.

It is acceptable to suspend the following LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still maintained and by the SRs:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality."

Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on THERMAL POWER and shutdown capability are maintained during the PHYSICS TESTS.

Shutdown capability is preserved by limiting THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow range instrumentation for unit control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics.

This LCO also allows suspension of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, and LCO 3.4.2, provided:

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ 5% RTP;

c. Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit is OPERABLE; and

d. SDM is within the limit provided in the COLR.

The limits of LCO 3.2.3 and LCO 3.2.4 are not exempted by this specification because they do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

APPLICABILITY

This LCO is applicable when the reactor is either subcritical or critical with THERMAL POWER ≤ 5% RTP. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions. This LCO is applicable for initial criticality or low power testing, as described in SAR Section 3A.9 (Ref. 3). In MODE 1, Applicability of this LCO is not required because LCO 3.1.8, "PHYSICS TESTS Exceptions," addresses PHYSICS TESTS exceptions in MODE 1. In MODES 3, 4, 5, and 6, a test exception LCO is not required because the excepted LCOs do not apply in these MODES.

ACTIONS

A.1

If THERMAL POWER exceeds 5% RTP, a positive reactivity addition could be occurring, and a nuclear excursion could result. To ensure that local LHR, DNBR, and RCS pressure limits are not violated, the reactor is immediately tripped. The necessary prompt action requires manual operator action to open the control rod drive trip breakers without attempts to reduce THERMAL POWER by actuating the control system (i.e., CONTROL ROD insertion or RCS boration).

B.1 and B.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

C.1

If the nuclear overpower trip setpoint is > 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

The nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is not required when the reactor power level is above the operating range of the instrumentation channel. For example, if the reactor power level is above the source range channel operating range, then only the intermediate range high startup rate CONTROL ROD withdrawal inhibit is required to be functional.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that THERMAL POWER is ≤ 5% RTP ensures that local LHR, DNBR, and RCS pressure limits are not violated and that entry into Actions Condition A is performed promptly. Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit.

SR 3.1.9.2

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established during PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SR 3.1.9.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. Xenon concentration;

g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);

h. Moderator defect, when above the POAH; and

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. SAR, Section 3A.9.
4. SAR, Section 13.3, 13.4 and 13.6.
5. SAR, Section 13.4, Table 13-2.
6. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial condition assumptions used in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect the core power distributions, the worth of a potential ejected rod, the assumptions of SDM, and the initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are described in SAR, Section 1.4, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are not violated.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). To achieve this approximately linear relationship, the regulating rod groups are withdrawn and operated in a predetermined sequence. The automatic control system controls reactivity by moving the regulating rod groups in sequence within analyzed ranges. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are normally controlled automatically by the automatic control system but can also be controlled manually. They are capable of rapid reactivity changes compared with borating or diluting the Reactor Coolant System (RCS).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that ensure that the criteria specified in 10 CFR 50.46 (Ref. 2) are not violated. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate limits in the COLR. Operation within the linear heat rate limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced

BACKGROUND (continued)

reactor coolant flow accident. In addition to the linear heat rate limits, certain reactivity limits are met by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and support the minimum required SDM in MODES 1 and 2.

This LCO is required to minimize fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accidents requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or abnormalities. The LCOs governing regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 4).

d. The CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM which assumes the highest worth CONTROL ROD stuck fully withdrawn.

Fuel cladding damage does not occur when the core is operated outside the conditions of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs with the simultaneous violation of one or more of the LCOs limiting the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local linear heat rates (LHRs).

The SDM requirement is met by limiting the regulating and safety rod insertion limits such that sufficient inserted reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes that the maximum worth rod remains fully withdrawn upon trip (Ref. 4). Operation at the SDM based regulating rod insertion limit may also indicate that the maximum ejected rod worth could be equal to the limiting value.

APPLICABLE SAFETY ANALYSES (continued)

Operation at the regulating rod insertion limits may cause the local core power to approach the maximum linear heat generation rate or peaking factor with the allowed QPT present.

The regulating rod and safety rod insertion limits ensure that the safety analysis assumptions for SDM, ejected rod worth, and power distribution peaking factors remain valid (Refs. 3 and 4).

The regulating rod insertion limits LCO satisfies Criterion 2 of 10 CFR 50.36 (Ref. 5).

LCO

The limits on regulating rod group physical insertion, sequence, and overlap, as defined in the COLR, must be maintained because they ensure that the resulting power distribution is within the range of analyzed power distributions and that the SDM and ejected rod worth are maintained.

The overlap between regulating groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod motion.

Error adjusted maximum allowable setpoints for regulating rod insertion are provided in the COLR. The setpoints are derived by an adjustment of the measurement system independent limits to allow for THERMAL POWER level uncertainty and rod position errors.

LCO 3.2.1 has been modified by a Note that suspends the LCO requirement for those regulating rods not within the limits of the COLR solely due to testing in accordance with SR 3.1.4.2, which verifies the freedom of the rods to move. This SR may require the regulating rods to move below the LCO limit, out of group sequence, or beyond group overlap requirements, which would otherwise violate the LCO.

APPLICABILITY

The regulating rod physical insertion, sequence, and overlap limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the validity of the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, because neither the power distribution nor ejected rod worth assumptions are exceeded in these MODES. SDM in MODES 3, 4, and 5 is governed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

ACTIONS

The regulating rod insertion setpoints provided in the COLR are based on the initial conditions assumed in the accident analyses and on the SDM. Specifically, separate insertion setpoints are specified to determine whether the unit is operating in violation of the initial conditions (e.g., the range of power distributions) assumed in the accident analyses or whether the unit is in violation of the SDM or ejected rod worth limits. Separate insertion setpoints are provided because different Required Actions and Completion Times apply, depending on which insertion setpoint has been violated. The area between the boundaries of the acceptable operation and unacceptable operation regions, illustrated on the regulating rod insertion setpoint figures in the COLR, is the restricted operation region. The actions required when operation occurs in the restricted operation region are described under Condition A. The actions required when operation occurs in the unacceptable operation region are described under Condition D. The actions required when operation occurs with the regulating rod group sequence or overlap requirements not met are described under Condition C.

A.1

Operation with the regulating rods in the restricted operation region shown on the regulating rod insertion setpoint figures specified in the COLR potentially violates the LOCA LHR limits, or the loss of flow accident DNB peaking limits.

For verification that LHRs are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that LHRs are within their limits ensures that operation with the regulating rods inserted into the restricted operation region does not violate the ECCS or DNB criteria. The required Completion Time of 2 hours is acceptable in that it allows the operator sufficient time for obtaining a power distribution map and for verifying the LHRs. Repeating SR 3.2.5.1 every 2 hours is acceptable because it ensures that continued verification of the LHRs is performed as core conditions (primarily regulating rod insertion and induced xenon redistribution) change.

Monitoring the LHRs does not provide verification that the reactivity insertion rate on the rod trip or the ejected rod worth limit is maintained, because worth is a reactivity parameter rather than a power peaking parameter. However, if the COLR figures do not show that a rod insertion setpoint is ejected rod worth limited, then the ejected rod worth is no more limiting than the SDM based rod insertion limit in the core design. Ejected rod worth limits are independently maintained by the Required Actions of Conditions A and D.

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

ACTIONS (continued)

A.2

Indefinite operation with the regulating rods inserted in the restricted operation region is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, reactivity limits may not be met and the abnormal regulating rod insertion may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long term fuel depletion pattern. Therefore, restoration of regulating rod groups to within their limits is required within 24 hours after discovery of failure to meet the requirements of this LCO. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions, thereby limiting the potential for an adverse xenon redistribution.

B.1

If the regulating rods cannot be positioned within the acceptable operation region shown on the figures in the COLR within the required Completion Time (i.e., Required Action A.2 not met), then the setpoints can be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems. Operation for up to 2 hours more in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the regulating rod position out of specification in this relatively short time period.

C.1

Operation with the regulating rod groups out of sequence or with the group overlap limits exceeded may represent a condition beyond the assumptions used in the safety analyses. The design calculations assume no deviation in nominal overlap between regulating rod groups. However, small deviations in group overlap, as allowed by the COLR, may occur and would not cause significant differences in core reactivity, in power distribution, or rod worth, relative to the design calculations. Group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order. The Completion Time of 4 hours is intended to restrict operation in this condition because of the potential severity associated with gross violations of group sequence or overlap requirements. The 4 hour Completion Time is based on operating experience which supports the restoration time without unnecessarily challenging unit operation and the low probability of an event occurring simultaneously with the limit out of specification.

ACTIONS (continued)

D.1

Operation in the unacceptable operation region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, the RCS boron concentration must be increased to restore the regulating rod insertion to a value that preserves the SDM and ejected rod worth limits. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted operation region, which restores the minimum SDM and reduces the potential ejected rod worth to within its limit.

D.2.1

The required Completion Time of 2 hours from initial discovery of a regulating rod group in the unacceptable operation region until its restoration to within the restricted operation region shown on the figures in the COLR allows sufficient time for borated water to enter the RCS from the chemical addition and makeup and purification systems, thereby allowing the regulating rods to be withdrawn to the restricted operation region. Operation in the restricted operation region for up to 2 hours is reasonable, based on limiting the potential for an adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

D.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems. Operation for up to 2 hours in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

ACTIONS (continued)

E.1

If the Required Actions and associated Completion Times of Conditions C or D are not met, then the reactor is placed in MODE 3, in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours is acceptable because little rod motion occurs during this period due to fuel burnup. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

Verification of the regulating rod insertion setpoints as specified in the COLR at a Frequency of 12 hours is sufficient to detect regulating rod banks that may be approaching the group insertion setpoints, because little rod motion due to fuel burnup occurs in 12 hours. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and establish the estimated critical position.

REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26 and GDC 28.
2. 10 CFR 50.46.
3. SAR, Chapter 3.
4. SAR, Chapter 14.
5. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

BASES

BACKGROUND

The insertion limits of the APSRs are initial condition assumptions in all safety analyses that are affected by core power distributions. The applicable criterion for these power distribution design requirements are SAR Section 1.4, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on APSR insertion have been established, and all APSR positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is maintained.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that meet the criteria specified in Reference 2. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate (LHR) limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. The APSRs do not insert upon a reactor trip.

This LCO is required to minimize fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or abnormalities. Acceptance criteria for the safety and regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM which assumes the highest worth CONTROL ROD stuck fully withdrawn (GDC 26, Ref. 1).

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result should an accident occur simultaneously with violation of one or more of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates.

Operation at the APSR insertion limits may approach the maximum allowable linear heat generation rate with the allowed QPT present.

In MODES 1 and 2 while critical, the APSR insertion limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODE 2 while subcritical, the APSR insertion limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The setpoints on APSR physical insertion as defined in the COLR must be maintained because they serve the function of controlling the power distribution within an acceptable range.

The fuel cycle design assumes APSR withdrawal at the EFPD burnup window specified in the COLR. Prior to this window, the APSRs are maintained in accordance with operating guidelines provided by reactor engineering during steady state operation. After this window, the APSRs are not allowed to be reinserted for the remainder of the fuel cycle.

APPLICABILITY

The APSR physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the power distribution within the range assumed in the accident analyses. In MODES 1 and 2, the limits on APSR insertion specified by this LCO maintain the axial fuel burnup design conditions assumed in the reload safety evaluation analysis. Applicability in MODES 3, 4, and 5 is not required, because the reactor is subcritical.

ACTIONS

For steady state power operation, a normal position for APSR insertion is specified in the station operating procedures. The APSRs may be positioned as necessary for transient AXIAL POWER IMBALANCE control until the fuel cycle design requires them to be fully withdrawn. (Not all fuel cycles may incorporate APSR withdrawal.) APSR position limits are not imposed for gray APSRs, with two exceptions. If the fuel cycle design incorporates an APSR withdrawal (usually near end of cycle (EOC)), the APSRs may not be maintained in the fully withdrawn position prior to the fuel cycle burnup for the APSR withdrawal. If this occurs, the APSRs must be restored to their normal inserted position. Conversely, after the fuel cycle burnup for the APSR withdrawal occurs, the APSRs may not be reinserted for the remainder of the fuel cycle. These restrictions apply to ensure the axial burnup distribution that accumulates in the fuel will be consistent with the expected (as designed) distribution.

A.1

For verification that the core linear heat rates are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Successful verification that the LHRs are within their limits ensures that operation with the APSRs inserted or withdrawn in violation of the setpoints specified in the COLR do not violate either the ECCS or DNB criteria. The required Completion Time of 2 hours is reasonable to allow the operator to obtain a power distribution map and to verify the LHRs. Repeating SR 3.2.5.1 every 2 hours is reasonable to ensure that continued verification of the LHRs is obtained as core conditions (primarily the regulating rod insertion and induced xenon redistribution) change.

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

A.2

Indefinite operation with the APSRs positioned in violation of the setpoints specified in the COLR is not prudent. Even if LHR monitoring per Required Action A.1 is continued, the abnormal APSR positioning may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may affect the long term fuel depletion pattern. Therefore, operation is allowed for up to 24 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the APSR position out of specification. In addition, it precludes long term depletion with the APSRs in positions that have not been analyzed, thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that the intended burnup distribution is maintained, and allows the operator sufficient time to reposition the APSRs to correct their positions.

ACTIONS (continued)

A.2 (continued)

Because the APSRs are not operated by the automatic control system, manual action by the operator is required to restore the APSRs to the positions specified in the COLR.

B.1

If the Required Action and associated Completion Time are not met, the reactor must be placed in MODE 3, in which this LCO does not apply. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

Fuel cycle designs that allow APSR withdrawal near end of cycle (EOC) do not permit reinsertion of APSRs after the time of withdrawal. Verification that the APSRs are within their insertion setpoints at a 12 hour Frequency is sufficient to ensure that the APSR insertion setpoints are preserved. The 12 hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. The Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

REFERENCES

1. SAR Section 1.4, GDC 10 and GDC 26.

2. 10 CFR 50.46.

3. SAR, Chapter 14.

4. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the linear heat rate (LHR) limits given in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions in the safety analyses related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum LHR so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on AXIAL POWER IMBALANCE are determined analytically by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate the assumptions used in the accident analyses regarding the core power distribution. The AXIAL POWER IMBALANCE setpoints provided in the COLR account for measurement system error and uncertainty.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The LCOs based on power distribution, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," preclude core power distributions that would violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

The regulating rod positions, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result should an accident occur with simultaneous violation of one or more of the LCOs governing the four process variables cited above. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The regulating rod insertion, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are monitored and controlled during power operation to ensure that the power distribution is within the bounds set by the safety analyses. The axial power distribution is maintained primarily by the AXIAL POWER IMBALANCE and the APSR position limits; and the radial power distribution is maintained primarily by the QPT limits. The regulating rod insertion limits affect both the radial and axial power distributions.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account when the reload safety evaluation analysis is performed.

Operation at the AXIAL POWER IMBALANCE limit must be interpreted as operating the core at the maximum allowable LHR assumed as initial conditions for the accident analyses with the allowed QPT present.

AXIAL POWER IMBALANCE satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The AXIAL POWER IMBALANCE envelope contained in the COLR represents the setpoints beyond which the core power distribution could either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during the loss of flow accident with the allowable QPT present and with the APSR positions consistent with the limitations on APSR withdrawal determined by the fuel cycle design and. specified by LCO 3.2.2.

The AXIAL POWER IMBALANCE maximum allowable setpoints (measurement system dependent limits) applicable for the full Incore Detector System, the Minimum Incore Detector System, and the Excore Detector System are provided in the COLR.

APPLICABILITY

In MODE 1, the limits on AXIAL POWER IMBALANCE must be maintained when THERMAL POWER is > 40% RTP to prevent the core power distribution from exceeding the LOCA and loss of flow assumptions used in the accident analyses. Applicability of these limits at ≤ 40% RTP in MODE 1 is not required. This operation is acceptable based on engineering judgment because the combination of AXIAL POWER IMBALANCE with the maximum allowable THERMAL POWER level will not result in LHRs sufficiently large to violate the fuel design limits. In MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor is not generating sufficient THERMAL POWER to produce fuel damage.

ACTIONS

A.1

The AXIAL POWER IMBALANCE operating setpoints that maintain the validity of the assumptions regarding the power distributions in the accident analyses of the LOCA and the loss of flow accident are provided in the COLR. Operation within the AXIAL POWER IMBALANCE setpoints given in the COLR is the acceptable region of operation. Operation in violation of the AXIAL POWER IMBALANCE setpoints given in the COLR is the restricted region of operation.

Operation with AXIAL POWER IMBALANCE in the restricted region shown on the AXIAL POWER IMBALANCE figures in the COLR potentially violates the LOCA LHR limits or the loss of flow accident DNB peaking limits or both. For verification that core local LHRs are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that core local LHRs are within their specified limits ensures that

ACTIONS (continued)

A.1 (continued)

operation with the AXIAL POWER IMBALANCE in the restricted region does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of 2 hours provides reasonable time for the operator to obtain a power distribution map and to determine and verify that the core local LHRs are within their specified limits. The 2 hour Frequency provides reasonable time to ensure that continued verification of the core local LHRs is obtained as core conditions (primarily regulating rod insertion and induced xenon redistribution) change, because little rod motion occurs in 2 hours due to fuel burnup, the potential for xenon redistribution is limited, and the probability of an event occurring in this short time frame is low.

A.2

Indefinite operation with the AXIAL POWER IMBALANCE in the restricted region is not prudent. Even if LHR monitoring per Required Action A.1 is continued, excessive AXIAL POWER IMBALANCE over an extended period of time may cause a potentially adverse xenon redistribution to occur. Therefore, LHR monitoring is only allowed for a maximum of 24 hours. This required Completion Time is reasonable based on the low probability of a limiting event occurring simultaneously with the AXIAL POWER IMBALANCE outside the setpoints of this LCO. In addition, this limited Completion Time precludes long term depletion of the reactor fuel with excessive AXIAL POWER IMBALANCE and gives the operator sufficient time to reposition the APSRs or regulating rods to reduce the AXIAL POWER IMBALANCE because adverse effects of xenon redistribution and fuel depletion are limited.

B.1

If the Required Actions and the associated Completion Times of Condition A are not met, the AXIAL POWER IMBALANCE may exceed its specified limits and the reactor may be operating with a global axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation and may result in an increased linear heat generation rate when the xenon redistributes. Reducing THERMAL POWER to ≤ 40% RTP reduces the maximum LHR to a value that does not exceed the LHR initial condition limits assumed in the accident analyses. The required Completion Time of 4 hours is reasonable based on limiting a potentially adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AXIAL POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints are derived from their corresponding measurement system independent limits by adjusting for both the system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis that defines the required error adjustment to convert the measurement system independent limits to full incore detector system limits assumes that 75% of the detectors in each quadrant are OPERABLE. Detectors located on the core major axes are assumed to contribute one half of their output to each quadrant; detectors in the center assembly are assumed to contribute one quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane, with one plane at the core midplane and one plane in each axial core half;

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

Figure B 3.2.3-1 (Minimum Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Minimum Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

Verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures that the AXIAL POWER IMBALANCE setpoints are not violated and takes into account other information and alarms available in the control room. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or control rod drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

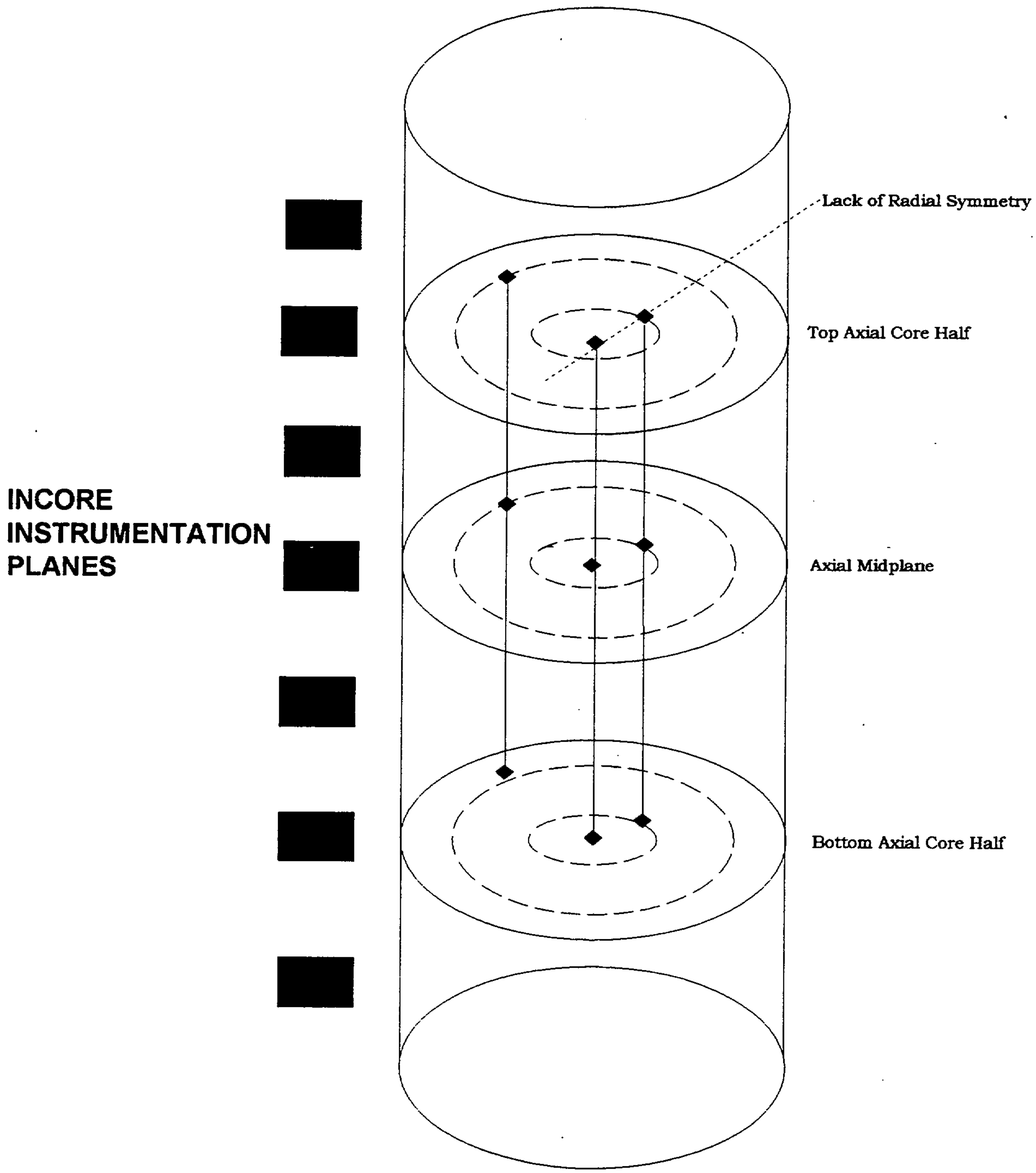
REFERENCES

1. 10 CFR 50.46.

2. 10 CFR 50.36.

Figure B 3.2.3-1

Minimum Incore System for AXIAL POWER IMBALANCE Measurement



B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT (QPT)

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1). Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate (LHR) limits given in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived by Emergency Core Cooling Systems (ECCS) analysis and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow, or other accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions used in the safety analysis related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum LHR so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

BACKGROUND (continued)

The measurement system independent limits on QPT are determined analytically by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate core power distribution assumptions used in the accident analysis. The error adjusted maximum allowable setpoints (measurement system dependent limits) for QPT are specified in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The LCOs based on power distribution (LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4) preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

QPT is one of the process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result if an accident occurs with simultaneous violation of one or more of the LCOs governing the core power distribution. Changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account during the reload safety evaluation analysis. An allowance for QPT is accommodated in the analysis and resultant LCO limits. The increase in peaking taken for QPT is developed from a database of full core power distribution calculations (Ref. 2). The calculations consist of simulations of many power distributions with tilt causing mechanisms (e.g., dropped or misaligned CONTROL RODS, broken APSR fingers fully inserted, misloaded assemblies, and burnup gradients). An increase of < 2% peak power per 1% QPT is supported by the analysis, therefore a value of 2% peak power increase per 1% QPT is used to bound peak power increases due to QPT.

Operation at the AXIAL POWER IMBALANCE or rod insertion limits must be interpreted as operating the core at the maximum allowable LHR for accident initial conditions with the allowed QPT present.

QPT satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The regulating rod insertion setpoints and the AXIAL POWER IMBALANCE boundaries contained in the COLR represent the measurement system dependent limits at which the core power distribution could either exceed the LOCA LHR limits or cause a reduction in DNBR below the safety limit during a loss of flow accident with the allowable QPT present and with an APSR position consistent with the limitations on APSR position determined by the fuel cycle design and specified by LCO 3.2.2.

The allowable setpoints for steady state and maximum setpoints for QPT applicable for the full symmetrical Incore Detector System, Minimum Incore Detector System, and Excore Detector System are provided in the COLR. The setpoints for the three systems are derived by adjustment of the measurement system independent QPT limits also given in the COLR to allow for system observability and instrumentation errors.

APPLICABILITY

In MODE 1, the limits on QPT must be maintained when THERMAL POWER is > 20% RTP to prevent the core power distribution from exceeding the design limits. The minimum power level of 20% RTP is large enough to obtain meaningful QPT indications without compromising safety.

In MODE 2, the combination of QPT with maximum ALLOWABLE THERMAL POWER level does not result in LHRs sufficiently large to violate the fuel design limits, and therefore, applicability in this MODE is not required. Although not specifically addressed in the LCO, QPTs greater than the maximum setpoint specified in the COLR in MODE 1 with THERMAL POWER < 20% RTP are allowed based on engineering judgement.

In MODES 3, 4, 5, and 6, this LCO is not applicable, because the reactor is not generating significant THERMAL POWER and QPT is indeterminate.

ACTIONS

A.1.1

The steady state setpoint specified in the COLR provides an allowance for QPT that may occur during normal operation. A peaking increase to accommodate QPTs up to the steady state setpoint is allowed by the regulating rod insertion limits of LCO 3.2.1 and the AXIAL POWER IMBALANCE limits of LCO 3.2.3.

ACTIONS (continued)

A.1.1 (continued)

Operation with QPT greater than the steady state setpoint specified in the COLR potentially violates the LOCA LHR limits, or loss of flow accident DNB peaking limits, or both. For verification that core local LHRs are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that core local LHRs are within their limits ensures that operation with QPT greater than the steady state setpoint does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of once per 2 hours is a reasonable amount of time to allow the operator to obtain a power distribution map and to verify the core local LHRs. Repeating SR 3.2.5.1 every 2 hours is a reasonable Frequency at which to ensure that continued verification of the core local LHRs is obtained as core conditions that influence QPT change.

A.1.2.1

The safety analysis has shown that a conservative corrective action is to reduce THERMAL POWER by 2% RTP or more from the ALLOWABLE THERMAL POWER for each 1% of QPT in excess of the steady state setpoint. This action limits the local LHR to a value corresponding to the assumed accident initial condition limits. The required Completion Time of 2 hours is reasonable, based on limiting the potential for xenon redistribution, the low probability of an accident occurring, and the steps required to complete the Required Action.

If QPT can be reduced to less than or equal to the steady state setpoint in < 2 hours, the reactor may return to normal operation without undergoing a power reduction. Significant radial xenon redistribution does not occur within this amount of time.

The required Completion Time of 2 hours after the last performance of SR 3.2.5.1 allows reduction of THERMAL POWER in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.1.2.2

Power operation is allowed to continue if THERMAL POWER is reduced in accordance with Required Action A.1.2.1. The same reduction (i.e., 2% RTP or more) is also applicable to the nuclear overpower based on Reactor Coolant System (RCS) flow and AXIAL POWER IMBALANCE trip setpoint, for each 1% of QPT in excess of the steady state limit. This reduction maintains both core protection and thermal margins at the reduced THERMAL POWER level similar to that at RTP. The required Completion Time of 10 hours or 10 hours after the last performance of SR 3.2.5.1 is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with the QPT limits not met, and the number of steps required to complete the Required Action.

ACTIONS (continued)

A.1.2.3

Power operation is allowed to continue if restrictions are imposed on the allowed degree of regulating group insertion. This Required Action requires a reduction in the regulating group insertion setpoints given in the COLR by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state setpoint. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with regulating rod group insertion into the core.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.1.2.4

Power operation is allowed to continue if restrictions are imposed on the allowed Operational Power Imbalance Setpoints given in the COLR. This Required Action results in a reduction in the allowed THERMAL POWER level as a function of AXIAL POWER IMBALANCE by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with the combined affects of operating with an AXIAL POWER IMBALANCE and a QPT.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.2

Although the actions directed by Required Action A.1.2.1 restore thermal margins, if the source of the QPT is not established and corrected, it is prudent to establish increased margins. A required Completion Time of 24 hours to reduce QPT to less than the steady state limit is a reasonable time for investigation and corrective measures.

ACTIONS (continued)

B.1

If the Required Actions and associated Completion Times of Condition A are not met, a further power reduction is required. Power reduction to < 60% of ALLOWABLE THERMAL POWER provides conservative protection from increased peaking due to xenon redistribution. The required Completion Time of 2 hours is reasonable to allow the operator to reduce THERMAL POWER to < 60% of ALLOWABLE THERMAL POWER without challenging unit systems.

B.2

Reduction of the nuclear overpower trip setpoint to ≤ 65.5% of ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of ALLOWABLE THERMAL POWER maintains both core protection and OPERABILITY margin at reduced power similar to that at full power. The required Completion Time of 10 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience.

C.1

If the Required Actions and associated Completion Times of Condition B are not met, then the reactor will continue in power operation with significant QPT. Either the power level has not been reduced to comply with the Required Action or the nuclear overpower trip setpoint has not been reduced within the required Completion Time. To preclude risk of fuel damage in any of these conditions, THERMAL POWER is reduced further. Operation below 20% RTP allows the operator to investigate the cause of the QPT and to correct it. Local LHRs with a large QPT do not violate the fuel design limits at or below 20% RTP. The required Completion Time of 4 hours is acceptable based on limiting the potential increase in local LHRs that could occur due to xenon redistribution with the QPT out of specification.

D.1

QPT in excess of the maximum setpoint specified in the COLR can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% RTP ensures local LHRs do not exceed allowable limits while the cause is being determined and corrected.

The required Completion Time of 4 hours is reasonable to allow the operator to reduce THERMAL POWER to ≤ 20% RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

QPT can be monitored by both the Incore and Excore Detector systems. The QPT setpoints are derived from their corresponding measurement system independent limits by adjustment for system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limit, the limits for the different systems are not identical because of differences in the errors applicable for these systems. For QPT measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Two sets of four detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

Figure B 3.2.4-1 (Minimum Incore Detector System for QPT Measurement) depicts an example of this configuration. The symmetric full Incore Detector System for QPT uses the Incore Detector System as described above and is configured such that at least 75% of the detectors in each core quadrant are OPERABLE.

SR 3.2.4.1

Checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and takes into account other information and alarms available to the operator in the control room. This procedure allows the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Operating experience has confirmed the acceptability of a Surveillance Frequency of 7 days.

Following restoration of the QPT to within the setpoint, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the setpoint at the increased THERMAL POWER level. In case QPT exceeds the setpoint for more than 24 hours (Condition A), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the setpoint again.

REFERENCES

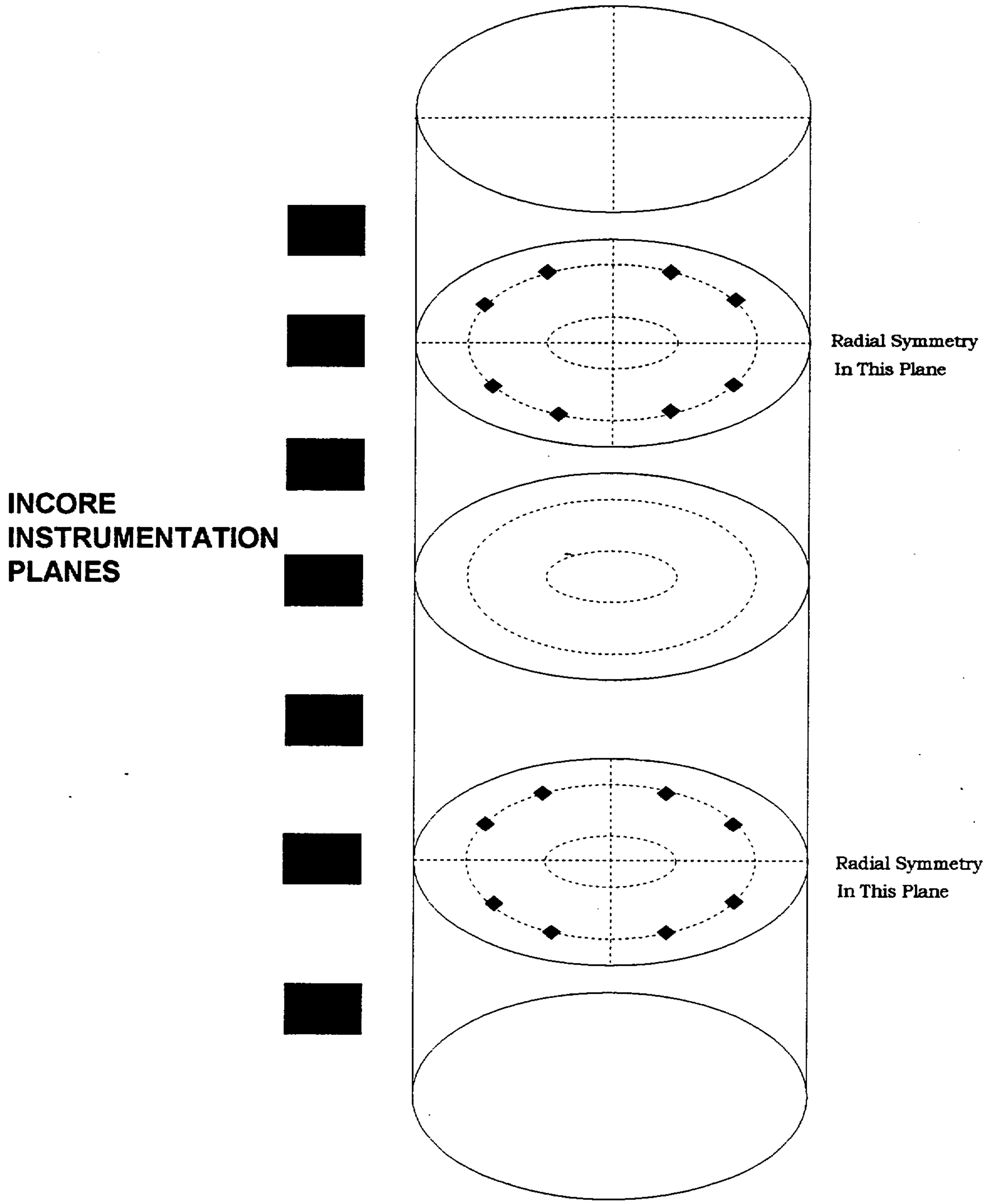
1. 10 CFR 50.46

2. BAW 10122A, "Normal Operating Controls," Rev. 1, May 1984.

3. 10 CFR 50.36

Figure B 3.2.4-1

Minimum Incore System for QUADRANT POWER Tilt Measurement



B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Power Peaking

BASES

BACKGROUND

The purpose of this LCO is to establish limits that constrain the core power distribution within design limits during normal operation, during abnormalities and such that accident initial condition protection criteria are preserved. The accident initial condition criteria are preserved by bounding operation within specified acceptable fuel design limits. This is accomplished by limiting the local linear heat rate (LHR) to three general constraints: 1) the LHR may not exceed a value that results in fuel centerline melt, 2) the LHR may not exceed a value that would result in peak cladding temperatures of greater than 2200°F during a loss of coolant accident (LOCA), and 3) the LHR may not exceed a value that would result in the minimum departure from nucleate boiling ratio (DNBR) dropping below the specified acceptable fuel design limits in the event of the limiting loss of flow transient.

The LOCA-limited LHR is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling Systems (ECCS) analysis. The LOCA-limited LHR is dependent upon core axial location and fuel batch design. The LOCA-limited LHR may be designated as LHR in units kW/ft or as a power peaking factor. When expressed as a power peaking factor, the LOCA-limited LHR is designated as $F_Q(Z)$. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions. Operation within the limits given by the LOCA LHR figure in the COLR prevents power generation rates that would exceed the LOCA-limited LHR limits derived from the analysis of the ECCS.

The LOCA-limited LHR bounds the fuel centerline melt LHR limit. Thus, compliance with the LOCA-limited LHR ensures compliance with the fuel centerline melt LHR.

The DNBR-limited LHR is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss of flow transient. DNBR is defined as the ratio of the heat flux that would cause departure from nucleate boiling (DNB) at a particular core location to the actual heat flux at that core location. The DNBR-limited LHR represents the linear power generation rate along the fuel rod on which the minimum DNBR occurs. Compliance with this LHR value may be accomplished: 1) by correlating the LHR at the limiting location to the critical heat flux (expressed as a LHR) for the limiting location, 2) by correlating the LHR to DNBR or DNB margin for the limiting location, or 3) by correlating the LHR to a power peaking factor (designated as $F^N_{\Delta H}$) for the limiting location.

The relationship between the observable parameters of neutron power, reactor coolant flow, temperature and pressure and the critical heat flux, DNBR or DNB margin is provided through use of a critical heat flux correlation. The critical heat

BACKGROUND (continued)

flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for SL 2.1.1. $F^N_{\Delta H}$ is defined as the ratio of the integral of linear power along the fuel rod on which the minimum DNBR occurs to the average integrated rod power. Operation within the DNBR-limited LHR limit prevents DNB during a postulated loss of forced reactor coolant flow accident.

Measurement of the core core peaking factors using the Incore Detector System to obtain a three dimensional power distribution map provides direct confirmation that LHRs are within their limits and may be used to verify that the core local LHRs remain bounded when one or more normal operating parameters exceed their limits.

APPLICABLE SAFETY ANALYSES

The LOCA-limited LHR limits are determined by the ECCS analysis in order to limit peak cladding temperatures to 2200°F during a LOCA. The maximum acceptable cladding temperature is specified by 10 CFR 50.46 (Ref. 1). Higher cladding temperatures could cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

The DNBR-limited LHR limits provide protection from DNB during a limiting loss of flow transient. Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that core location. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB. The critical heat flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for SL 2.1.1.

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

APPLICABLE SAFETY ANALYSES (continued)

The reload safety evaluation analysis determines limits on global core parameters that characterize the core power distribution. The primary parameters used to monitor and control the core power distribution are the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. These parameters are normally used to monitor and control the core power distribution because their measurements are continuously observable. Limits are placed on these parameters to ensure that the core power peaking factors remain bounded during operation in MODE 1 with THERMAL POWER greater than 20% RTP. Nuclear design model calculational uncertainty, manufacturing tolerances (e.g., the engineering hot channel factor), effects of fuel densification and rod bow, and modeling simplifications (such as treatment of the spacer grid effects) are accommodated as necessary through use of peaking augmentation factors in the reload safety evaluation analysis (Ref. 2).

LHR limitations satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO for power peaking ensures that the core operates within the LHR bounds assumed for the ECCS and thermal hydraulic analyses. Verification that LHR is within the limits of this LCO as specified in the COLR allows continued operation when the Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," LCO 3.2.1, "Regulating Rod Group Insertion Limits," LCO 3.2.2, "APSR Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT," are entered. Conservative THERMAL POWER reductions are required if the limits on LHR are exceeded. Verification that LHR is within the limits is also required during MODE 1 PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 1."

Measurement uncertainties are applied when LHR is determined using the Incore Detector System. The measurement uncertainties applied to the measured values account for uncertainties in observability and instrument string signal processing.

APPLICABILITY

In MODE 1 with THERMAL POWER > 20% RTP, the limits on LHR must be maintained in order to prevent the core power distribution from exceeding the limits assumed in the analyses of the LOCA and loss of forced reactor coolant flow accidents. In MODE 1 with THERMAL POWER ≤ 20% RTP and in MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor has insufficient stored energy in the fuel or energy being transferred to the coolant to require a limit on the distribution of core power.

APPLICABILITY (continued)

The minimum THERMAL POWER level of 20% RTP was chosen based on the ability of the incore detection system to satisfactorily obtain meaningful power distribution data.

ACTIONS

The operator must take care in interpreting the relationship of the LHRs, DNBRs, and power peaking factors to their limits. Limiting values may be expressed as an LHR, DNBR, margin to DNB or as power peaking factors. When expressed as power peaking factors, the value must be adjusted in inverse proportion to the THERMAL POWER level of the core as the power is reduced from RTP. Thus, the allowable peaking factors will increase as THERMAL POWER decreases.

A.1

When the LHR is determined not to be within its specified limit as determined by a three dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the limiting LHR in the core. The Completion Time of 2 hours provides an acceptable time to reduce power in an orderly manner and without allowing the unit to remain in an unacceptable condition for an extended period of time.

B.1

If the Required Action and associated Completion Time for Condition A are not met, then THERMAL POWER operation should be reduced. The reactor is placed in MODE 1 with THERMAL POWER less than or equal to 20% RTP where this LCO does not apply. The required Completion Time of 4 hours is a reasonable amount of time for the operator to reduce THERMAL POWER in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Core power distribution monitoring is performed using the Incore Detector System to obtain a three dimensional power distribution map. Maximum LHR values obtained from this map may then be compared with the limits in the COLR to verify that the limits have not been exceeded. Minimum DNBR values or DNB margins determined from the core power distribution mapping may also be compared to their limits or correlated to LHR values to verify that the limits have not been exceeded. Measurement of the core power distribution in this manner may be used to verify that the measured LHR values remain within their specified limits when one or more

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.5.1 (continued)

of the limits specified by LCO 3.1.4, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is exceeded, or when LCO 3.1.8 is applicable. If the local LHRs remain within their limits when one or more of these parameters exceed their limits, operation at THERMAL POWER may continue because the true initial conditions (the core power distribution) remain within their specified limits.

Because the limits on LHR are preserved when the parameters specified by LCO 3.1.4, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 are within their limits, a Note is provided in the SR to indicate that monitoring core local LHRs is required only when complying with the Required Actions of these LCOs and when LCO 3.1.8 is applicable.

Frequencies for monitoring of the core local LHRs are specified in the Action statements of the individual LCOs. These Frequencies are reasonable based on the low probability of a limiting event occurring simultaneously with LHR exceeding its limit, and they provide sufficient time for the operator to obtain a power distribution map from the Incore Detector System. Indefinite THERMAL POWER operation in a Required Action of LCO 3.1.4, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is permitted, because the core local LHRs assumed in the accident analyses are within analyzed core power distributions and spatial xenon distributions.

REFERENCES

1. 10 CFR 50.46.

2. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2, October 1997.

3. 10 CFR 50.36.