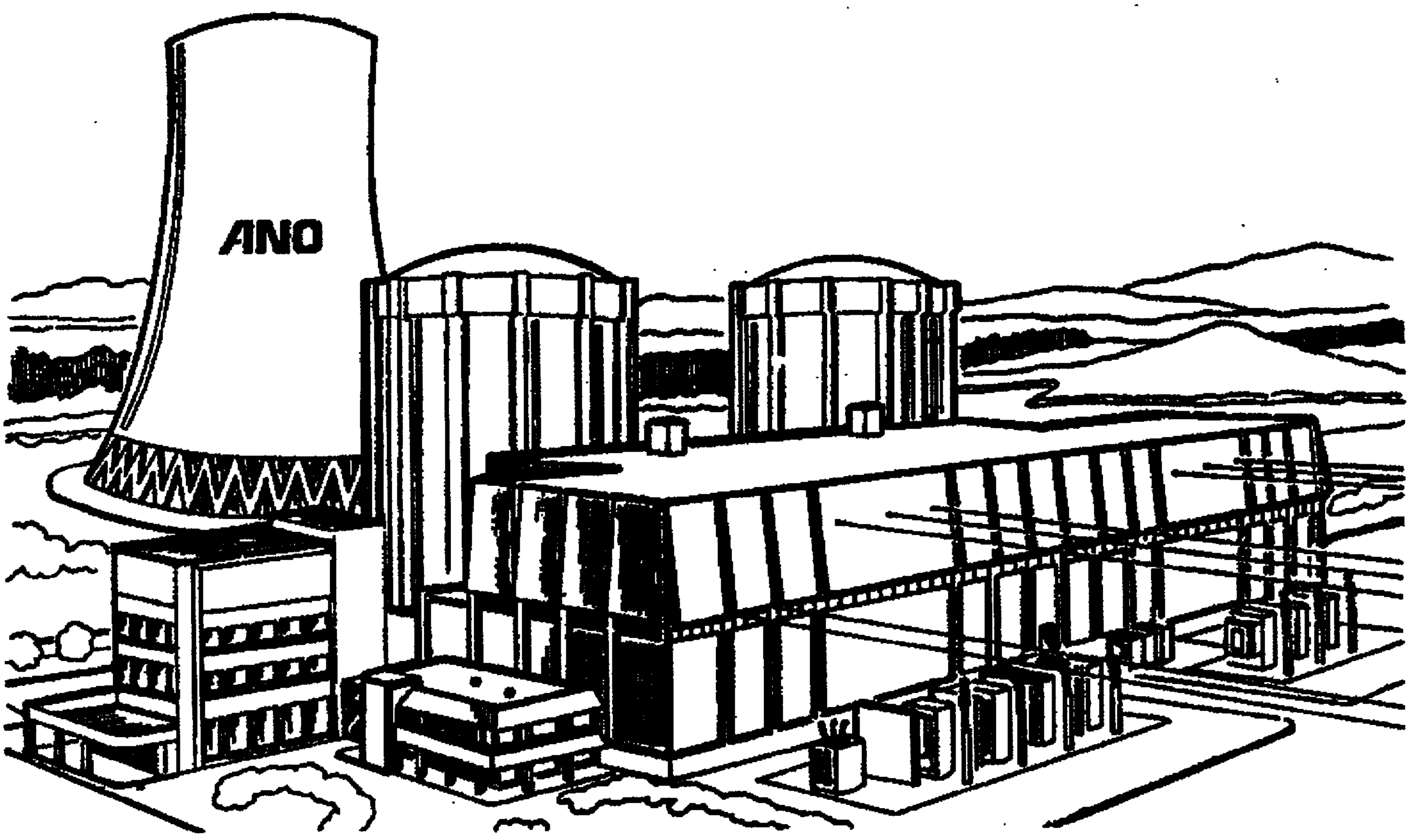
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED
TECHNICAL SPECIFICATIONS
SUBMITTAL



05/01/01 Supplement

Volume 2 of 2

(Sections 3.7 and 3.8)

Entergy

May 1, 2001

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Seven MSSVs shall be OPERABLE on each main steam line.

--NOTE--
During main steam system hydrotesting in MODE 3, one MSSV is required to be OPERABLE on each main steam line with lift setpoints adjusted to allow testing.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Reduce power in accordance with Table 3.7.1-1.	4 hours
	AND	
	A.2 Reduce the nuclear overpower trip setpoint in accordance with Table 3.7.1-1.	36 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
OR	AND	
One or more steam generators with less than two MSSVs OPERABLE.	B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.1.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
Allowable Power Level and RPS Nuclear Overpower Trip
Allowable Value versus OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVS OPERABLE (PER SG)	MAXIMUM ALLOWABLE POWER LEVEL (% RTP)	RPS NUCLEAR OVERPOWER TRIP ALLOWABLE VALUE (% RTP)
6	85.7	89.9
5	71.4	74.9
4	57.1	59.9
3	42.8	44.9
2	28.5	29.9

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. <u>AND</u> 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. <u>AND</u> 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.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	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. AND A.2 Restore QCST to OPERABLE status.	4 hours AND Once per 12 hours thereafter 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance on steam generator for heat removal.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify QCST volume is $\geq$ 32,300 gallons.	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. <u>AND</u> B.2 Be in MODE 5.	6 hours 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. <u>OR</u> Both PRVS trains inoperable for reasons other than Condition B.	C.1 Be in MODE 3. <u>AND</u> 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 $\geq$ 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 $\geq$ 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.

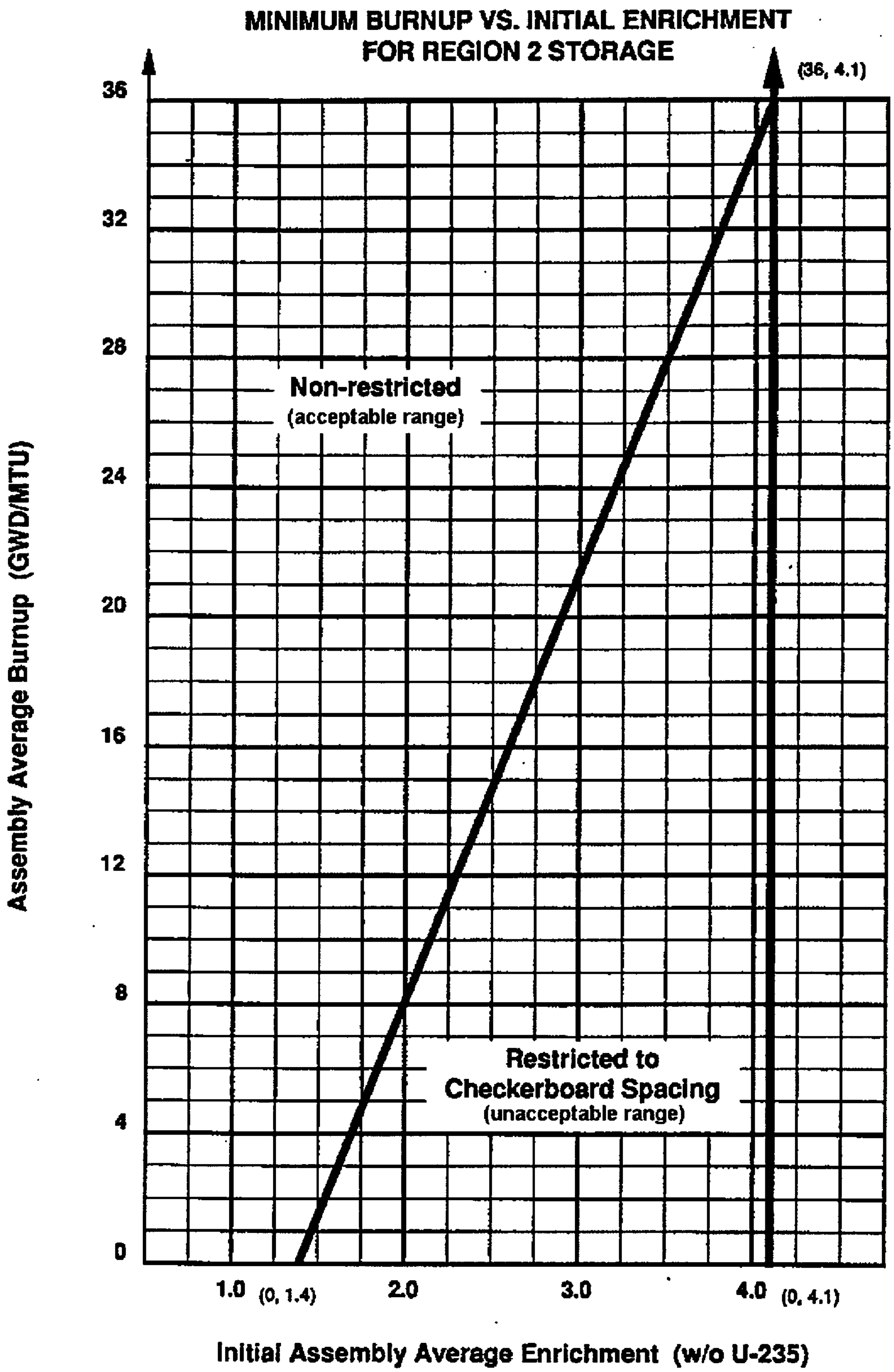
ACTIONS

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 (page 1 of 1)
Burnup versus Enrichment Curve for
Spent Fuel Storage Racks



B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Eight MSSVs are located on each main steam header, outside the reactor building, upstream of the main steam isolation valves, as described in the SAR, Section 10.3 (Ref. 1). The MSSV capacity is adequate to meet the requirements of the ASME Code, Section III (Ref. 2). The total capacity of 14 MSSVs is greater than the total steam flow at 102% RTP. The MSSV design includes staggered setpoints (Ref. 1) so that only the needed number of valves will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open the valves.

APPLICABLE SAFETY ANALYSES

The design basis of the MSSVs (Ref. 2) is to limit secondary system pressure to $\leq$ 110% of design pressure when passing 102% of design steam flow (100% plus 2% heat balance error). The MSSVs ensure that the design basis requirements are met for any abnormality or accident considered in the SAR.

The events that may assume use of the MSSVs are those characterized as decreased heat removal events. MSSV use may be assumed during mitigation of the following events:

a. Loss of Load (SAR, Chapter 14 (Ref. 3));

b. Steam generator tube rupture; and

c. Small break loss of coolant (Ref. 3).

The full power turbine trip coincident with a loss of condensate heat sink establishes the required MSSV relief capacity (Ref. 4).

In MODES 1 and 2, the MSSVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODE 3, the MSSVs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The MSSVs are provided to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires fourteen MSSVs (seven on each main steam line) to be OPERABLE to ensure compliance with the ASME Code following DBAs initiated at full power. Operation with less than the required complement of MSSVs requires a limitation on unit THERMAL POWER and adjustment of the Reactor Protection System (RPS) nuclear overpower trip setpoint. The minimum number of OPERABLE MSSVs per steam generator for various power levels and the associated maximum allowable nuclear overpower trip setpoint are identified in Table 3.7.1-1. This effectively limits the Main Steam System steam flow while the MSSV relieving capacity is reduced due to valve inoperability. To be OPERABLE, lift setpoints must remain within limits, according to SR 3.7.1.1.

The safety function of the MSSVs is to open, relieve steam generator overpressure, and reseat when pressure has been reduced.

OPERABILITY of the MSSVs requires periodic surveillance testing in accordance with the Inservice Testing Program.

With all MSSVs OPERABLE, at least one MSSV per steam generator is set at 1050 psig nominal, while the remaining MSSVs per steam generator are set at varied pressures up to and including 1100 psig nominal. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform the design safety function.

The LCO is modified by a Note that allows all but one MSSV on each main steam header to be gagged and the setpoints for the two (one on each header) OPERABLE MSSVs to be reset for the duration of hydrotesting in MODE 3. This is necessary to allow the hydrotest pressure to be attained.

APPLICABILITY

In MODES 1, 2, and 3, the MSSVs are required to be OPERABLE to prevent overpressurization of the main steam system.

In MODES 4 and 5, there is no credible transient requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized. There is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets ASME Code requirements for the power level. Operation may continue, provided the ALLOWABLE THERMAL POWER and RPS nuclear overpower trip setpoint are reduced as required by Table 3.7.1-1. These values are based on the following formulas:

$$RP = \frac{Y}{Z} \times 100\%$$

and

$$SP = \frac{Y}{Z} \times W$$

where:

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1, "Reactor Protection System (RPS)";

Y = Total OPERABLE MSSV relieving capacity per steam generator based on a summation of individual OPERABLE MSSV relief capacities per steam generator (the available capacity of each MSSV is 801,428 lbm/hour);

Z = Required relieving capacity per steam generator of 5,610,000 lbm/hour;

RP = Reduced power requirement (not to exceed RTP); and

SP = Nuclear overpower trip setpoint (not to exceed W).

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, on operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

With one or more steam generators with less than two MSSVs OPERABLE, or if the Required Actions and associated Completion Times are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of MSSV lift setpoints in accordance with the Inservice Testing Program. The safety and relief valve tests are performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6) and include the following for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference 6 provides the activities and frequencies necessary to satisfy the requirements and allows an as-found $\pm$ 3% setpoint tolerance. Although not required by the IST Program, the valves are reset to $\pm$ 1% during the Surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. SAR, Section 10.3.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. SAR, Chapter 14.

4. Framatome Document 86-1266156-00, "ANO-1 Overpressure Protection," dated October 31, 1997.

5. 10 CFR 50.36.

6. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a main steam line break. MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside of, but close to, the reactor building. The MSIVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Turbine Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam line isolation (MSLI) signal as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." The EFIC System is designed to prevent the simultaneous blowdown of both steam generators. The MSIVs may also be actuated manually.

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the analysis for the steam line break (SLB), as discussed in the SAR, Section 14.2 (Ref. 1). The EFIC System design precludes the blowdown of more than one steam generator, assuming a single active component failure as discussed in the SAR, Section 7.1.4 (Ref. 2).

The SLB outside the reactor building upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The SLB at full power is the limiting case for a post trip return to power. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown.

The MSIVs serve only a closing safety function in the event of an SLB and remain open during power operation.

In MODES 1 and 2, the MSIVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 3, the MSIVs satisfy Criterion 4 of 10 CFR 50.36.

LCO

This LCO requires that the MSIV in each steam line be OPERABLE. For an MSIV to be considered OPERABLE, the isolation time must be within limits and the MSIV must close on an isolation actuation signal when required.

This LCO provides assurance that the MSIVs will perform their design safety function to isolate an SLB.

APPLICABILITY

The MSIVs must be OPERABLE to provide isolation of potential main steam line breaks in MODES 1, 2, and 3, when there is significant mass and energy in the RCS and steam generators.

In MODE 4, the steam generator energy is low. Therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators are depressurized and the MSIVs are not required for isolation of potential main steam line breaks.

ACTIONS

A.1

With one or more MSIVs inoperable in MODE 1 or 2, action must be taken to restore the component to OPERABLE status within 24 hours. Some repairs can be made to the MSIV with the unit hot. The 24 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. Although not credited, the turbine throttle valves may be available to provide isolation for some postulated accidents.

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in MODE 3 within the next 12 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 3.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODE 3, the inoperable MSIV(s) may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the Required Actions and associated Completion Times of Condition C are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 4 within 24 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from MODE 3 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is as specified in the Inservice Testing Program. The MSIV isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

The Frequency for this SR is in accordance with the Inservice Testing Program.

This test is normally conducted in MODE 3, with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The Frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.2.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

REFERENCES

1. SAR, Section 14.2.
2. SAR, Section 7.1.4.
3. 10 CFR 50.36.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

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

BASES

BACKGROUND

The main feedwater isolation valves (MFIVs), Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves isolate main feedwater (MFW) flow to the secondary side of the steam generators. Closing the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside the reactor building and reducing the cooldown effects for SLBs.

The MFIVs close on receipt of a main steam line isolation (MSLI) signal as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." EFIC maintains the Low Load Feedwater Control Valves and Startup Feedwater Control Valves closed by sending a signal to the Rapid Feedwater Reduction (RFR) circuit of the Integrated Control System (ICS). The Main Feedwater Block Valves are independently closed by a signal from the Reactor Protection System (RPS) upon a reactor trip. The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves can also be closed manually.

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves is established by the analysis for the SLB as discussed in SAR Section 14.2.2.1 (Ref. 1).

Failure of an MFIV, and an associated Main Feedwater Block Valve, Low Load Feedwater Control Valve or Startup Feedwater Control Valve to close following an SLB, can result in additional mass being delivered to the steam generators, contributing to cooldown.

In MODES 1 and 2, the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODE 3, the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves satisfy Criterion 4 of 10 CFR 50.36.

With the exception of the MFIVs, the valves are non-Q and powered from non-vital sources. This is acceptable when crediting feedwater isolation during a SLB since off-site power is assumed to remain available during this event.

LCO

This LCO ensures that the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves will isolate MFW flow to the steam generators following a main steam line break.

All MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are required to be OPERABLE. For an MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve or Startup Feedwater Control Valve to be considered OPERABLE, the isolation times must be within limits and the valve must close on an isolation actuation signal when required.

Failure to meet the LCO requirements can result in a more severe cooldown transient and in additional mass and energy being released to the reactor building following an SLB inside the reactor building.

APPLICABILITY

The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves must be OPERABLE in MODES 1, 2, and 3 to ensure that, in the event of an SLB, the amount of feedwater provided to the affected steam generator is limited. Their closure terminates normal feedwater flow to limit the overcooling transient and to limit the amount of energy that could be added to the reactor building in the case of a secondary system pipe break inside the reactor building.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable to allow repairs and, if unsuccessful, to isolate the flow path.

Inoperable MFIVs that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analyis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one Main Feedwater Block Valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE associated MFIV and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable Main Feedwater Block Valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

With one Low Load Feedwater Control Valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE associated MFIV and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable Low Load Feedwater Control Valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1 and D.2

With one Startup Feedwater Control Valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE associated MFIV and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable Startup Feedwater Control Valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

E.1

With two inoperable valves in the same flow path there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure to two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path and as such is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. The 8 hour Completion Time is reasonable to isolate the affected flow path.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve is as specified in the Inservice Testing Program.

The MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage. The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are not tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

This SR verifies that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.3.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when the steam generator pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

REFERENCES

1. SAR, Section 14.2.2.1.

2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, abnormalities, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational Leakage") of primary coolant at the limit of 3.5 μCi/gm (LCO 3.4.12, "RCS Specific Activity"). The thyroid dose conversion factors used in the calculation of DOSE EQUIVALENT I-131 are those identified in Section 1.1, "Definitions."

Operating a unit at the allowable limits could result in a 2 hour exclusion area boundary (EAB) exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside the reactor building and a loss of load incident were considered (Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975 (Ref. 2)).

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released (Ref. 2).

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for LCO 3.4.13 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for LCO 3.4.13. For the less probable accident of a steam line break, the assumption is made that a loss of 10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident (Ref. 2).

In MODES 1 and 2, secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, secondary specific activity limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ 0.17 μCi/gm DOSE EQUIVALENT I-131 maintains the radiological consequences of a Design Basis Accident (DBA) significantly less than the Reference 1 guideline doses.

Monitoring the specific activity of the secondary coolant ensures that, when secondary specific activity limits are exceeded, appropriate actions are taken, in a timely manner, to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are at low pressure and primary to secondary LEAKAGE is minimal. Therefore, secondary specific activity is not a concern.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis assumptions. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the analysis assumptions are met. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.

2. Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975.

3. 10 CFR 50.36

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The EFW pumps take suction from the safety related condensate storage tank (QCST) (LCO 3.7.6, "Q Condensate Storage Tank (QCST)"), and pump to the steam generator secondary side through the EFW nozzles. The core decay heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)"), or atmospheric dump valves (ADVs). If the main condenser is available, steam may be released via the turbine bypass valves.

The EFW System includes one turbine driven EFW pump, and one safety grade motor driven EFW pump. Thus, diversity in motive power sources is provided for the EFW System. The turbine driven EFW pump receives steam from either of the two main steam headers, upstream of the main steam isolation valves (MSIVs).

The EFW System supplies a common header capable of feeding either or both steam generators. Either pump is sufficient to remove decay heat and cool the unit to decay heat removal (DHR) entry conditions. The EFW System initially receives a supply of water from the QCST. The assured safety grade source of water is supplied by the Service Water System (SWS). Valves on the supply piping are manually opened to transfer the water supply from the QCST to the SWS. Water can be supplied from other sources by manually aligning nonsafety grade condensate storage tanks to the EFW pump suction.

The EFW System is capable of supplying feedwater to the steam generators, if required, during normal unit startup and shutdown evolutions, and during hot standby conditions. However, EFW does not provide a normal source of feedwater during these conditions. The normal supplement to the main feedwater system under these conditions is provided by the auxiliary feedwater system.

The EFW actuates automatically (e.g., on loss of main feedwater pumps, low steam generator level, low steam generator pressure, or loss of four reactor coolant pumps) as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation."

The EFW System is discussed in the SAR, Sections 7.1.4 and 10.4.8 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The EFW System is sized to prevent exceeding 110% RCS design pressure for a specified loss of feedwater scenario (Ref. 3).

The design basis of the EFW System is to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure.

The EFW System design is such that it can perform its function with only one EFW train available.

In MODES 1 and 2, the EFW System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODE 3 and MODE 4 when steam generator(s) are relied upon for heat removal, the EFW System satisfies Criterion 4 of 10 CFR 50.36.

LCO

This LCO provides assurance that the EFW System will perform its design function to mitigate the consequences of events that could result in overpressurization of the reactor coolant pressure boundary. Two independent trains are required to be OPERABLE to ensure the availability of residual heat removal capability.

For both EFW trains to be considered OPERABLE, the components and flow paths are required to be capable of providing EFW flow to both steam generators. This requires that the turbine driven EFW pump be OPERABLE with two steam supplies (one from each of the main steam lines upstream of the MSIVs) and capable of supplying EFW flow to the steam generators. The safety grade motor driven EFW pump is also required to be OPERABLE and capable of supplying EFW flow to the steam generators. The piping, valves, instrumentation, and controls in the required flow paths must also be OPERABLE. The primary and secondary sources of water to the EFW System are required to be OPERABLE. The associated flow paths from the EFW System primary and secondary sources of water to both EFW pumps also are required to be OPERABLE.

The LCO is modified by a Note indicating that only one EFW train, which includes the motor driven EFW pump, is required in MODE 4. This is because of reduced heat removal requirement, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE in order to function in the event that the main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

In MODE 4, the EFW System must be OPERABLE when the steam generators are relied upon for decay heat removal since EFW is the safety related source of feedwater to the steam generators. In MODE 4, the steam generators are normally used for heat removal until the DHR System is in operation.

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

ACTIONS

A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, or if the turbine driven EFW pump is inoperable in MODE 3 immediately following refueling, action must be taken to restore the steam supply to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of a steam supply to the turbine driven EFW pump, the 7 day Completion Time is reasonable since there is a redundant steam line for the turbine driven pump.

b. For the inoperability of the turbine driven EFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven EFW pump while in MODE 3 immediately following a refueling, the 7 day Completion Time is reasonable due to the availability of the redundant OPERABLE motor driven EFW pump; and due to the low probability of an event requiring the use of the turbine driven EFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required EFW components to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

Condition A is modified by a Note which limits the applicability of the Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows one EFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

When one of the required EFW trains (pump or flow path) is inoperable, action must be taken to restore the train to OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven EFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of an event requiring EFW occurring during this time period. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required EFW components to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

With the Required Action and associated Completion Time of Condition A or B not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW train is restored to OPERABLE status.

With both EFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

E.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. Correct alignment for automatic valves may be other than the post-accident position provided the valve is otherwise OPERABLE. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the established acceptance criteria during the cycle. Flow and differential head are indicators of pump performance required by Section XI of the ASME Code (Ref. 5). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 5) satisfies this requirement.

This SR is modified by a Note indicating that the SR may be deferred until suitable test conditions are established. This deferral is required because there may be insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an Emergency Feedwater Initiation and Control (EFIC) System signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Each automatic valve is also verified to be capable of manual operation by over-riding the actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on operating experience and design reliability of the equipment.

This SR is modified by a Note which states that the SR is not required to be met in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.4

This SR verifies that each EFW pump starts in the event of any accident or transient that generates an EFIC signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This SR is modified by a Note which states that the SR is not required to be met in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.5

This SR ensures that the EFW system is properly aligned by verifying the position of manual valves in the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable in view of other administrative controls, such as SR 3.7.5.1, to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of manual valves has occurred. This SR ensures that the flow path from the QCST to the steam generator is properly aligned.

SR 3.7.5.6

This SR ensures that the EFW flowpath to each steam generator is open and that water reaches the steam generators from the EFW System. This test is performed during shutdown to minimize thermal cycles to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater. The motor-driven EFW pump is specified because of its availability at the low steam generator pressure conditions that exist in the shutdown condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. SAR, Section 7.1.4.

2. SAR, Section 10.4.8.

3. NRC Letter dated January 12, 1981, (1CNA018103).

4. 10 CFR 50.36.

5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.6 Q Condensate Storage Tank (QCST)

BASES

BACKGROUND

The condensate storage tank (QCST) provides a source of demineralized water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The QCST provides the preferred source of water to the Emergency Feedwater (EFW) System (LCO 3.7.5, "Emergency Feedwater (EFW) System").

Because the QCST is the normally aligned source to EFW, it is designed to withstand earthquakes and other natural phenomena, and a portion is protected from missiles that might be generated by natural phenomena. The QCST is designed as Seismic Category I to ensure availability of the initial EFW supply. Feedwater is also available from alternate sources.

A description of the QCST is found in the SAR, Section 10.4.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The QCST provides the initial source of cooling water to remove decay heat and cool down the unit following any event with a loss of normal feedwater.

A portion of the QCST (T-41B) is protected from tornado generated missiles. The protected volume is sufficient to provide a thirty minute supply of water which is adequate to allow manual operator action, if required, to transfer suction of the EFW pumps to the Service Water System (SWS).

The QCST satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The OPERABILITY of the QCST with the minimum required water volume ensures that sufficient water is available to support EFW operation on both units for at least 30 minutes. This provides adequate time for the operators to manually switch the EFW suction alignment to the Service Water System (SWS), if required. The SWS provides the assured long-term source of cooling water. The required volume considers that the EFW suctions of both units may be aligned to the QCST simultaneously.

The required minimum usable volume includes an allowance for losses due to Unit 2 recirculation line flow. The required volume of 32,300 gallons is equivalent to a tank level of 3 feet 10 inches. This parameter value does not include allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

The tank has sufficient capacity to support more than four hours of cooling in MODE 3 or MODE 4 conditions for both units. This capability is not considered to be a safety related design function and is not controlled by the Technical Specifications.

APPLICABILITY

In MODES 1, 2, 3, and 4 when a steam generator is being relied upon for heat removal, the QCST is required to be OPERABLE.

In MODE 4 when a steam generator is not being relied upon for heat removal, and in MODES 5 and 6, the QCST is not required because the EFW System is not required.

ACTIONS

A.1 and A.2

As an alternative to unit shutdown, the OPERABILITY of the backup water supply should be verified within 4 hours and once every 12 hours thereafter. The OPERABILITY of the backup feedwater supply must include verification, by administrative means, of the OPERABILITY of the flow paths from the backup supply to the EFW pumps and availability of the required volume of water in the backup supply. The QCST must be restored to OPERABLE status within 7 days because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period, requiring the use of water from the QCST.

B.1 and B.2

If the Required Action and associated Completion Times are not met, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generators for heat removal, within 24 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the QCST contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the QCST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in QCST levels.

REFERENCES

1. SAR, Section 10.4.8.

2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.7 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a transient or Design Basis Accident (DBA). During normal operation and normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related portion is covered by this LCO.

The SWS consists of two independent but interconnected, 100% capacity safety related cooling water loops. Three 100% capacity pumps are provided to supply the two loops. Each loop consists of a pump, piping, valving, sluice gates and instrumentation. The pumps, sluice gates and valves are remote manually aligned. In the unlikely event of a loss of coolant accident (LOCA) essential valves are aligned to their post accident positions upon receipt of an engineered safeguards actuation signal. The SWS provides cooling directly to required plant equipment. The system is also the assured safety related source of water to the emergency feedwater pumps, and can also provide a source of makeup water to the emergency cooling pond, and to the spent fuel pool.

The requirements of the SWS for cooling water are more severe during normal operation (at full power) than under accident conditions. Normal operation requires at least two of the three service water pumps, and the pumps in operation are periodically rotated. Normal operation also includes the addition of a biocide during the reactor building emergency cooler surveillance, when the water temperature is between 60°F and 80°F, to prevent biological fouling of the coolers. This water temperature range provides conditions under which Asian clams can spawn and produce larvae which could pass through SWS strainers.

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the SAR, Section 9.3 (Ref. 1). The principal safety related function of the SWS is the transfer of heat from the reactor and safety related components to the heat sink.

APPLICABLE SAFETY ANALYSES

The primary safety function of the SWS is for one SWS loop, in conjunction with the Low Pressure Injection System and the Reactor Building Cooling System, (reactor building spray, reactor building air coolers, or a combination) to remove core decay heat following a design basis LOCA, as discussed in the SAR, Sections 6.2 and 6.3 (Refs. 2 and 3, respectively).

The SWS is designed to perform its function with a single failure of any active component, assuming loss of offsite power.

The SWS also cools the unit from Decay Heat Removal (DHR) System entry conditions to MODE 5 during normal and post accident operation, as discussed in the SAR, Section 9.5 (Ref. 4). The time required for this evolution is a function of the number of DHR loops that are operating.

The SWS is also required to transfer heat from the diesel generators (DGs).

In MODES 1 and 2, the SWS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 3 and 4, the SWS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two SWS loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

For an SWS loop to be considered OPERABLE, it must have:

a. One OPERABLE pump; and

b. The associated piping, valves, sluice gates, and instrumentation and controls required to perform the safety related function OPERABLE.

In addition to the requirements above, for both SWS loops to be considered OPERABLE the required SW pumps must be powered from independent essential buses, to provide redundant and independent flow paths.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS. Therefore, the SWS is required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports. Although the systems it supports may be required to be OPERABLE, the SWS is not required to meet the same OPERABILITY requirements in MODES 5 and 6 as it must in MODES 1, 2, 3, and 4. The definition of OPERABILITY embodies the principle that a system can perform its function(s) only if all necessary support systems are capable of performing their related support functions. If the supported system is capable of performing its safety function without reliance on the SWS, then the SWS is not required to be OPERABLE.

Similarly, operation with the SWS in a less than fully qualified state is acceptable provided an assessment has been performed to determine that the supported system remains capable of performing its safety function.

ACTIONS

A.1

If one SWS loop is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS loop is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS loop could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable SWS loop results in an inoperable DG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable SWS loop results in an inoperable DHR loop. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the Required Action and associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on the existence of procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of components or systems supported by the SWS does not affect the OPERABILITY of the SWS . However, such isolation may render those components inoperable.

SR 3.7.7.2

The SR verifies proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR requires verification that the normally operating SWS pumps (A and C) automatically restart following restoration of power to the respective bus. In addition, the B SWS pump, normally in the standby condition, must be verified to start to support each SWS train for which it is expected to be aligned upon associated ES actuation (with time delay) with simulated failure of the normally operating pump for that train.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at an 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. SAR, Section 9.3.
2. SAR, Section 6.2.
3. SAR, Section 6.3.

4. SAR, Section 9.5.

5. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.8 Emergency Cooling Pond (ECP)

BASES

BACKGROUND

The ECP provides a shared heat sink for removing operating heat from safety related components if the heat sink provided by the Dardanelle Reservoir is unavailable. This is done utilizing the Service Water System (SWS).

The ECP is a portion of the complex of water sources which fulfill the ultimate heat sink requirements for ANO. This complex includes the necessary retaining structures and the piping connecting the sources with, but not including, the SWS intake structure, as discussed in the SAR, Section 9.3 (Ref. 1). The principal function of the ECP is dissipation of residual heat after a reactor shutdown.

The basic performance requirements are that a 30 day supply of water be available for both units, and that the design basis temperatures of safety related equipment not be exceeded. Additional information on the design and operation of the system can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The ECP is the sink for heat removal from the reactor core following an abnormality in which the unit is cooled down and placed on decay heat removal following a loss of the Dardanelle Reservoir inventory which would be considered a single failure.

The operating limits are based on conservative heat transfer analyses for the worst case initial conditions that could be present considering a Unit 2 Design Basis Accident concurrent with a normal shutdown of Unit 1 and a loss of the Dardanelle Reservoir water inventory. Reference 1 provides the details of the assumptions used in the analysis. The minimum ECP requirements take into account: water loss from evaporation due to heat load and climatological conditions, fire pump usage, ECP bottom irregularities, suction pipe level at the ECP, and operator action in transferring the SWS from the Dardanelle Reservoir. Operator action is credited in the inventory analysis during the transfer of the SWS to the ECP. Specifically, pump returns are transferred to the ECP shortly after the Dardanelle Reservoir loss of inventory event begins and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the ECP, lake water is pumped into the ECP, increasing level. This additional water is required, along with that maintained in the ECP, to ensure a 64.5 inch depth, which corresponds to a 30 day supply of cooling water. The ECP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water.

The ECP satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The ECP is a backup system that is required to be OPERABLE to support the SWS. To be considered OPERABLE, the ECP must contain a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate for at least 30 days following the design basis event without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the ECP initial temperature should not exceed 100°F, and the volume of water should not fall below 70 acre-feet during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECP is a backup system that is required to support the OPERABILITY of the equipment serviced by the SWS and is required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ECP are determined by the systems it supports. Although the systems it supports may be required to be OPERABLE, the ECP is not required to meet the same OPERABILITY requirements in MODES 5 and 6 as it must in MODES 1, 2, 3, and 4. The definition of OPERABILITY embodies the principle that a system can perform its function(s) only if all necessary support systems are capable of performing their related support functions. If the supported system is capable of performing its safety function without reliance on the ECP, then the ECP is not required to be OPERABLE. Similarly, operation with the ECP in a less than fully qualified state is acceptable provided an assessment has been performed to determine that the supported system remains capable of performing its safety function. It is important to recognize that single failure criteria is not applicable in MODES 5 and 6. Therefore, the availability of Lake Dardanelle as a heat sink during periods of ECP unavailability may be acceptable provided the probability of a loss of lake and the time to respond to a loss of lake event are considered when planning ECP unavailability periods.

ACTIONS

A.1 and A.2

If the ECP is inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR (together with SR 3.7.8.3 and SR 3.7.8.4) verifies that adequate long term (30 days) cooling inventory is available. The level specified also ensures NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the ECP level during the applicable MODES. This SR verifies that the ECP indicated water level is ≥ 5 ft.

SR 3.7.8.2

This SR provides assurance that the heat sink for the SWS can dissipate the maximum accident or normal heat loads for 30 days following the design basis event. The temperature, measured at the point of discharge from the ECP, is considered a conservative average of total ECP conditions since solar gain, wind speed, and thermal current effects throughout the ECP will essentially be at equilibrium conditions under initial stagnant conditions. The 24 hour Frequency is based on operating experience related to the trending of the ECP temperature during the applicable MODES. This SR verifies that the ECP average water temperature at the point of discharge from the ECP (i.e., SWS suction) is ≤ 100°F.

This SR is modified by a Note indicating that the temperature monitoring is required to be performed only during the summer months (i.e., June 1 to September 30). During other periods of the year, the ECP temperature will not have the potential to reach the temperature limit.

SR 3.7.8.3

This SR (together with SR 3.7.8.1 and 3.7.8.4) verifies that adequate inventory exists to support long term (30 days) cooling. Soundings are performed to ensure the water volume is within limits and that the indicated water level is indicative of an equivalent water volume for accident mitigation. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

SR 3.7.8.4

This SR (together with SR 3.7.8.1 and 3.7.8.3) verifies that adequate inventory exists to support long term (30 days) cooling. Visual inspections of the loose stone (riprap) placed on the banks of the ECP and of the concrete slab spillway are performed to ensure any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation is performed of any apparent changes in visual appearance or other abnormal degradation to determine OPERABILITY. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

REFERENCES

1. SAR, Section 9.3.

2. Regulatory Guide 1.27, Rev. 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974.

3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS is a shared system which provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CREVS consists of two independent, redundant, fan and filter assemblies. Each fan circulates control room air through a filter train consisting of a roughing filter, a high efficiency particulate air (HEPA) filter, and a charcoal adsorber. For control room pressurization, each train provides additional outside air filtered through a four inch bed of charcoal adsorber.

The CREVS is an emergency system. Upon receipt of a unit specific high radiation signal, the control room envelope is isolated, the associated unit's normal control room ventilation system is shutdown, and the associated unit's CREVS is started. The control room envelope is maintained sufficiently leak tight to minimize unfiltered air inleakage. The CREVS operation is discussed in the SAR, Section 9.7 (Ref. 1).

The CREVS is designed to maintain the control room for 30 days of continuous occupancy after a Design Basis Accident (DBA), without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The shared CREVS components are arranged in two safety related ventilation trains, which ensure an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators for the design basis loss of coolant accident fission product release and for a fuel handling accident.

The worst case single active failure of a CREVS component, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

In MODES 1 and 2, and during the movement of irradiated fuel assemblies, the CREVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the CREVS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two CREVS trains are required to be OPERABLE to ensure that at least one is available if a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

For a CREVS train to be considered OPERABLE, the CREVS train must include the associated:

a. OPERABLE fan capable of being powered from both a normal and an OPERABLE emergency power source. (Note: Because this is a shared system, and may be powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREVS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.9 must be applied for inoperable CREVS train(s).);

b. OPERABLE HEPA filter and charcoal adsorber; and

c. OPERABLE ductwork and dampers sufficient to maintain air circulation and provide adequate makeup air flow.

In addition, the control room envelope, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis.

The LCO is modified by two Notes. Note 1 allows the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated. Note 2 requires that one CREVS train be capable of automatic actuation. The other train may be started manually, on failure of the first train.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the control room will remain habitable during and following a DBA.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with a release due to a fuel handling accident.

ACTIONS

A.1

With one CREVS train inoperable due to other than the loss of capability for automatic actuation of one fan or one or more isolation dampers in one CREVS train, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room radiation protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1 and B.2

If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactivity, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possible repair, and test most problems with the control room boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4 if the inoperable CREVS train or control room boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

During movement of irradiated fuel assemblies, if the Required Action and associated Completion Time of Condition A are not met, the OPERABLE CREVS train must immediately be placed in the emergency recirculation mode. This action ensures that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend movement of irradiated fuel assemblies since this is an activity that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude movement of fuel to a safe position.

E.1

During movement of irradiated fuel assemblies, when two CREVS trains are inoperable, action must be taken immediately to suspend movement of irradiated fuel assemblies since this is an activity that could release radioactivity that could enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude movement of fuel to a safe position.

F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable control room boundary (i.e., Condition B), the CREVS may not be capable of performing the intended function and a loss of safety function has occurred. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks this system. This test is conducted on alternating trains semi-monthly by initiating flow through the HEPA filters and charcoal adsorbers. The CREVS is designed without heaters and need only be operated for $\geq$ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and two train redundancy available.

SR 3.7.9.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.9.3

This SR verifies that the CREVS automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks on an actual or simulated actuation signal. The Frequency of 18 months is consistent with the guidance provided in Regulatory Guide 1.52 (Ref. 3).

SR 3.7.9.4

This SR verifies the ability of the CREVS to provide outside air at a flow rate of approximately 333 cfm ±10%. Many factors must be taken into account to determine the overall expected dose consequences for control room personnel during various off-normal events. The CREVS makeup airflow is one of these factors that must be considered. Excessive makeup air or the inability of the CREVS units to supply design flow rates could result in an increase in the overall dose consequence to control room personnel. The flow verification ensures that an assumed amount of makeup air is available to account for boundary leak paths. If control room boundary leakage to adjacent areas is minimal, the makeup airflow rate will decrease accordingly as the differential pressure between the control room and adjacent areas increases. Therefore, the verification of makeup airflow capability may require creating leak paths (opening a door) when the control room envelope leak paths are minimal. The flowrate verification is consistent with SRP Section 6.4 (Reference 4) for those control rooms having a design makeup rate of ≥ 0.5 volume changes per hour. The Frequency of 18 months is considered adequate to detect any degradation of the outside air flow rate before it is reduced to a point at which sufficient pressurization will not occur.

REFERENCES

1. SAR, Section 9.7.

2. 10 CFR 50.36.

3. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

4 Standard Review Plan, Section 6.4, "Control Room Habitability System," Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Air Conditioning System (CREACS)

BASES

BACKGROUND

The CREACS provides temperature control for the control room following isolation of the control room.

The CREACS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, dampers, and instrumentation also form part of the system. During operation, the CREACS maintains the temperature in a range consistent with personnel comfort and long term equipment operation. The CREACS is a subsystem providing air temperature control for the control room.

The CREACS is an emergency system. On detection of high radiation, the control room envelope is isolated, the normal control room ventilation system is shut down, and the CREACS is started. A single train will provide the required temperature control. The CREACS operation to maintain control room temperature is discussed in the SAR, Section 9.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CREACS is to maintain control room temperature for 30 days of continuous occupancy.

The CREACS components are arranged in redundant, safety related trains. A single active failure of a CREACS component does not impair the ability of the system to perform as designed. The CREACS is designed in accordance with Seismic Category I requirements. The CREACS is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements, to ensure a habitable environment and equipment OPERABILITY.

In MODES 1 and 2, and during movement of irradiated fuel assemblies, the CREACS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the CREACS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two independent and redundant trains of the CREACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the control room temperature exceeding limits in the event of an accident.

For a CREACS train to be considered OPERABLE, the individual components that are necessary to maintain control room temperature must be OPERABLE. (Note: Because this is a shared system and is normally powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREACS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.10 must be applied for inoperable CREACS train(s).) These components include the cooling coils, condensing units, and associated temperature control instrumentation. In addition, the CREACS must be capable of maintaining air circulation.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREACS must be OPERABLE to ensure that the control room temperature will not exceed habitability and equipment OPERABILITY requirements following isolation of the control room.

ACTIONS

A.1

With one CREACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CREACS train could result in a loss of CREACS function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining train can provide the required capabilities, and alternate nonsafety related cooling means that are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

C.1 and C.2

During movement of irradiated fuel, if the Required Action and associated Completion Time of Condition A are not met, the OPERABLE CREACS train must be placed in operation immediately. This action ensures that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require the isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of irradiated fuel assemblies, with two CREACS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREACS trains are inoperable in MODE 1, 2, 3, or 4, a loss of safety function has occurred, and LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 and SR 3.7.10.2

These SRs, in conjunction with periodic preventative maintenance activities, provide verification that the CREACS will maintain the control room temperature within acceptable bounds. SR 3.7.10.1 is performed on a staggered basis with one train being tested every two weeks. The Frequencies (31 days and 18 months) are appropriate as periodic preventative maintenance activities are routinely performed and significant degradation of the CREACS is not expected over these time periods.

REFERENCES

1. SAR, Section 9.7.

2. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.11 Penetration Room Ventilation System (PRVS)

BASES

BACKGROUND

The PRVS filters air from the penetration areas in the event of penetration leakage from the reactor building during a loss of coolant accident (LOCA).

The PRVS consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, and an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the penetration rooms following receipt of an engineered safeguards actuation system (ESAS) signal.

Following a LOCA, an ESAS signal starts the lead PRVS and if proper flow is not achieved within 20 seconds, the lead system is automatically stopped and 5 seconds later the standby system starts. Upon receipt of the ESAS signal, normal air discharges from the area are isolated, and the air is discharged through the system filters. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PRVS is discussed in the SAR, Sections 6.5 and 14.2.2.5 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the PRVS is established by the LOCA. The system provides filtration for the most likely location of reactor building leakage, i.e., at the penetrations. The analysis of the effects and consequences of a LOCA is presented in Reference 2.

In MODES 1 and 2, the PRVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, the PRVS satisfies Criterion 4 of 10 CFR 50.36.

LCO

Two redundant trains of the PRVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power.

For a PRVS train to be considered OPERABLE, its associated:

a. Fan must be OPERABLE;

b. HEPA filter and charcoal adsorber must not be excessively restricting flow, and must be capable of performing their filtration functions; and

c. Required ductwork, and dampers must be OPERABLE.

The LCO is modified by a Note allowing the PRVS negative pressure boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for PRVS negative pressure boundary isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the OPERABILITY requirements of the reactor building.

In MODES 5 and 6, the PRVS is not required to be OPERABLE since the reactor building is not required to be OPERABLE.

ACTIONS

A.1

With one PRVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the PRVS safety function. However, the overall reliability is reduced because a single failure in the OPERABLE PRVS train could result in loss of PRVS function.
The 7 day Completion Time is appropriate because the risk contribution is less than that of the reactor building (1 hour Completion Time), and this system is not a direct support system for the reactor building. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

If the PRVS negative pressure boundary is inoperable, the PRVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE PRVS negative pressure boundary within 24 hours. During the period that the PRVS negative pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent, as applicable, of GDC 64 and

10 CFR Part 100) should be utilized to control and minimize the release of radioactive materials from the reactor building to the environment in post accident conditions. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possible repair, and test most problems with the PRVS negative pressure boundary.

C.1 and C.2

If the Required Action and the associated Completion Time are not met, or with both PRVS trains inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. The 31 day Frequency is based on known reliability of equipment and the two train redundancy available.

SR 3.7.11.2

This SR verifies that the required PRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.11.3

This SR verifies that each PRVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with the guidance provided in Regulatory Guide 1.52 (Ref. 4).

REFERENCES

1. SAR, Section 6.5.

2. SAR, Sections 14.2.2.5 and 14.2.2.6.

3. 10 CFR 50.36.

4. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Handling Area Ventilation System (FHAVS)

BASES

BACKGROUND

The FHAVS filters airborne radioactive material from the area of the spent fuel pool following a fuel handling accident.

The FHAVS consists of portions of the normal Auxiliary Building Heating, Ventilation, and Air Conditioning System. The FHAVS consists of a single train which includes a supply fan, prefilter, high efficiency particulate air (HEPA) filter, activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two exhaust fans. Ductwork, dampers, and instrumentation also form part of the system.

During operation, the exhaust from the fuel handling area is passed through the FHAVS exhaust filter and is discharged through the station vent stack.

The FHAVS is discussed in the SAR, Sections 9.7 and 14.2.2 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The FHAVS design basis is established by the fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, credits the FHAVS for a reduction in the amount of airborne radioactive material released to the environment. The assumptions and the analysis are further discussed in Reference 2.

The FHAVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

During movement of irradiated fuel, the FHAVS is required to be OPERABLE and operating.

For the FHAVS to be considered OPERABLE:

1. One exhaust fan must be OPERABLE;

2. HEPA filter and charcoal adsorber must not be excessively restricting flow, and must be capable of performing their filtration functions; and

3. Ductwork and dampers must be OPERABLE.

The FHAVS must be operating since it does not automatically start following a fuel handling accident. A supply fan may be operating, but is not required for FHAVS OPERABILITY.

APPLICABILITY

During movement of irradiated fuel assemblies in the fuel handling area, the FHAVS is always required to be OPERABLE and operating to mitigate the consequences of a fuel handling accident.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note which states that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the FHAVS is inoperable or not in operation during movement of irradiated fuel assemblies in the fuel handling area, immediate action must be taken to preclude the occurrence of an accident. This is achieved by immediately suspending movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Periodic verification of the operation of the FHAVS assures immediate availability of filtration following a fuel handling accident. A 12 hour Frequency is sufficient, considering the system indications and alarms available to the operator for monitoring the FHAVS in the control room.

SR 3.7.12.2

This SR verifies that the required FHAVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. SAR, Section 9.7.
2. SAR, Section 14.2.2.
3. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumption of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pool design is given in the SAR, Section 9.6.1.3, Reference 1. The Spent Fuel Pool Cooling and Cleanup System is given in the SAR, Section 9.4 (Ref. 2). The assumptions of the fuel handling accident are given in the SAR, Section 14.2.2.3 (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the spent fuel pool is an initial condition design parameter in the analysis of the fuel handling accident in the fuel handling building postulated by Regulatory Guide 1.25 (Ref. 4). A minimum water level of 23 feet above the top of the irradiated fuel assemblies seated in the storage racks (Regulatory Position C.1.c of Ref. 4) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 4) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the spent fuel pool water. The fuel pellet to cladding gap is assumed to contain 12% of the total fuel rod iodine inventory (Ref. 3).

The fuel handling accident analysis inside the fuel handling building is described in Reference 3. With a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated in the storage racks, and a minimum decay time of 100 hours prior to fuel handling, the analysis demonstrates that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 5).

The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6).

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the spent fuel pool at less than the required level, the movement of fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of loads over the spent fuel. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

During refueling operations, the level in the spent fuel pool is at equilibrium with that in the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

REFERENCES

1. FSAR, Section 9.6.1.3.
2. FSAR, Section 9.4.
3. FSAR, Section 14.2.2.3.
4. Regulatory Guide 1.25.

5. 10 CFR 100.11.

6. 10 CFR 50.36

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the Bases for LCO 3.7.15, "Spent Fuel Pool Storage," fuel assemblies are stored in the spent fuel pool racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 1600 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron in the spent fuel pool water.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.15-1. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with Specification 4.3.1.1.e.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978, NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Thus, for accident conditions, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic condition. For example, accident scenarios are postulated which could potentially increase the reactivity and reduce the margin to criticality. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the high density storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.15, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.15.1.

APPLICABLE SAFETY ANALYSES

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more that eight inches of water separating it from the active fuel in the rest of the rack which precludes interaction). However, accidents can be postulated which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Thus, for accident conditions, the presence of soluble boron in the storage pool water is assumed as a realistic initial condition.

The presence of 1600 ppm boron in the pool water will decrease reactivity by approximately 30% ΔK. Thus $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The specified concentration ≥ 1600 ppm of dissolved boron in the spent fuel pool conservatively preserves the assumption used in the analyses of the potential accident scenarios. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movement in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. In addition, action must be immediately initiated to restore the spent fuel pool boron concentration to within its limit. An acceptable alternative is to immediately initiate performance of a spent fuel pool verification to ensure proper locations of the fuel since the last movement of fuel assemblies in the spent fuel pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. Either of these actions are acceptable, and once initiated must be continued until the action is completed. The immediate Completion Time for initiation of these actions reflects the importance of maintaining a controlled environment for irradiated fuel.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SAR, Section 14.2.2.3.

3. Safety Evaluation Report, Section 2.1.3, License Amendment No. 76, April 15, 1983.

4. 10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel assembly storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The spent fuel pool is sized to store 968 fuel assemblies. The spent fuel storage cells are installed in parallel rows with center to center spacing of 10.65 inches in each direction.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.15-1. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with paragraph 4.3.1.1.e in SAR Section 4.3, Fuel Storage.

APPLICABLE SAFETY ANALYSES

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies in Region 1. Region 2 controls fuel assembly interaction by fixing the minimum separation between assemblies and by setting enrichment and burnup criterion to limit fissile materials. This is sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to 4.10%. However, fuel assemblies to be stored in the spent fuel pool Region 2 which do not meet enrichment and burnup criterion must be stored in a checkerboard pattern to maintain a k_{eff} of 0.95 or less. In order to prevent inadvertent fuel assembly insertion into two adjacent storage locations, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (unrestricted) are physically blocked before any such fuel assembly is placed in Region 2 (Ref. 1). In addition, the area designated for checkerboard arrangement is divided from the normal storage in Region 2 by a row of vacant storage spaces (Ref. 2).

The spent fuel pool storage satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to Figure 3.7.15-1, ensure that the k_{eff} of the spent fuel pool will always remain ≤ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool. Fuel assemblies not meeting the enrichment and burnup criteria shall be stored in accordance with Specification 4.3.1.1.

In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.15-1 or Specification 4.3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 in the accompanying LCO or Specification 4.3.1.1. For fuel assemblies in the unacceptable range of

Figure 3.7.15-1, performance of the SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. SAR, Section 9.6.2.

2. SER for ANO-1 License Amendment No. 76, Section 2.1 (0CNA048314), dated April 15, 1983.

3. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.7: Plant Systems

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 RSTS Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The CTS 4.8.1.b phrase "each EFW flowpath" is clarified to include both the water flow paths and both steam supply flow paths in proposed SR 3.7.5.1. This change is consistent with NUREG-1430.

A4 NUREG 3.7.8 (ITS 3.7.7) Required Action A.1 Notes 1 and 2 are incorporated to retain the CTS cascading inoperability for affected emergency diesel generators and decay heat removal subsystems. Since these would be considered inoperable under the CTS, the addition of these Notes is an administrative change (necessary due to the differing format and implementation of ITS) to retain the CTS requirements. This change is consistent with NUREG-1430.

A5 An explicit Applicability is included for CTS 3.10 as MODES 1, 2, 3, and 4. This is considered equivalent to the CTS even though no explicit applicability is identified with the LCO. The associated Surveillance is identified in CTS Table 4.1-3, item 5, and Notes (7) and (10) identify the applicability for the requirements. In MODES 5 and 6 (CTS cold shutdown and refueling) and when the steam generators are not generating steam (also considered to be cold shutdown and refueling), the secondary coolant is at low temperature and pressure with minimal opportunity for significant release. Therefore, the secondary specific activity is not important. As such, the proposed Applicability is considered equivalent to the current application. This change is consistent with NUREG-1430.

A6 An additional Condition is included for CTS 3.9.1 and 3.9.2 to direct entry into LCO 3.0.3 if both trains of the control room emergency ventilation system (CREVS) or the control room emergency air conditioning system (CREACS) while in MODES 1, 2, 3, or 4. This is equivalent to the CTS requirements and is needed as an explicit condition only due to differences in the implementation. This change is consistent with NUREG-1430.

ANO-334 A7 Not used.

A8 3.7-14 The "at greater than 1600 ppm" requirement for boron concentration of the spent fuel pool in CTS 3.8.17 has been revised to "≥ 1600 ppm" in ITS 3.7.14. These are considered to be essentially equivalent since the parameter can be less than the limit, but be so close as to be imperceptible. This change is consistent with design basis and with NUREG-1430.

A9 Not used.

A10 This information has been removed from the ITS since it duplicates requirements provided in the regulations. Such duplication is unnecessary and results in additional administrative burden to revise the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, removal of the duplicative information is considered an administrative change. Further, changes to the requirements are controlled by the NRC. This change is consistent with NUREG-1430.

CTS Location	Duplicated Regulation
3.12.3	10 CFR 30, 40 & 70

A11 Not used.

ANO-362 A12 Not used.

A13 Not used.

A14 This page is not yet approved as provided in this package. Therefore, this markup is dependent on the expected NRC approval of the January 27, 2000, (Ref. 0CAN010004) license amendment request (LAR) related to the Q Condensate Storage Tank volume.

3.7-07 A15 CTS 3.3.1.E requires both low pressure injection (LPI) coolers and their cooling water supplies to be operable whenever containment integrity is established. The portion of CTS 3.3.1.E specifying the LPI coolers is contained in ITS 3.5.2 and ITS 3.5.3. However, the portion of CTS 3.3.1.E specifying the cooling water supplies is incorporated in ITS 3.7.7. ITS 3.7.7 requires two loops of service water to be OPERABLE. This is acceptable because the cooling water supply to the LPI coolers is the service water system, and the service water system is required to be OPERABLE in the same MODES as the LPI system. This maintains the proper support system relationship for the service water system and the LPI coolers.

ANO-239 A16 The CTS 3.9.2 requirements for the control room emergency ventilation system (CREVS) have been revised to include a note stating that one train of the CREVS shall be capable of automatic actuation. Amendment 10, dated February 18, 1976, incorporated technical specifications for the control room emergency air conditioning system. Specification 3.9.1 required that two independent circuits of the control room emergency air conditioning system be operable whenever reactor building integrity was required and that one of the systems shall be capable of automatic initiation (Specification 3.9.1.e). The Bases associated with these requirements stated that one circuit is designed to automatically start upon control room isolation and the other circuit to be manually started on failure of the first circuit. In the Safety Evaluation associated with Amendment 10, the NRC staff concluded that the proposed specifications provided reasonable assurance that the system would function, when needed, as described in the Final Safety Analysis Report and the NRC staff's Safety Evaluation dated June 6, 1973.

Specification 3.9.1.e was deleted by Amendment 196, dated May 19, 1999. In the request to modify the technical specifications, dated April 4, 1995, ANO stated that the requirements of Specification 3.9.1.e would be maintained by the proposed changes to Specification 3.5.1.10, the incorporation of TS 3.5.1.17, by inclusion of the control room radiation monitoring system in Table 3.5.1-1, and by the existing Table 3.5.1-1 requirements on the chlorine detection system. Relocation of this requirement was considered to be administrative in nature. In response to NRC comments and questions, and to incorporate changes due to other license amendment requests, ANO revised the submittal by letter dated December 12, 1996, and again by letter dated August 6, 1998. Neither of these resubmittals provided any changes that would have required both circuits of the control room emergency ventilation system to be capable of automatic actuation. This was confirmed in the Amendment 196 Safety Evaluation, dated May 19, 1999, in which the NRC staff concluded that the relocation of the Specification 3.9.1.e requirements to Specification 3.5.1.13 and Table 3.5.1-1 was administrative and acceptable. Unfortunately, the specifications, as implemented, do not present this specific statement in a clear manner.

The intent of the relocation of the Specification 3.9.1.e requirement that one circuit be capable of automatic initiation is clear from both the ANO and NRC correspondence. There was no intent to require the capability of both circuits of control room emergency ventilation to be capable of automatic initiation. This position has been evaluated by the ANO 10 CFR 50.59 process as a Bases change to the current technical specifications, and found to be acceptable. Therefore, the addition of the ITS 3.7.9 LCO Note stating that one train shall be capable of automatic initiation is considered to be an administrative change in that it clarifies the intent of the current license basis.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 The CTS 3.4.2 shutdown requirements have been revised to adopt the RSTS Completion Times which requires the reactor to be subcritical in 6 hours rather than 12 hours. The RSTS Completion Times also do not allow the additional 48 hours to attempt restoration of compliance. Finally, the RSTS Completion Times for placing the unit in a cold shutdown condition within 12 hours are adopted in lieu of the CTS allowance for an additional 24 hours. These Completion Times are considered to be reasonable and sufficient, considering operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This is considered to be an additional restriction on unit operation which is consistent with NUREG-1430.

M2 The CTS 3.4.1.2 requirements for MSSVs indicate only that 14 of the steam system safety valves are required to be OPERABLE. The CTS does not indicate that these 14 MSSVs must be arranged such that 7 are OPERABLE on the steam line associated with one steam generator and 7 are OPERABLE on the steam line associated with the other steam generator. This specificity is considered to be more restrictive than CTS, but it consistent with the safety analysis and NUREG-1430.

M3 The CTS 3.15.1 requirements are revised to also specifically include a requirement for OPERABILITY of the Fuel Handling Area Ventilation System (FHAVS). Although specific performance criteria are included in the CTS, other ITS requirements for OPERABILITY such as the OPERABILITY of supporting systems could be interpreted as not applicable to the FHAVS. OPERABILITY requirements are appropriate to assure the FHAVS will perform its function when required. This change is considered to be additional restriction on unit operation consistent with NUREG-1430.

3.7-02 M4 The CTS 3.4.1, 3.4.1.5, 3.4.2, and Table 4.1-2, Item 14 requirements have been revised to incorporate the main feedwater block valves, low load feedwater control valves and startup feedwater control valves. These valves are credited in the MSLB analysis and per 10 CFR 50.36, Criterion 3, should be retained in the ITS. Incorporating these valves in the ITS results in more restrictive requirements than currently specified.

The Required Actions for an inoperable component have been revised to allow an inoperable component to exist for 72 hours, instead of the CTS 24 hours. This aspect of the change is discussed in DOC-L4. If the component is not restored to Operable within 72 hours, the ITS will require the unit to be placed in MODE 3 within 6 hours and MODE 4 within an additional 6 hours, instead of the currently specified 12 hours to Hot Shutdown and if not restored in an additional 48 hours, in Cold Shutdown within 24 hours. This results in a more restrictive requirement with respect to exiting the MODE of Applicability since the ITS will allow a total of 84 hours where the CTS allowed a total of 108 hours.

The ITS is proposed to contain requirements for periodic verification of the closed status of MSIVs, MFIVs, main feedwater block valves, low load feedwater control valves and startup feedwater control valves which have been closed as the result of Required Actions. These actions are not currently required since the CTS does not allow continued operation with these valves inoperable, but closed (see related DOCs L3 & L4). These requirements for periodic verification are additional restrictions on unit operation consistent with NUREG-1430.

3.7-02 The CTS 3.4 requirements have also been revised to provide a Required Action in the event two valves in the same flow path are inoperable for one or more flow paths. This change recognizes the addition of the main feedwater block valves, low load feedwater control valves and startup feedwater control valves to the ITS. Each main feedwater line has three possible flow paths; startup feedwater flow via the startup feedwater control valve and the MFIV, low load feedwater flow via the Low Load Feedwater Control Valve and the MFIV, and main feedwater flow via the Main Feedwater Block Valve and the MFIV. Should the MFIV become inoperable concurrent with an inoperability of the main feedwater block valve, low load feedwater control valve or startup feedwater control valve in the same main feedwater line, the ITS will require the flow path to be isolated within 8 hours. This Completion Time is appropriate, since the MSLB analysis assumes that the main feedwater flow line is isolated and is acceptable, based on the low probability of an MSLB occurring during any specific 8 hour period of time.

M5 The CTS requirement (Table 4.1-2, items 13.b & 14.b) to cycle the MSIVs and MFIVs is revised to include the stroke time testing and functional testing of the isolation capability on an actuation signal as normally required for isolation valves. However, since the testing should be accomplished under conditions of operating pressure and temperature and may be required to verify OPERABILITY following work on the valve during a shutdown, a Note is included to allow the testing to be conducted in MODE 3. Allowing testing in MODE 3 (rather than MODE 4, 5, or 6) more closely simulates the conditions under which the valve may be required to perform its safety function. These additional test requirements are considered to be additional restrictions on unit operation consistent with NUREG-1430.

M6 The CTS requirements for OPERABILITY of the Condensate Storage Tank (CST) are expanded to include MODE 4 when the steam generator is relied upon for heat removal. This is consistent with the OPERABILITY requirements for the Emergency Feedwater System and with RSTS LCO 3.7.6. This additional applicability is an additional restriction on unit operation consistent with NUREG-1430.

M7 The CTS 3.4.2 requirements for actions with an inoperable T41B are revised to those presented in NUREG-1430 for the CST. Required Action A.1 has been added which requires the verification by administrative means the operability of the backup water supply (for ANO-1 this is the service water system). This additional Required Action is an additional restriction on unit operation consistent with NUREG-1430.

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Additionally, if the CST is not restored to operable status or the backup water supply is not verified to be operable, the Completion Time for placing the unit in a subcritical condition is reduced to 6 hours from 12 hours, and the Completion Time for placing the unit in a condition in which the LCO does not apply after becoming subcritical is reduced from 72 hours to 18 hours. These Completion Times provide sufficient time to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems and are consistent with NUREG-1430.

Finally, a Surveillance Requirement is incorporated to periodically verify the volume of the CST is within limits. The surveillance is necessary to periodically verify the primary EFW water source is available as assumed in the safety analysis. These changes are also additional restrictions on unit operation consistent with NUREG-1430.

M8 An additional Completion Time has been added to those in CTS 3.4.4 to not only require the steam supply to be restored within 7 days from discovery of the inoperable pump (proposed Required Action A.1), or the train within 72 hours (proposed Required Action B.1), but also within 10 days from discovery of failure to meet any of the requirements of the LCO. Currently, for example, if the motor driven pump and one steam supply to the turbine driven pump are concurrently inoperable, separate Actions are entered and the associated Actions are performed with separate Completion Times. Since there are multiple Conditions for different components that are inoperable, it is possible, (however it is extremely unlikely), that the unit can have at least one component inoperable for an unlimited time, and yet a shutdown would never be required (i.e., individual components are repaired within these required restoration times, but there is always at least one component inoperable). The new Completion Time establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO. This is an additional restriction on unit operation consistent with NUREG-1430.

M9 CTS 3.3.1 (C) and (I) requires that the service water system pumps and valves be OPERABLE "whenever containment integrity is established as required by Specification 3.6.1." CTS 3.6.1 requires containment integrity whenever RCS pressure is ≥ 300 psig, RCS temperature is ≥ 200°F, and fuel is in the reactor. The ITS requirement for service water pumps is independent of RCS pressure. The pumps and valves will be required with fuel in the reactor and RCS temperature ≥ 200°F. This is an additional restriction on unit operation consistent with NUREG-1430.

M10 CTS 3.3.1(I) requires the valves associated with the service water system to be OPERABLE or locked in the engineered safeguards position, but there are no surveillance requirements specified to verify this requirement. RSTS SR 3.7.8.1 is proposed to be adopted (as ITS SR 3.7.7.1) to periodically verify the position of valves which are not secured in the correct position. ITS SR 3.7.7.1 is also proposed with a Note that indicates that isolation of flow to individual components does not render the SWS inoperable. Overall, this new surveillance is considered an additional restriction on unit operation consistent with NUREG-1430.

M11 CTS 3.3.6 requires that for an inoperable service water subsystem, the unit be placed in a subcritical condition (hot shutdown) within 36 hours of noncompliance, and allows an additional 72 hours to achieve a cold shutdown condition. The ITS provides only 6 hours to achieve MODE 3 (HOT SHUTDOWN) and an additional 30 hours to achieve MODE 5 (COLD SHUTDOWN). The times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This is an additional restriction on unit operation consistent with NUREG-1430.

M12 The CTS functional test of the service water components required by Table 4.1-2, item 9, is expanded to identify more detail as to the content of the test requirements. ITS SR 3.7.7.2 requires each automatic valve that is not secured in its correct post-accident position to be verified to actuate to its correct position on an actual or simulated actuation signal and ITS 3.7.7.3 requires a verification that each required SW pump starts automatically on an actual or simulated signal. This additional detail is considered an additional restriction on unit operation consistent with NUREG-1430.

3.7-08

M13 CTS 3.11.1 requires the emergency cooling pond to be OPERABLE whenever containment integrity is established as required by (CTS) Specification 3.6.1. CTS 3.13.1 similarly requires the penetration room ventilation system (PRVS) to be OPERABLE whenever reactor building integrity is required. CTS 3.6.1 requires that reactor building integrity be (established and) maintained whenever all three of the following conditions exist: (a) reactor coolant pressure is 300 psig or greater, (b) reactor coolant temperature is 200°F or greater, and (c) nuclear fuel is in the core. The proposed Applicability for ITS 3.7.8 and ITS 3.7.11 is MODES 1, 2, 3, and 4 which incorporates items (b) and (c) of CTS 3.6.1. However, the ITS requirements will be applicable regardless of reactor coolant pressure. This is an additional restriction on unit operation consistent with NUREG-1430. (For CTS 3.13.1 requirements per CTS 3.6.2, see DOC L17.)

M14 The CTS 3.10 requirement to place the unit in a Hot Standby condition within 6 hours if the secondary specific activity limits are not met is revised to require the unit to be placed in MODE 3 in 6 hours. ITS MODE 3 requires the unit to be subcritical, whereas the CTS Hot Standby required that the unit be at less than 2% of rated power. This is an additional restriction on unit operation consistent with NUREG-1430.

M15 Not used.

M16 The CTS Table 4.1-3, item 4 requirement to perform a spent fuel pool boron concentration verification on a monthly Frequency is revised to a weekly verification (during the times the Specification is applicable; see DOC L15). This is an additional restriction on unit operation consistent with NUREG-1430.

M17 Appropriate Required Actions are incorporated for a condition of noncompliance with CTS 3.8.16 and 3.8.17. The proposed action for CTS 3.8.16 requires the immediate initiation of action to move the noncomplying fuel assembly. The proposed action for CTS 3.8.17 requires prompt restoration of the boron concentration to within limits or removal of the potential for a fuel handling accident. These actions are not explicitly identified in the CTS, and therefore, are additional restrictions on unit operation consistent with NUREG-1430.

M18 CTS 3.9.1 and 3.9.2 contain requirements for OPERABILITY of the CREVS and CREACS during movement of irradiated fuel in the reactor building but does not include an Applicability for movement of irradiated fuel in the fuel handling area, nor does the CTS include ACTIONS for an inoperable train of CREVS or CREACS during these fuel movements. The addition of the Applicability and Required Actions is an additional restriction on unit operation consistent with the safety analysis and with NUREG-1430.

M19 Not used.

M20 An additional intermediate Required Action is added to CTS 3.13.3 to place the unit in MODE 3 within 6 hours if an inoperable penetration room ventilation system (PRVS) train is not restored to OPERABLE status within 7 days. This is an additional restriction on unit operation consistent with NUREG-1430.

M21 Not used.

M22 Not used.

M23 Not used.

M24 NUREG Required Action A.1 is included in ITS 3.7.1 to ensure sufficient MSSV capacity to mitigate an overpressure event. This action is not required in the CTS since continued operation for an indefinite period of time with less than 14 MSSVs is prohibited (see DOC L1). This is an additional restriction on unit operation consistent with NUREG-1430.

NUREG Required Action A.2 is included in ITS 3.7.1 for extra conservatism. Therefore, requirements for reduced maximum allowable nuclear overpower - high trip settings are included based on the number of OPERABLE MSSVs. This is an additional restriction on unit operation consistent with NUREG-1430.

M25 An additional surveillance, beyond CTS 4.17, is included to periodically verify "in operation" as it is required by ITS 3.7.12. This is necessary to verify the assumptions of the safety analysis are met during conditions in which a fuel handling accident may occur. This is an additional restriction on unit operation consistent with NUREG-1430 as modified for unit specific design and analysis. (See also DOD 35.)

M26 CTS 4.8.1.a.1 requires that the turbine driven emergency feedwater pump be tested within 24 hours after reaching the Hot Shutdown condition following a plant heatup and prior to criticality. This is revised in the ITS SR 3.7.5.2 Note to require the testing to be performed with 24 hours after reaching ≥ 750 psig. Since 750 psig occurs prior to reaching 525°F, this test is required to be performed earlier in the startup than it is currently performed. However, the proposed conditions are sufficient to allow the test to be performed and verify OPERABILITY earlier in the conditions applicable to the required equipment. This is an additional restriction on unit operation consistent with NUREG-1430.

M27 CTS 4.8.1.e.2 requires that the automatic actuation of the turbine driven emergency feedwater pump steam supply valves (and the associated turbine driven pump) be tested within 24 hours after reaching the Hot Shutdown condition (if it is not current). This is revised in the ITS SR 3.7.5.3 and SR 3.7.5.4 to require the testing to be performed prior to entry into MODE 3 (i.e., ≤ 280°F). Since the pump is only required to start (and is not required to reach full flow for this test), the test can be performed at less than the 750 psig required for pump flow functional testing. This assures system performance verification occurs prior to entering unit conditions where such performance may be needed to respond to an event. This is an additional restriction on unit operation consistent with NUREG-1430 as modified for unit specific design. (See also DOD 14.)

3.7-14

M28 NUREG-1430 LCO 3.7.14 has been incorporated as ITS LCO 3.7.13. This LCO provides requirements for the spent fuel pool level that are not specified in the CTS. Spent fuel pool level is an assumption of the fuel handling accident and therefore meets the requirements of 10 CFR 50.36, Criterion 2, for inclusion in the ITS. Since the spent fuel pool level is currently controlled administratively, incorporation of spent fuel pool level is considered to be an additional restriction on operation, and therefore, more restrictive.

3.7-12 M29. The CTS 4.10.2 testing requirements have been revised to include a test to verify that the control room emergency ventilation system makeup flow rate is ≥300 and ≤366 cfm when supplying the control room with outside air. SRP Section 6.4 Rev. 2 (dated July 1981) recommends that this test be performed periodically (every 18 months) for control rooms, like ANO-1, which are designed for a pressurization rate of ≥ 0.5 volume changes per hour. Although the ANO-1 Operating License predates the SRP, the incorporation of SR 3.7.9.4 will provide assurance that the control room will be supplied sufficient outside air to provide a pressurized environment. The addition of this SR is considered to be an additional restriction on operation, and therefore, more restrictive.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 The CTS 3.4.1.2 requirements for 14 OPERABLE MSSVs regardless of the power level have been revised to require only the number of MSSVs required to mitigate an overpressure event initiated at specified power levels. The specific number of MSSVs required for various power levels are identified in ITS Table 3.7.1-1.

CTS 3.4.2 allows operation with less than 14 OPERABLE MSSVs for a period of 24 hours, after which the unit must shutdown. The proposed Required Actions of ITS 3.7.1 Condition A will allow continued operation for an indefinite period of time provided that reactor power is reduced to a level consistent with that provided in Table 3.7.1-1 within 4 hours, and the nuclear overpower trip setpoint is reduced in accordance with Table 3.7.1-1 within 36 hours. These proposed actions will ensure that the relief capacity of the remaining MSSVs is sufficient to mitigate an overpressure event during operation with less than 14 MSSVs OPERABLE. Although this allowance to continue operation beyond 24 hours with less than 14 MSSVs results in a less restrictive requirement, additional restrictions on unit operation (i.e., required power reduction within 4 hours and nuclear overpower trip setpoint reduction within 36 hours) are implemented (See also DOC M24).

3.7-17

The CTS 3.4.1.2 requirements have also been revised to allow separate condition entry for each inoperable MSSV. CTS 3.4.1.2 requires action to be taken in the event less than 14 MSSVs are Operable. Therefore, separate entry into the actions for each MSSV was not required. With the incorporation of ITS 3.7.1, operation with more than two inoperable MSSV is allowed if the appropriate actions, as discussed above, are taken. Separate Condition entry is required to be implemented in the ITS due to the structure and format of the ITS. Separate Condition entry recognizes the fact that MSSVs may become inoperable at different times, thus requiring accelerated actions in responding to the second inoperability requiring entry into the condition. Without a Separate Condition Entry allowance, If the Actions of Condition A have already been implemented due to one required MSSV inoperable, a subsequent failure of an MSSV four hours later would require an immediate power reduction. The Separate Condition Entry allowance ensures that the operator has sufficient time to prepare for and implement a power reduction, while the Completion Time of the associated Required Action ensures that the action is taken in a timely manner.

L2 The CTS requirement (Table 4.1-2, item 4) for the testing of the MSSV setpoints is revised to allow in-situ testing in MODE 3 during startup. Currently, this testing may be performed either during the pressure and temperature reduction for a shutdown, or during the refueling outage by bench testing. The addition of the Note for ITS SR 3.7.1.1 will allow entry into MODE 3 and testing in MODE 3 during the startup following an outage. This is consistent with current practice at many nuclear power plants and is considered an acceptable method for testing of these valves. This change is consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

L3 The CTS 3.4.2 requirements for placing the unit in cold shutdown if the other Required Actions are not met is revised to require only that the unit be placed in a condition in which the requirements for the inoperable equipment are not applicable. For the MSSVs, MSIVs, and MFIVs, this will require only that the unit be placed in MODE 4. The CTS required that the unit be placed in cold shutdown (equivalent to ITS MODE 5) even though the equipment was only required above 280°F. This is consistent with NUREG-1430 general application for Required Actions.

L4 The CTS 3.4.2 requirements for shutdown if one MSIV is inoperable are proposed to be revised to allow continued operation in MODE 3 if the isolation valve is closed and periodically verified to remain closed. This is appropriate since the only safety function of the isolation valves is closure. The Completion Time is appropriate since the valve isolates a closed system which provides an additional barrier for containment isolation. Therefore, the CTS allowed time for continued operation in MODE 3 prior to any action, i.e., 48 hours, is retained as the proposed Completion Time for isolation valve closure. Since each such inoperability will require an additional closure, a Note is included to allow separate entry into the Condition for each inoperable MSIV (or MFIV). This Note is consistent with NUREG-1430.

3.7-02

The CTS 3.4.2 requirements for shutdown if one MFIV is inoperable are proposed to be revised to allow continued operation in MODE 3 if the isolation valve is closed and periodically verified to remain closed. This is appropriate since the only safety function of the isolation valves is closure. The Completion Time to restore an inoperable MSIV to Operable status has been revised from 24 hours to 72 hours. This Completion Time is acceptable due to the presence of a redundant set of valves (Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves) in each main feedwater line. Since each such inoperability will require an additional closure, a Note is included to allow separate entry into the Conditions for each inoperable MFIV. These changes are consistent with NUREG-1430.

L5 The CTS Table 4.1-2 (items 13.a and 14.a) quarterly exercising of the MSIVs and MFIVs is omitted. This exercising, while typically required by Section XI for isolation valves, is normally excepted for MSIVs and MFIVs since even partial stroke testing of these valves increases the risk of a valve closure with the unit generating power. Such a valve closure would result in an unnecessary transient. The normal stroke testing of these valves during startup following a refueling outage (see related DOC M5) provides sufficient verification of the OPERABILITY of these valves. This change is consistent with NUREG-1430.

L6 Not used.

CTS DISCUSSION OF CHANGES

L7 The CTS 3.4.2 requirements for shutdown with an inoperable condensate storage tank (CST) are proposed to be revised to allow continued operation for up to 7 days. Two safety related sources of water are provided for the emergency feedwater (EFW) pumps. The first, and preferred source, is the "Q" CST, T-41B, which is seismically qualified and partially tornado protected. The second, and backup source, is the safety related and seismically qualified service water system. The portion of T-41B which is tornado protected provides a 30 minute supply of water for the EFW pumps which provides time for the operators to manually align the EFW pumps to the alternate source. Since the service water system is required to be OPERABLE (see related DOC M7), the extended Completion Time for an inoperable CST has no significant effect on safety. This 7 day Completion Time is consistent with NUREG-1430.

Additionally, the Required Actions are revised to require the unit to be placed in MODE 4 without reliance on a steam generator for heat removal rather than MODE 5. The proposed action is sufficient to place the unit in a condition which is outside the Applicability of the LCO. This change is consistent with NUREG-1430.

L8 The CTS 3.4.4 requirements for actions to be taken with inoperable EFW equipment include requirements for a shutdown with both EFW pumps inoperable if the nonsafety grade auxiliary feedwater (AFW) pump is available. This requirement for a shutdown is proposed to be deleted. While all available documentation may indicate that the AFW pump is available, its actual availability cannot be determined until the unit is partially shutdown to the point that AFW would be placed into service. If AFW is determined to be unavailable at this point, no other source of feedwater is readily available to support continuing the shutdown.

The proposed Required Actions will require that immediate action be taken to restore one EFW pump to OPERABLE status and, if required, initiate shutdown. This proposed action does not remove the normal feedwater system (which is providing feedwater to the steam generators) from service to depend on nonsafety grade equipment for which there is no assurance of availability. This is consistent with the Bases provided for NUREG LCO 3.7.5, Required Action D.1 which states: "the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip." This change is consistent with NUREG-1430.

L9 The CTS 4.8 Surveillance Frequency for EFW pump testing is revised to be consistent with the ASME Section XI requirements. CTS 4.8.1 requires EFW pump testing on a monthly basis. As discussed in NUREG-1366, Section 9.1, industry studies indicate that EFW pump testing on a monthly basis may be contributing to equipment unavailability and that changing the test Frequency to quarterly is reasonably expected to increase the availability of the EFW system. A quarterly Frequency is also consistent with the ASME Section XI requirements. This change is consistent with NUREG-1430 as modified by TSTF-101.

L10 The CTS 4.8 Surveillances are revised to exclude functional requirements for automatic actuation capability to be consistent with the requirements for OPERABILITY of the automatic actuation system. During these excluded operating conditions (i.e., MODE 4), there is more time available for operator action in response to an event which requires emergency feedwater initiation than in higher MODES.

The CTS 4.8 Surveillance Frequency is also revised to exclude that portion of the CTS requirements for performing the turbine driven feedwater pump testing prior to criticality. This is acceptable since the pump is required to be OPERABLE upon entry into the applicable conditions of ITS LCO 3.7.5, and the testing is only a verification of that OPERABILITY. As indicated in Generic Letter 87-09, "it is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed because the vast majority of surveillances do in fact demonstrate that systems or components are OPERABLE." Further, the 24 hours is consistent with the time allowed by SR 3.0.3 to perform the surveillance if it is discovered while in MODE 1 to not have been performed on schedule. This change is consistent with NUREG-1430.

L11 CTS 3.12.2 and 6.12.5.e require that a Special Report be submitted when radioactive material source leakage is identified above certain limits. This report is proposed to be eliminated. This reporting is not required by ITS, and is in addition to the reporting required of other 10 CFR Part 30, Part 40, and Part 70 licensees. The testing for leakage and associated corrective actions, when necessary, are retained under administrative controls (see DOC LA3) but the Special Report is an unnecessary use of licensee and regulator resources since it does not provide a significant corresponding benefit. The reporting criteria of 10 CFR Parts 30, 40, and 70 provide sufficient information. As before, any deficiency which is reportable under 10 CFR Part 30, Part 40, and Part 70, will be reported in accordance with the regulations. This change is consistent with NUREG-1430 and the regulations.

L12 CTS 4.8.1.c is revised to reflect that the verification of manual valve position in each required EFW flow path must be performed prior to entering MODE 2 rather than "prior to relying on the steam generator for heat removal." As discussed in the CTS 4.8.1.c Bases and the Bases for NUREG SR 3.7.5.5, this verification must be made prior to relying on the EFW system for decay heat removal following a subsequent unit shutdown. This change is acceptable because no appreciable change in decay heat magnitude will have occurred during the transition from MODE 5 to MODE 3. This change is consistent with NUREG-1430.

L13 The general CTS 3.3.5 and 3.3.6 requirements which are applicable to an inoperable service water train are revised to be consistent with specific RSTS requirements for an inoperable service water train. CTS 3.3.5 allows a service water train to be made inoperable for up to 24 hours for maintenance, but only if the redundant component in the other train is demonstrated OPERABLE within 24 hours prior to beginning the maintenance. However, the performance of maintenance on one train does not change the basis for believing that the redundant train is OPERABLE, therefore, this requirement is omitted. CTS 3.3.5 is marked as being less restrictive with respect to ITS LCO 3.7.7 because this explicit requirement is not retained in the ITS. The ITS Completion Times are based on the capabilities provided by the OPERABLE train and the low probability of a design basis accident occurring during this time period. This change is consistent with NUREG-1430.

The Completion Time for restoring an inoperable service water train (regardless of the reason for the inoperability) is extended from 36 hours to 72 hours. These Completion Times are based on the capabilities provided by the OPERABLE train and the low probability of a design basis accident occurring during this time period. This change is consistent with NUREG-1430.

L14 The CTS 4.11.5 required time for operation of the penetration room ventilation system (PRVS) is reduced from 1 hour to 15 minutes since the system does not have heaters. Similarly, the CTS 4.17.4 requirements for the FHAVS to operate for at least 10 hours is deleted since the system does not include heaters. Requiring the system to be operated for longer than 15 minutes is unnecessary since the system is required to be in operation during fuel movement. Much longer periods of operation are necessary if the system contains heaters that must operate to periodically dry out the charcoal in the filters. However, this shorter period of operation has been determined to be sufficient for determination that the system functions properly when the system contains no heaters. This change is consistent with NUREG-1430.

L15 The CTS 3.8.17 applicability for spent fuel pool boron concentration has been revised from "at all times" to "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." Once fuel assembly movement has ceased and it is verified that there are no misloaded fuel assemblies, there is no further potential for a misloaded fuel assembly or a dropped fuel assembly, either of which could result in a positive reactivity effect which decreases the margin to criticality. Other control of the boron concentration would be for reasons not related to assurance of the results of criticality accident analysis, and therefore, not consistent with the criteria of 10 CFR 50.36 for the content of Technical Specifications. This change is consistent with NUREG-1430.

L16 The CTS 4.10.2.d.2 requirement to test the CREVS actuation with a "control room high radiation test signal" is replaced with the phrase "actual or simulated actuation signal." This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill the surveillance requirements. OPERABILITY is adequately demonstrated in either case since the system can not discriminate between "actual" or "simulated" signals. This change is consistent with NUREG-1430.

L17 CTS 3.13.1 requires the penetration room ventilation system (PRVS) to be OPERABLE whenever reactor building integrity is required. CTS 3.6.2 requires reactor building integrity be (established and) maintained whenever the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown, i.e., enough negative reactivity to remain subcritical by 1% $\Delta k/k$ even with all rods removed and RCS temperature at ~140°F, are not met. The proposed Applicability for ITS 3.7.11 is MODES 1, 2, 3, and 4, and includes no requirements for MODE 6 (refueling shutdown condition), or for MODE 5 with the reactor coolant system otherwise open to the atmosphere.

The PRVS functions to filter reactor building leakage in a post accident environment. In MODE 5 with the reactor coolant system open to the atmosphere, no such accidents are postulated to occur. Therefore, the PRVS function is not required.

ITS 3.9.1 provides requirements for MODE 6 boron concentration. The Required Actions for ITS 3.9.1 provide protection by suspending activities that may initiate an accident and initiating restoration of the required boron concentration. These preventive measures are provided in lieu of actions to provide for mitigation of the event. Typically, the suspension of fuel movement would occur much more rapidly than the reactor building integrity could be established from an unexpected condition. Once there is no potential for an accident, there is no need to require mitigation equipment such as the PRVS. (For CTS 3.13.1 requirements per CTS 3.6.1, see DOC M13.) This change is consistent with NUREG-1430.

L18 Not used.

L19 CTS 3.3.1(I) and 3.3.4(D) require that the engineered safety features valves for the service water system (CTS 3.3.1(C)) be OPERABLE or locked in the Engineered Safeguards (ES) position whenever RB integrity is established and when the reactor is critical. NUREG 3.7.8 requires that the service water system be OPERABLE during MODES 1, 2, 3 and 4. The ES valves, which are components of the service water system, are verified OPERABLE by NUREG SR 3.7.8.2 (which is renumbered and adopted as ITS SR 3.7.7.2). In the NUREG, the ES valves may be verified OPERABLE by actuation to the correct position or by being locked, sealed or otherwise secured in position. These expanded options for ES valve verification will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

ANO-290 L20 The requirements of CTS 3.9.2 have been revised to allow the control room boundary to be opened intermittently under administrative controls, and to allow both CREVS trains to be inoperable for 24 hours if due to a control room boundary inoperability. This condition is not allowed by the CTS, and would result in an entry into the requirements of LCO 3.0.3. Requiring the unit to enter LCO 3.0.3 for this condition is excessive, as it does not provide sufficient time to attempt a repair. The proposed change is acceptable because of the low probability of a design basis accident during any given 24 hour period and because entry into the Condition is expected to be very infrequent. The allowance to have the control room boundary open intermittently is acceptable as the administrative controls that must be implemented will ensure that the control room boundary can be rapidly closed when a need for control room isolation is indicated. This change is consistent with NUREG-1430, as modified by TSTF-287, Rev. 5.

ANO-290 L21 The requirements of CTS 3.13 have been revised to allow the penetration room ventilation system (PRVS) negative pressure boundary to be opened intermittently under administrative controls, and to allow both PRVS trains to be inoperable for 24 hours if due to a PRVS negative pressure boundary inoperability. This condition is not allowed by the CTS, and would result in an entry into the requirements of LCO 3.0.3. Requiring the unit to enter LCO 3.0.3 for this condition is excessive, as it does not provide sufficient time to attempt a repair. The proposed change is acceptable because of the low probability of a design basis accident during any given 24 hour period and because entry into the Condition is expected to be very infrequent. The allowance to have the PRVS negative pressure boundary open intermittently is acceptable as the administrative controls that must be implemented will ensure that the PRVS negative pressure boundary can be rapidly closed when a need for PRVS negative pressure boundary isolation is indicated. This change is consistent with NUREG-1430, as modified by TSTF-287, Rev. 5.

ANO-292 L22. The requirements of CTS 3.4.4 have been revised to allow the turbine driven EFW pump to be inoperable in MODE 3 if the unit has not entered MODE 2 following refueling, for a period of seven days. This change is acceptable due to the minimal decay heat levels in this condition (MODE 3 if the unit has not entered MODE 2 following refueling), the redundant capabilities afforded by the EFW system (i.e., the motor driven EFW pump), the time needed to perform repairs and testing of the turbine driven pump, and the low probability of a DBA during this seven day time period that would require operation of the turbine driven pump. This change is consistent with NUREG-1430, as modified by TSTF-340, Rev. 3.

CTS DISCUSSION OF CHANGES

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

	CTS Location	New Location
3.7-31	3.3.1.C	Bases 3.7.7, LCO
	3.4.1.2, Note *	Bases 3.7.1, LCO
3.7-15	3.8.16	Bases 3.7.15, LCO
3.7-09	3.15.2	Bases 3.7.12, RA
	4.8.1.e.5	Bases 3.7.5, SR 3.7.5.3
	4.10.1.a	Bases 3.7.10, SR 3.7.10.1
	4.10.2.a	Bases 3.7.9, SR 3.7.9.1
	4.10.2.d.2	Bases 3.7.9, SR 3.7.9.3
3.7-10	4.13.1.2	Bases 3.7.8, SR
	4.13.1.3	Bases 3.7.8, SR
	4.13.1.4	Bases 3.7.8, SR
	5.2.3	Bases 3.7.11, Background

LA2 This information has been moved to the Inservice Testing (IST) Program. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Inservice Testing Program will be controlled by 10 CFR 50.55a and 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
Table 4.1-2, #4	IST Program
Table 4.1-2, #13.b	IST Program
Table 4.1-2, #14.b	IST Program
4.5.1.2.2	IST Program
4.5.2.2.2	IST Program
4.8.1.a	IST Program
4.8.1.d	IST Program

LA3 This information has been moved to the Technical Requirements Manual (TRM) or the Safety Analysis Report (SAR). This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM and the SAR will be controlled by 10 CFR 50.59 and 10 CFR 50.71, as applicable. This change is consistent with NUREG-1430.

	CTS Location	New Location
3.7-31	Figure 3.8.1	SAR Fig. 9-53
3.7-15	Table 4.1-3, #4 w/Note (9)	TRM
3.7-19	4.5.1.1.2 (b)	TRM
	4.5.2.1.2 (c) (3)	TRM
3.7-21	4.11.5	TRM

3.7-19
3.7-21

LA4 The requirements of CTS 3.12.1, "Miscellaneous Radioactive Materials Sources," and CTS 4.14, "Radioactive Materials Sources Surveillance," have been moved to the Technical Requirements Manual. The requirements specified by CTS 3.12.2 and 3.12.3 are addressed in DOC-L11 and DOC-A10, respectively. The requirements of CTS 3.12 and 4.14 are intended to assure that leakage from byproduct, source, and special nuclear material sources does not exceed allowable limits. Criteria for inclusion of requirements in the Technical Specifications are provided in 10 CFR 50.36. The requirements associated with radioactive materials sources have been evaluated with respect to the four criteria of 10 CFR 50.36, as follows:

Criterion 1

These sources are not considered to be installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

These sources are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

These sources are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4

Radioactive materials sources are not addressed in the ANO-1 Probabilistic Safety Assessment. Therefore, radioactive materials sources are not considered to be risk significant from a reactor safety point of view.

Therefore, this proposed relocation is acceptable since the requirements associated with the radioactive materials sources do not meet any of the 10 CFR 50.36 criteria for inclusion in the Technical Specifications..

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems. A1

Objectivity

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

MODES 1, 2, 3 & 4

3.7.7 Appl. & <LATER> (3.3D, 3.5, 3.6)

3.3.1 The following equipment shall be operable whenever containment integrity is established as required by Specification 3.6.1. M9

<LATER> (3.6)

(A) One reactor building spray pump and its associated spray nozzle header. — LATER

(B) One train of reactor building emergency cooling. A1

loops

3.7-31

3.7.7 LCO

(C) Two out of three service water pumps shall be operable, powered from independent essential buses, to provide redundant and independent flow paths. LA1 Bases

<LATER> (3.5)

(D) Two engineered safety feature actuated Low Pressure Injection (LPI) pumps shall be operable. — LATER

3.7-07

3.7.7 LCO

(E) Both low pressure injection coolers and their cooling water supplies shall be operable. A15

<LATER> (3.3D)

(F) Two Borated Water Storage Tank (BWST) level instrument channel shall be operable. — LATER

<LATER> (3.5)

(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open. — LATER

(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote manually operable.

3.7.7 LCO
+<LATER>
(3.5, 3.6)

(I) The engineered safety features valves associated with each of the above systems shall be operable or locked (, sealed or otherwise secured) in the ES position. L19 + LATER

~~3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350F and irradiated fuel is in the core:~~

<LATER>
(3.5)

~~(A) Two out of three high pressure injection (makeup) pumps shall be maintained operable, powered from independent essential buses, to provide redundant and independent flow paths.~~

~~(B) Engineered safety features valves associated with 3.3.2.a above shall be operable or locked in the ES position.~~

~~3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:~~ LATER

~~(A) The two core flooding tanks shall each contain an indicated minimum of 13 ± 0.4 feet (1040 ± 30 ft³) of borated water at 600 ± 25 psig.~~

~~(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.~~

~~(C) The electrically operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.~~

~~(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.~~

~~3.3.4 The reactor shall not be made critical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable.~~

<LATER>
(3.6)

~~(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.~~ LATER

~~(B) The sodium hydroxide tank shall contain a volume of ≥9,000 gallons of sodium hydroxide at a concentration >5.0 wt% and <16.5 wt%.~~

~~(C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be locked open.~~

<Add SR 3.7.7.1 with Note> M10

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3.7.7 LCO

<LATER> (3.5, 3.6)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, sealed, or otherwise secured in the ES position. L19

4 LATER

<LATER> (3.5, 3.6)

3.3.5 ~~Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~ L13

LATER

<Add 3.7.7 RA A.1 Notes 1 & 2> A4

3.7.7 RA A.1 & <LATER> (3.5, 3.6)

~~systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.~~ L13 & LATER

& <LATER> (3.3D, 3.5, 3.6)
3.7.7 RA A.1
3.7.7 RA B.1
3.7.7 RA B.2

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in ~~hot shutdown~~ MODE 3 condition within 36 hours, and, if not corrected, in ~~cold shutdown~~ MODE 5 condition within an additional 72 hours. L13 & LATER A1 M11

<LATER> (3.3D)

~~3.3.7 Exceptions to 3.3.6 shall be as follows:~~

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ LATER

<LATER> (3.5)

~~(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.~~ LATER

<LATER> (3.6)

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~ LATER

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~

<LATER> (3.6)

(E) If the conditions of Specification 3.3.4(A) cannot be met becaus· one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore the inoper. reactor building emergency cooling train to operable status with days of initial loss or be in at least hot shutdown within the n· 6 hours and in cold shutdown within the following 30 hours.

LATER

A2

Bases

The requirements of Specification 3.3.1 assure that below 350°F, adequate lo term core cooling is provided. Two low pressure injection pumps are specifi· However, only one is necessary to supply emergency coolant to the reactor in event of a loss-of-coolant accident.

The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirem assure that the required post-accident components are available for both rea· building emergency cooling and iodine removal. Specification 3.3.1 assures the required equipment is operable.

A train consists of two coolers and their associated fans which have suffici capacity to meet post accident heat removal requirements. Conservatively ea reactor building emergency cooling train consists of two fans powered from t same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.

The borated water storage tank is used for three purposes:

(A) As a supply of borated water for accident conditions.

(B) As an alternate supply of borated water for reaching cold shutdown.(2)

(C) As a supply of borated water for flooding the fuel transfer cana during refueling operation.(3)

A2

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core.

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

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A2

REFERENCES

(1) FSAR, Section 14.2.5

(2) FSAR, Section 3.2

(3) FSAR, Section 9.5.2

(4) FSAR, Section 9.3.1

(5) FSAR, Section 6.3

(6) ANO Calculation 91-E-0019-01

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~ (A1)

~~Applies to the turbine cycle components for removal of reactor decay heat.~~

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

3.7.1 APPL & <LATER> (3.4A) 3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~ (A1) & LATER

~~1. Capability to remove decay heat by use of two steam generators.~~ LATER

3.7.1 LCO *2. Fourteen ~~of the steam system~~ safety valves are operable. (L1) (M2)

See pg 40-5 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B. (A1)

~~4. (Deleted)~~ <ADD TABLE 3.7.1-1> (L1)

See pg 40-2 & See pg 40-3 5. Both main steam block valves and both main feedwater isolation valves are operable.

Reduce power per Table 3.7.1-1 -- 4 hours (M24)

3.7.1 RA A.1 3.7.1 RA B.1, B.2 & <LATER> (3.4A) 3.4.2 ~~Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours,~~ the reactor shall be placed in the hot shutdown (MODE 3) condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the ~~cold shutdown condition~~ (MODE 4) within 24 hours. (L1) (A1) (M1) (L3)

3.4.3 Two (2) EFW trains shall be operable as follows:

See pg 40-4

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<ADD 3.7.1 ACTIONS Note> (L1)

<Add 3.7.1 RA A.2> (M24)

3.7.1 LCO NOTE * Except ~~that~~ during hydrotests, with the reactor subcritical, ~~fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.~~ (LA1) BASES

See pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

37-01

(A14)

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~ A1

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

MODE 1, 2 & 3

3.7.2 APPL & <LATER> (3.4A) 3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~ A1 & LATER

~~1. Capability to remove decay heat by use of two steam generators.~~ LATER

see pg 40-1 2. Fourteen of the steam system safety valves are operable.*

see pg 40-5 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ A1

3.7.2 LCO see pg 40-3 5. Both main steam block valves and both main feedwater isolation valves are operable.

3.7.2 RA A.1, B.1, C.1, D.1 & <LATER> (3.4A) 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the ~~hot shutdown~~ MODE 3 condition within 12 hours. A1 ~~If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.~~ L4 MODE 4 L3

see pg 40-4 3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<Add 3.7.2 Cond C Note> L4

<Add 3.7.2 RA C.2> M4

see pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

see pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

LAR

A14

3.7.3

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~ (A1)

~~Applies to the turbine cycle components for removal of reactor decay heat.~~

~~Objective~~

3.7-02

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications MODES 1, 2, + 3 except when all MFIVs, Main Feed water Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are closed + deactivated or isolated by a manual valve (L23)

3.7.3 APPL

\$ <LATER> (3.4A)

3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~ (A1) \$ LATER

~~1. Capability to remove decay heat by use of two steam generators.~~ LATER

See pg 40-1 — *2. Fourteen of the steam system safety valves are operable.

See pg 40-5 — 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ (A1)

3.7-02

See pg 40-2

3.7.3 LCO

5. Both main steam block valves and (A11) both main feedwater isolation valves, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are operable. (M4)

3.7-02

3.7.3 A.1, B.1, C.1, D.1, F.1, F.2

\$ <LATER> (3.4A)

3.4.2 ~~Components~~ (72) required to be operable by ~~Specification~~ (72) 3.4.1 shall not ~~be removed~~ from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the ~~hot shutdown~~ MODE 3 condition within ~~12~~ 6 hours. ~~If the requirements of Specification 3.4.1 are not met within an additional 48 hours,~~ the reactor shall be placed in the ~~cold shutdown condition~~ MODE 4 within ~~24~~ 12 hours. (A1) (M4) (L4) (L3)

See pg 40-4

3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

3.7-02

<Add 3.7.3 Actions Note> (L4)

<Add 3.7.3 R.A. A.2, B.2, C.2, D.2, + E.1> (M4)

See pg 40-1

* Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

See pg 40-4

** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

(A14)

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~ (A1)

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

See pgs 40-1,2,3,5 — 3.4.1 The reactor shall not be heated above 280°F unless the following conditions are met:

<LATER> (3.4A) — ~~1. Capability to remove decay heat by use of two steam generators.~~ LATER

See pg 40-1 — *2. Fourteen of the steam system safety valves are operable.

see pg 40-5 — 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ (A1)

see pgs 40-2,3 — 5. Both main steam block valves and both main feedwater isolation valves are operable.

See pgs 40-1,2,3,5 — 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.7.5 LCO 3.4.3 Two (2) EFW trains shall be operable as follows:

3.7.5 APPL Note — 1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

3.7.5 APPL — 2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

See pg 40-1 — * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

SR 3.7.5.2 Note — ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

(A14)

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat. (A1)

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

<MODES 1,2,3,&4 when rely on SG>

3.7.6 APPL & <LATER> (3.4A) 3.4.1 The reactor shall not be heated above 280°F unless the following conditions are met: (M6) & LATER

1. Capability to remove decay heat by use of two steam generators. LATER

see pg 40-1 2. Fourteen of the steam system safety valves are operable.

3.7.6 LCO 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ (A1)

see pg 40-2,3 5. Both main steam block valves and both main feedwater isolation valves are operable.

3.7.6 A.2 3.7.6 B.1, B.2 & <LATER> (3.4A) 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within ~~12~~ 6 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours. (L7) & LATER (M7) (L7)

3.4.3 Two (2) EFW trains shall be operable as follows:

see pg 40-4

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<Add 3.7.6 RA A.1> (M7)

<Add SR 3.7.6.1> (M7)

see pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

see pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

(A14)

3.4.4 If the conditions specified in 3.4.3 cannot be met:

3.7.5 RA E.1

1. With the motor driven EFW pump or its associated flow path inoperable and RCS conditions above CSD and RCS temperature < 280°F and any Steam Generator relied upon for heat removal, immediately initiate action to restore the EFW train to operable status.

3.7.5 RA A.1
3.7.5 RA C.1
3.7.5 RA C.2

2 With the RCS temperature ≥ 280°F and one steam generator supply path to the turbine driven EFW pump inoperable, restore the steam generator supply path to operable status within 7 days or be in Hot Shutdown within 6 hours and reduce RCS temperature to < 280°F within the next 12 hours.

3.7.5 RA B.1
3.7.5 RA C.1
3.7.5 RA C.2

3. With the RCS temperature ≥ 280°F and one EFW pump or its associated flow path inoperable, restore the EFW train to operable status within 72 hours or be in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.

~~4. With the RCS temperature ≥ 280°F, both EFW pumps or their associated flow paths inoperable, and the Auxiliary Feedwater pump available, in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.~~ L8

3.7.5 RA D.1
3.7.5 RA D.1 Note

5. With the RCS temperature ≥ 280°F and both EFW pumps or their associated flow paths inoperable, ~~and the Auxiliary Feedwater pump unavailable,~~ immediately initiate action to restore one EFW train ~~the Auxiliary Feedwater pump~~ to operable status. LCO 3.0.3 and all other LCO Required Actions requiring mode changes are suspended until one EFW train ~~or the Auxiliary Feedwater pump~~ is restored to operable status.

<Add 10 day Completion Time for 3.7.5 RA A.1 and RA B.1> M8

<Added second entry Condition to LCO 3.7.5 Condition A + Note> L22

ANO-292

3.7.5

A2

Bases

The Emergency Feedwater (EFW) system is designed to provide flow sufficient to remove heat load equal to 3½ percent full power operation. The system minimum flow requirement to the steam generator(s) is 500 gpm. This takes into account a single failure, pump recirculation flow, seal leakage and pump wear.

In the event of loss of main feedwater, feedwater is supplied by the emergency feedwater pumps, one which is powered from an operable emergency bus and one which is powered from an operable steam supply system. Both EFW pumps take suction from tank T41B. Decay heat is removed from a steam generator by steam relief through the turbine bypass, atmospheric dump valves, or safety valves. Fourteen of the steam safety valves will relieve the necessary amount of steam for rated reactor power.

The EFW system is considered to be operable when the components and flow paths required to provide EFW flow to the steam generators are operable. This requires that the turbine driven EFW pump be operable with redundant steam supplies from each of the main steam lines upstream of the MSIVs (CV-2617 and CV-2667) and capable of supplying EFW flow to either of the tv steam generators. The motor driven EFW pump and associated flow path to th EFW system is also required to be operable. The piping, valves, instrumentation, and controls in the required flow paths shall also be operable. One EFW train, which includes the motor driven EFW pump, is required to be operable when above CSD and below 280°F with any steam generator relied upon for heat removal. This is because of reduced heat removal requirements, the short duration EFW would be required, and the insufficient steam supply available in this condition to power the turbine driven EFW pump.

When one of the required EFW trains is inoperable, action must be taken to restore the train to operable status within 72 hours. This condition includes loss of the steam supply to the turbine driven EFW pump. The 72 hour completion time is reasonable, based on the redundant capabilities afforded by the EFW system, time needed for repairs, and the low probabili of a DBA occurring during this time period.

With two EFW trains inoperable, the unit must be placed in a mode in which the LCO does not apply using the Auxiliary Feedwater pump. With RCS temperature < 280°F the Decay Heat Removal system may be placed in operatic

With both EFW trains inoperable and the Auxiliary Feedwater pump unavailab the unit is in a seriously degraded condition with only limited means for conducting a cooldown using nonsafety grade equipment. In such a conditior the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires tl action be started immediately to restore at least one EFW pump or the Auxi Feedwater pump to Operable status. LCO 3.0.3 is not applicable, as it cou force the unit into a less safe condition.

~~The OPERABILITY of the condensate storage tank with the minimum required water volume ensures that sufficient water is available to support EFW operation for both units for at least 30 minutes. This provides adequate time for the operators to manually switch the EFW suction alignment to the Service Water System (SWS), if required. The SWS provides the assured long-term source of cooling water. The required volume considers that the EFWS of both units may be aligned to T41B simultaneously. The tank is seismically qualified and the required volume is also protected from tornado missiles.~~

~~The required minimum usable volume includes an allowance for losses due to Unit 2 recirculation line flow. It does not include any allowance for instrument uncertainty or for the unusable volume due to the suction piping configuration. This volume is equivalent to a tank level of 3'-10".~~

~~The tank has sufficient capacity to support more than four hours of cooling flow for both units. This capability is not considered to be a safety related design function and is not controlled by the Technical Specifications.~~

A2

LAR

3.7.2
3.7.3

3.7-18

<LATER> (3.3D) 3.5.1.13 Two control room ventilation radiation monitoring channels shall be operable whenever the reactor coolant system is above the cold shutdown condition or during handling of irradiated fuel. LATER

<LATER> (3.3D) 3.5.1.14 The Main Steam Line Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 10^{-1} to 10^{4} mR/hr, whenever the reactor is above the cold shutdown condition. LATER

<LATER> (3.3C) 3.5.1.15 Initiate functions of the EFIC system which are bypassed at cold shutdown conditions shall have the following minimum operability conditions: LATER

a. "low steam generator pressure" initiate shall be operable when the main steam pressure exceeds 750 psig.

b. "loss of 4 RC pumps" initiate shall be operable when neutron flux exceeds 10% power.

c. "main feedwater pumps tripped" initiate shall be operable when neutron flux exceeds 10% power.

SR 3.7.2.2 Note 2 SR 3.7.3.2 Note 2 + <LATER> (3.3C) 3.5.1.16 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig. + LATER

3.7.13
3.7.14
3.7.15

<LATER> (4.0) 3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable. LATER

3.7-14 3.7-15

3.8.16 Storage in Region 2 ~~(as shown on Figure 3.8.1)~~ of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable.

3.7.15 LCO + Appl
SR 3.7.15.1
+<LATER> (4.0)
3.7.15 RA A.1 Note

LA3 SAR
LA1 Bases

3.8.17 The boron concentration in the spent fuel pool shall be maintained ~~at all times~~ at greater than 1600 parts per million.

3.7.14 LCO + Appl
L15
A8

3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable ~~as required by Specification 3.9~~.

3.7.9 LCO + Appl
3.7.10 LCO + Appl
A1

Bases

A2

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1)

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron concentration is sufficient to maintain the core keff ≤ 0.99 if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and

3.7-14

<Add 3.7.14 Cond A> M17
<Add 3.7.15 R.A. A.1> M17
<Add LCO 3.7.13 + Bases> M28

3.7.14
3.7.15

3.7-14

replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

A2
+ R
TRM

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

<LATER>
(3.9)

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface.

LATER

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours [3]; and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

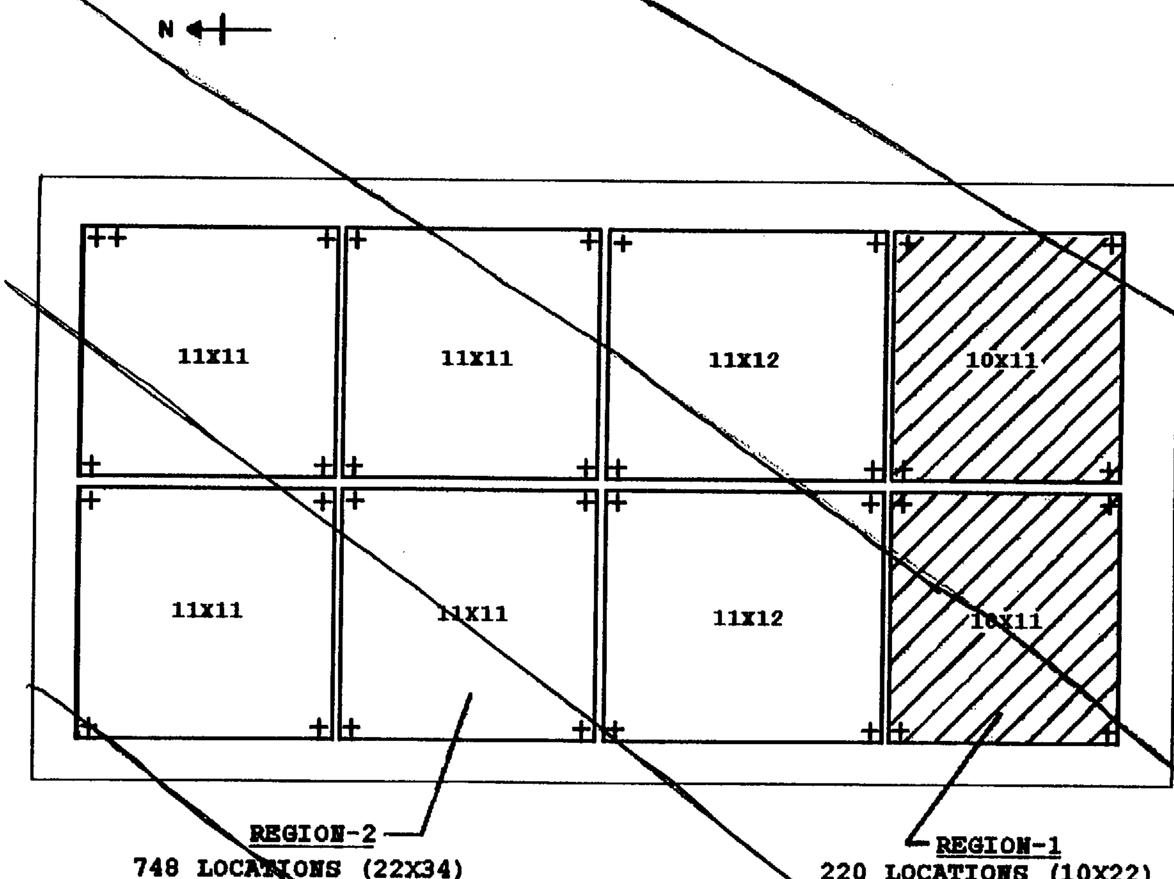
(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

Spent Fuel Pool Arrangement Unit 1

Figure 3.8.1



LA3
SR

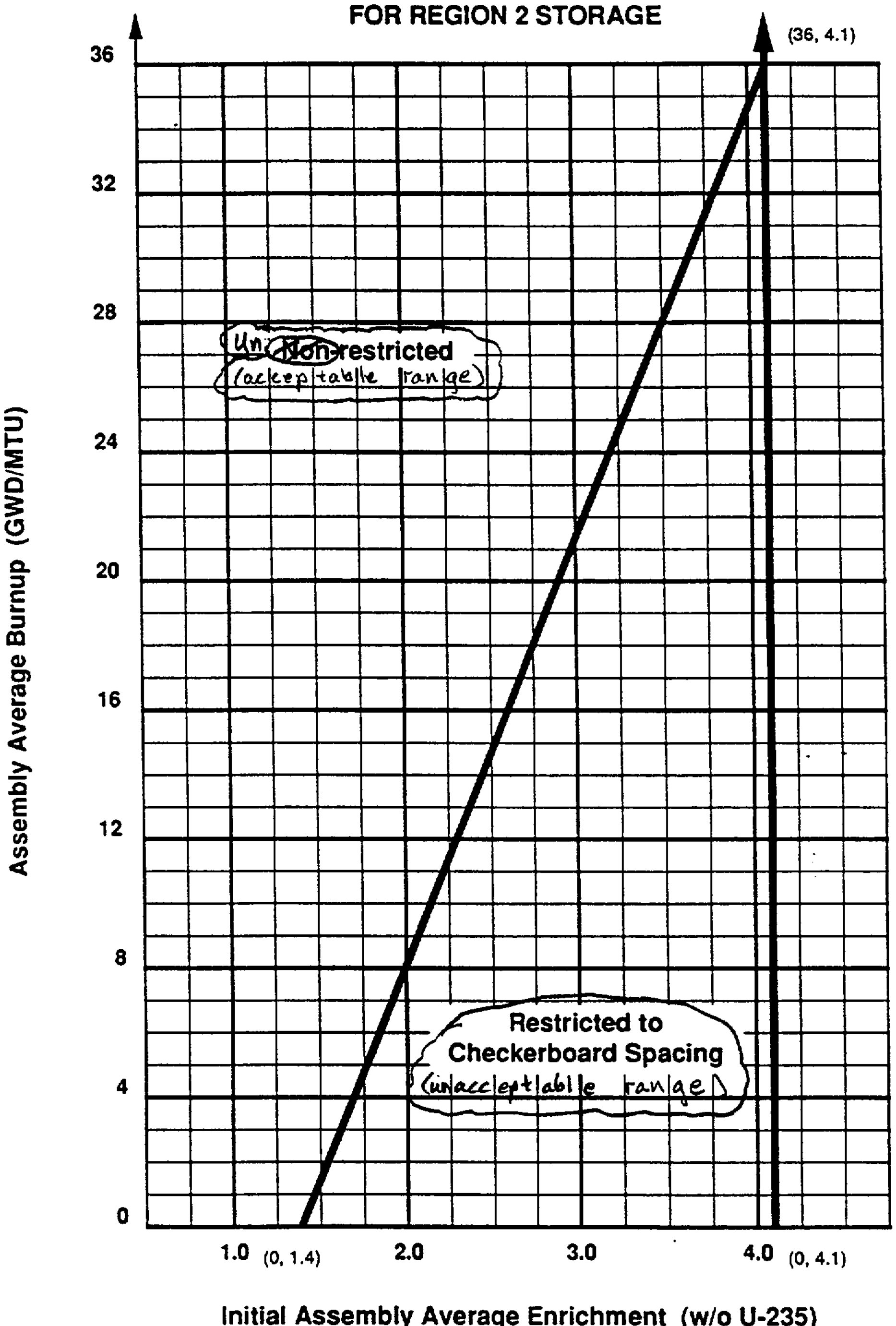
3.7.15

3.7.15

F 3.7.15-1

3.7-14

~~FIGURE 3.8.2~~
MINIMUM BURNUP VS. INITIAL ENRICHMENT
FOR REGION 2 STORAGE



edit

edit

Figure 3.7.15-1

edit

3.9 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEMS

~~**Applicability**~~

~~Applies to the operability of the control room emergency ventilation and air conditioning systems.~~

~~**Objective**~~

~~To ensure that the control room emergency ventilation and air conditioning systems will perform within acceptable levels of efficiency and reliability.~~ A1

Specification

3.9.1 Control Room Emergency Air Conditioning System

3.7.10 LCO + Appl. — 3.9.1.1 Two independent trains of the control room emergency air conditioning system shall be operable whenever the reactor coolant system is ~~above the cold shutdown condition~~ in MODES 1,2,3 or 4 or during handling of irradiated fuel. A1

3.7.10 RA A.1
3.7.10 RA B.1
3.7.10 RA B.2 — 3.9.1.2 With one control room emergency air conditioning system inoperable, restore the inoperable system to Operable status within 30 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.9.2 Control Room Emergency Ventilation System

3.7.9 LCO + Appl. — 3.9.2.1 Two independent trains of the control room emergency ventilation system shall be operable whenever the reactor coolant system is ~~above the cold shutdown condition~~ in MODES 1,2,3, or 4 or during handling ofirradiated fuel. A1

3.7.9 RA A.1
3.7.9 RA C.1, C.2 — 3.9.2.2 With one control room emergency ventilation system inoperable, restore the inoperable system to Operable status within 7 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

<Add 3.7.9 Conds D & E> M18
<Add 3.7.9 Cond F> A6
<Add 3.7.10 Conds C & D> M18
<Add 3.7.10 Cond E> A6
<Add LCO 3.7.9 LCO Note 1 and Cond B> L20
<Add LCO 3.7.9 LCO Note 2> A16

ANO-239
ANO-290

Amendment No. ~~10~~, ~~32~~, 196

Bases

The control room emergency ventilation and air conditioning system is designed to isolate the combined control rooms to ensure that the control rooms will remain habitable for Operations personnel during and following all credible accident conditions and to ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system. The design configuration of the system is based on limiting the radiation exposure to personnel occupying the control room to 5 REM or less whole body, or its equivalent, in accordance with the requirements of General Design Criteria 19 of Appendix A, 10 CFR 50. A2

Unit 1 and Unit 2 control rooms are a single environment for emergency ventilation and air conditioning concerns. Since the control room emergency ventilation and air conditioning equipment is shared between units, the plant status of both units must be considered when determining applicability of the specification.

Due to the unique situation of the shared emergency ventilation and air conditioning equipment, the components may be cross fed from the opposite unit per predetermined contingency actions/procedures. Unit 1 may take credit for operability of these systems when configured to achieve separation and independence regardless of normal power and/or service water configuration. This will be in accordance with pre-determined contingency actions/procedures.

The control room emergency ventilation system consists of two independent filter and fan trains, two independent actuation channels and the Control Room isolation dampers. The control room dampers isolate the control room within 10 seconds of receipt of a high radiation signal.

If the actuation signal can not start the emergency ventilation recirculation fan, operating the affected fan in the manual recirculation mode and isolating the control room isolation dampers provides the required design function of the control room emergency ventilation system to isolate the combined control rooms to ensure that the control rooms will remain habitable for operations personnel during and following accident conditions. This contingency action should be put in place immediately (within 1 hour) to fully satisfy the design functions of the control room emergency ventilation system.

The control room emergency air conditioning system (CREACS) provides temperature control for the control room following isolation of the control room. It is manually started from the Unit 2 Control Room. The CREACS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment.

With both trains of the control room emergency ventilation and/or emergency air conditioning inoperable, the function of the control room emergency air systems have been lost, requiring immediate action to place the reactor in a condition where the specification does not apply.

3.10 SECONDARY SYSTEM ACTIVITY

Applicability

~~Applies to the limiting conditions of secondary system activity for operation of the reactor.~~

A1

Objective

~~To limit the maximum secondary system activity.~~

Specification

3.7.4 LCO
3.7.4 RA A.1
3.7.4 RA A.2

The I-131 dose equivalent of the radioiodine activity in the secondary coolant shall not exceed 0.17 μCi/gm. With the secondary coolant activity in excess of 0.17 μCi/gm I-131, be in at least Hot Standby MODE 3 within 6 hours and in Cold Shutdown MODE 5 within the following 30 hours.

M14

Bases

A1

A2

~~For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside containment and a loss of load incident were considered.~~

~~The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released.~~

~~The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for Specification 3.1.4.1 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for Specification 3.1.4.1. For the less probable accident of a steam line break, the assumption is made that a loss of 1 x 10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident.~~

3.11 EMERGENCY COOLING POND

~~Applicability~~

~~Applies to the emergency cooling pond.~~

~~Objective~~

~~To assure the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.~~

A1

Specification

37-09

3.7.8 LCO 3.11.1 The emergency cooling pond shall be operable ~~whenever containment integrity is established as required by Specification 3.6.1~~ ~~with:~~ MODES 1, 2, 3 & 4 M13

3.7.8 Appl.

SR 3.7.8.1 — 1. A minimum contained water volume of 70 acre-feet (equivalent to an indicated water level of 5 feet).

SR 3.7.8.2

SR 3.7.8.3 — 2. An average water temperature of ≤ 100°F.

3.7.8 RA A.1 3.11.2 With the requirements of Specification 3.11.1 not satisfied, be in the hot shutdown condition within 6 hours and in the cold shutdown condition within the following 30 hours.

3.7.8 RA A.2

~~Bases~~

~~The requirements of Specification 3.11.1 provide for sufficient water inventory in the emergency cooling pond to mitigate within acceptable limits the effects of a DBA with a concurrent failure of the Dardanelle Reservoir. The minimum water depth takes into account (1) water loss from evaporation due to heat load and climatological conditions, (2) pond bottom irregularities, (3) suction pipe level at the pond and (4) operator action in transferring the service water system from the Dardanelle Reservoi Operator action is credited in the inventory analysis during the transfer of the service water system to the pond. Specifically, pump returns are transferred to the pond shortly after a loss of lake event and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the pond, lake water is pumped into the pond, increasing level. This additional water is required, along with that maintained by Technical Specifications, to ensure a 64.5 inch pond depth, which corresponds to a 30 day supply of cooling wate~~

~~The values are based on worst case initial conditions which could be present considering a simultaneous normal shutdown of Unit 1 and emergency shutdown of Unit 2 following a LOCA in Unit 2, using the ECP as a heat sink. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions.~~

A2

3.7-19

LA4
TRM

3.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source and special nuclear radioactive material sources does not exceed allowable limits.

Specification

3.12.1 The source leakage test performed pursuant to Specification 4.14 shall be capable of detecting the presence of 0.005 μCi of radioactive material on the test sample. If the test reveals the presence of 0.005 μCi or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired, or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 μCi or less of beta and/or gamma emitting material or 5 μCi or less of alpha emitting material. The provisions of Specification 3.0.3 are not applicable.

3.12.2 A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.12.5 within 90 days if source leakage tests reveal the presence of ≥0.005 microcuries of removable contamination. L11

3.12.3 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times. A10

3.13 PENETRATION ROOM VENTILATION SYSTEM

Applicability

Applies to the operability of the penetration room ventilation system. A1

Objective

To ensure that the penetration room ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.7.11 LCO + Appl
SR 3.7.11.2

3.13.1 Two independent circuits of the penetration room ventilation system shall be operable ~~whenever reactor building integrity is required~~ MODES 1, 2, 3, and 4 with the following performance capabilities: M13 L17

<LATER> (5.0) LATER

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flow (± 10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis from the charcoal adsorber banks shall show the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

c. Fans shall be shown to operate within ± 10% of design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (± 10%).

e. Air distribution shall be uniform within ± 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system.

SR 3.7.11.3 f. Each circuit of the system shall be capable of automatic initiation.

3.7.11 RA A.1

3.13.2 If one circuit of the penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days provided that during such seven days all active components of the other circuit shall be operable. A1

3.7.11 RA C.2

3.13.3 If the requirements of Specifications 3.13.1 and 3.13.2 cannot be met, the reactor shall be placed in the cold shutdown condition MODE 5 within 36 hours. A1

<Add 3.7.11 RA C.1> M20

<Add 3.7.11 Cond. B and LCO Note> L21

AND-290

AND-334

37.11

A2

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of sealed penetration rooms, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building engineered safety features signal and initially requires no operator action. Each filter train is constructed with a prefilter, a HEPA filter and a charcoal adsorber in series. The design flow rate through each of these filters is 2000 scfm, which is significantly higher than the 1.25 scfm maximum leakage rate from the reactor building at a leak rate of 0.1% per day.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should ensure a radioactive methyl iodide removal efficiency of a least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If one circuit of the penetration room ventilation system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue for a limited period of time while repairs are being made.

ANO-334

3.15 FUEL HANDLING AREA VENTILATION SYSTEM

A1

~~Applicability~~

~~Applies to the operability of the fuel handling area ventilation system.~~

~~Objective~~

~~To ensure that the fuel handling area ventilation system will perform within acceptable levels of efficiency and reliability.~~

Specification

OPERABLE and — M3

3.7.12 LCO + Appl — 3.15.1 The fuel handling area ventilation system shall be in operation whenever irradiated fuel handling operations are in progress in the fuel handling area of the auxiliary building and shall have the

SR 3.7.12.2 — following performance capabilities:

<LATER> (5.0) — LATER

a. ~~The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.~~

b. ~~The results of laboratory carbon sample analysis shall show the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.~~

c. ~~Fans shall be shown to operate within ±10% design flow.~~

d. ~~The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%).~~

e. ~~Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system.~~

3.7.12 RA A.1 — 3.15.2 If the requirements of Specification 3.15.1 cannot be met, irradiated fuel movement shall not be started. ~~Any irradiated fuel assembly movement in progress may be completed.~~ The provisions of — LA1 Bases

3.7.12 ACTIONS Note — Specification 3.0.3 are not applicable.

A2

~~Bases~~

~~The fuel handling area ventilation system is designed to filter the auxiliary building atmosphere during fuel handling operations to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing two exhaust fans and a filter train. The fans are redundant and only one is required to be operating. The filter train consists of a prefilter, a HEPA filter and a charcoal adsorber in series.~~

AND 334

A2

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine absorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should ensure a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

ANO-334

3.7.1
3.7.7

Table 4.1-2
Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.1)	1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown	LATER
	2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions	
<LATER> (3.4B)	3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month	LATER
SR 3.7.1.1	4. Main Steam Safety Valves <Add SR 3.7.1.1, Note 1>	Setpoint	Four Valves Every 18 Mo	LA2 IST; L2
	5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown	R TRM
<LATER> (3.4B)	6a. Reactor Coolant System Leakage	Evaluate	Daily	LATER
	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	
	7. Emergency-powered Pressurizer Heaters	Power availability	Daily	
		Heater capacity functional test	Every 18 Months	
<LATER> (3.6)	8. Reactor Building Isolation Trip	Functioning	Every 18 Months	LATER
SR 3.7.7.2	9. Service Water Systems	Functioning	Every 18 Months	M12
	10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool	R TRM

<LATER> (3.1) 1/ Same as tests listed in Section 4.7 — LATER

Notes:

<LATER> (3.4B) — LATER

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

<Add SR 3.7.2.2 & SR 3.7.3.2
with Note 1 with Note 1> — M5

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.4B)	~~11. Decay heat removal system isolation valve automatic closure and isolation system~~	~~Functioning~~	~~Each Refueling Shutdown~~	LATER
<LATER> (5.0)	~~12. Flow limiting annulus on main feedwater line at reactor building penetration~~	~~Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.~~	~~One year, two years, three years, and every five years thereafter measured from date of initial test.~~	LATER
SR 3.7.2.1	13. Main steam isolation valves <Add Note for SR 3.7.2.1>	~~a. Exercise through approximately 10% travel~~	~~a. Quarterly~~	L5, M5
		b. Cycle	~~b. Every 18 months~~	LA2 IST, M4
SR 3.7.3.1	14. Main feedwater isolation valves <Main feedwater block Valves, Low Load Feed Water Control valves, Startup Feed water Control valves> <Add Note for SR 3.7.2.1>	~~a. Exercise through approximately 5% travel~~	~~a. Quarterly~~	L5, M5
		b. Cycle	~~b. Every 18 months~~	LA2 IST, M5
<LATER> (3.4A)	~~15. Reactor internals vent valves~~	~~Demonstrate operability by: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).~~	~~Each refueling shutdown~~	LATER

3.7-02

3.7.4
3.7.13

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Margin	Item	Test	Frequency	Margin
⟨LATER⟩ (3.4B)	1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
		b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
⟨LATER⟩ (3.1)		c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
⟨LATER⟩ (3.4A)		d. Dissolved Gases	d. Weekly (7)	LATER
		e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
⟨LATER⟩ (3.9)		f. Boron Concentration	f. 3 times/week	LATER & Ⓡ TRM
⟨LATER⟩ (3.4B)		g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)	LATER
⟨LATER⟩ (3.5)	2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER
	3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	M16
3.7-14 SR 3.7.14.1	4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LA3 TRM
SR 3.7.4.1	5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10	Ⓡ TRM
		b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	
⟨LATER⟩ (3.6)	6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	LATER

⟨LATER⟩ (3.4B) — LATER

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

3.7.4
3.7.13

<LATER> (3.4B)

(2) A radiochemical analysis shall consist of the quantitative measurement the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activitie: for the individual isotopes shall be used in the determination of $\bar{E}$. , radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activi· shall be performed if the measured gross activity changes by more than µCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall as given in "Table of Isotopes" (1967) and beta energy per disintegrat· shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes. — LATER

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 µCi/gm from the previous measured level.

SR 3.7.4.1 & <LATER> (3.4B)

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity. & LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. — R TRM

<LATER> (3.4B)

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less th: 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above. — LATER

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prio to any reactor criticality and analyzed for radioactive iodines of I-1: through I-135 and gross radioactivity as well as the coolant sample an‹ analyses required by above.

3.7.4 Appl. & <LATER> (3.1 & 3.4B)

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition. & R TRM & LATER

<LATER> (3.4A)

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition. — LATER

3.7.13 Appl.

(9) Required only when fuel is in the pool ~~and prior to transferring fuel to the pool.~~ LA3 TRM

3.7.4 Appl.

(10) Not required when not generating steam in the steam generators. — A5 & R TRM

<LATER> (3.4B)

(11) The following shall be required until the end of Cycle 2 operation: — LATER

a. Gross radioiodine shall be determined at least three times per wee! during power operation.

4.5 EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

⟨LATER⟩ (3.5)

LATER

4.5.1 ~~Emergency Core Cooling Systems~~

~~Applicability~~

~~Applies to periodic testing requirement for emergency core cooling systems.~~

~~Objective~~

~~To verify that the emergency core cooling systems are operable.~~

~~Specification~~

4.5.1.1 ~~System Tests~~

4.5.1.1.1 ~~High Pressure Injection System~~

(a) ~~Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation.~~

(b) ~~The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.~~

4.5.1.1.2 ~~Low Pressure Injection System~~

SR 3.7.7.2 & ⟨LATER⟩ (3.5)

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

(1) ~~A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.~~

SR 3.7.7.2

(2) Verification of the engineered safeguard function of the service water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers. A1

& ⟨LATER⟩ (3.5)

(b) ~~The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.~~ LA3 TRM & LATER

3.7.7

<LATER>
(3.5)

4.5.1.1.3 Core Flooding System

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During this test, verification shall be made that the check valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

LATER

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

Approximately quarterly, the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ± 10% of the initial level of performance as determined using test flow paths.

4.5.1.2.2 Valves - Power Operated

(a) At intervals not to exceed three months, each engineered safety feature valve in the emergency core cooling systems and each engineered safety feature valve associated with emergency core cooling in the service water system which are designed to open in the event of a LOCA shall be tested to verify operability.

(b) The acceptable performance of each power operated valve will be that motion is indicated upon actuation by appropriate signals.

LA2
1ST
& LATER

& <LATER>
(3.5)

Bases

The emergency core cooling systems are the principle reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The high pressure injection system under normal operating conditions has one pump operating. At least once per month, operation will be rotated to another high pressure injection pump. This will help verify that the high pressure injection pumps are operable.

A2

The requirements of the service water system for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis will verify that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the borated water storage tank recirc line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

<LATER> (3.6)

(2) Verifying that each operational cooling fan operates for at least 15 minutes. — LATER

SR 3.7.7.2 & <LATER> (3.6)

(c) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the reactor building emergency cooling operation.

SR 3.7.7.2

(2) Verification of the engineered safety features function of the service water system which supplies the reactor building emergency coolers shall be made to demonstrate operability of the coolers. — A1

SR 3.7.7.2 & <LATER> (3.6)

(3) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly. — LA3 TRM & LATER

<LATER> (3.6)

4.5.2.2 Component Tests — LATER

4.5.2.2.1 Pumps

At intervals not to exceed 3 months the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

4.5.2.2.2 Valves

At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building emergency cooling system and each engineered safety features valve associated with reactor building emergency cooling in the service water system shall be tested to verify that it is operable. — LA2 IST

Bases — A2

The reactor building emergency cooling system and reactor building spray system are redundant to each other in providing post-accident cooling of the reactor building atmosphere to prevent the building pressure from exceeding the design pressure. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building emergency cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable engineered safeguard equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.

A2

Addition of a biocide to service water is performed during reactor building emergency cooler surveillance to prevent buildup of Asian clams in the coolers when service water is pumped through the cooling coils. This is performed when service water temperature is between 60F and 80F since in this water temperature range Asian clams can spawn and produce larva which could pass through service water system strainers.

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or smoke can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the reactor building emergency cooling system are arranged so that they can be visually inspected. The cooling fans and coils and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.

As the reactor building fans are normally operating, starting for testing is unnecessary for those verified to be operating.

<u>Reference</u>

FSAR, Section 6

4.8 EMERGENCY FEEDWATER PUMP TESTING

Applicability

Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.

Objective

To verify that the emergency feedwater pump and associated valves are operable.

A1

Specification

4.8.1 Each EFW train shall be demonstrated operable:

L9

a) By verifying on a STAGGERED TEST BASIS:

M26

SR 3.7.5.2 & Note

1. at least once per 31 days or within 24 hours after reaching the Hot Shutdown condition following a plant heatup and prior to criticality, that the turbine-driven pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psig at a flow of ≥ 500 gpm through the test loop flow path.

L10

LA2 IST

SR 3.7.5.2

2. at least once per 31 days by verifying that the motor driven EFW pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psi at a flow of ≥ 500 gpm through the test loop flow path.

L9

LA2 IST

SR 3.7.5.1

b) At least once per 31 days by verifying that each valve (manual, power operated or automatic) in each EFW flowpath that is not locked, sealed, or otherwise secured in position is in its correct position.

A3

SR 3.7.5.5

c) Prior to ~~relying upon any steam generator for heat removal~~ entering MODE 2 whenever the plant has been in ~~CSD or less~~ MODE 5, MODE 6, or defueled for > 30 days, verify proper alignment of each manual valve in each required EFW flow path, which if mispositioned may degrade EFW operation, from the 'Q' condensate storage tank to each steam generator.

L12

A1

d) At least once per 92 days by cycling each motor-operated valve in each flowpath through at least one complete cycle.

LA2 IST

e) At least once per 18 months by functionally testing each EFW train and:

SR 3.7.5.3

1. Verifying that each automatic valve in each flowpath actuates automatically to its correct position on receipt of an actual or simulated actuation signal.

<Add SR 3.7.5.3, Note> L10

<Add SR 3.7.5.4, Note> L10

SR 3.7.5.3
SR 3.7.5.4

2. Verifying that the automatic steam supply valves associated with the steam turbine driven EFW pump actuate to their correct positions upon receipt of an actual or simulated actuation signal. ~~This test is not required to be performed until 24 hours after reaching the Hot Shutdown condition.~~ M27

SR 37.5.4

3. Verifying that the motor-driven EFW pump starts automatically upon receipt of an actual or simulated actuation signal.

SR 3.7.5.6

4. Verifying that feedwater is delivered to each steam generator using the electric motor-driven EFW pump.

SR 3.7.5.3

5. Verifying that the EFW system can be operated manually by over-riding automatic signals to the EFW valves. LA1

Bases

A2

Bases

The monthly testing frequency will be sufficient to verify that both emergency feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps. The cycling of the emergency valves assures valve operability when called upon to function. Testing of the turbine dri EFW pump is delayed until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perfor the test at 280°F. Testing may occur at a lower steam generator pressure if operational experience shows that sufficient steam pressure to perform the exists.

Surveillance Requirement 4.8.1.c ensures that the EFW system is properly al by verifying the flow paths to each steam generator prior to relying upon a steam generator for heat removal after more than 30 days in Cold Shutdown o below. Operability of the EFW flow paths must be demonstrated before suffi core heat is generated that would require the operation of the EFW system o a subsequent shutdown. This requirement is reasonable, based on engineerin judgment, in view of other administrative controls to ensure that the flow are operable. To further ensure EFW system alignment, flow path operabilit verified following extended outages to determine no misalignment of valves occurred. This SR ensures that the flow path from the 'Q' CST to the steam generators is properly aligned.

The functional test, performed once every 18 months, will verify that the flow path to the steam generators is open and that water reaches the steam generators from the emergency feedwater system. The test is done during shutdown to avoid thermal cycle to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater.

The automatic actuation circuitry testing and calibration will be performed per Surveillance Specification 4.1, and will be sufficient to assure that this circuitry will perform its intended function when called upon.

3.7.9
3.7.10

4.10 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the control room emergency ventilation and air conditioning systems. A1

Objective

To verify an acceptable level of efficiency and operability of the control room emergency ventilation and air conditioning systems.

Specification

4.10.1 Each train of control room emergency air conditioning shall be demonstrated Operable:

SR 3.7.10.1

a. At least once per 31 days ~~on a staggered test basis~~ by: LA1 BASES

1. Starting each unit and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤84°F D.B.

SR 3.7.10.2

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ±10%.

4.10.2 Each Control Room Emergency Ventilation System shall be demonstrated Operable:

SR 3.7.9.1

a. At least once per 31 days ~~on a Staggered Test Basis~~ by ~~initiating, from the Control Room, flow through the HEPA filters and charcoal adsorbers and~~ verifying that the system operates for at least 15 minutes. LA1 Bases

SR 3.7.9.2 + <LATER> (5.0)

b. At least once per 18 months or 1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or 2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10%. LATER

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

   a. ≤2.5% for 2 inch charcoal adsorber beds, or

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

3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

SR 3.7.9.2 + <LATER> (5.0)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, 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. ≤ 2.5% for 2 inch charcoal adsorber beds, or

2. ≤ 0.5% for 4 inch charcoal adsorber beds.

+ LATER

SR 3.7.9.2
SR 3.7.9.3
+ <LATER (5.0)>

d. At least once per 18 months by:

+ LATER

SR 3.7.9.2 + <LATER (5.0)>

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches of water while operating at a flowrate of 2000 cfm ±10%.

L16

SR 3.7.9.3

2. Verifying that on a Control Room high radiation test signal, the system automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

LA1 Bases

SR 3.7.9.2 + <LATER> (5.0)

e. After each complete or partial replacement of the HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

+ LATER

Bases

A2

The purpose of the control room emergency ventilation system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with 100 percent capacity filter trains which consist of a prefilter, high efficiency particulate filters, charcoal adsorbers and a fan.

Since the emergency ventilation system is not normally operated, a periodic test is required to insure operability when needed. During this test the system will be inspected for such things as water, oil, or other foreign material; gasket deterioration, adhesive deterioration in the HEPA units; and unusual or excessive noise or vibration when the fan motor is running. Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

<Added SR 3.7.9.4> M29

3.7-12

Bases (Continued)

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52. A2

The operability of the control room emergency air conditioning Systems ensure that the ambient air temperature does not exceed the allowable temperature for the equipment and instrumentation cooled by this system and the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.

Operation of the systems for 15 minutes every month will demonstrate operability of the emergency ventilation and emergency air conditioning systems. All dampers and other mechanical and isolation systems will be shown to be operable.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

4.11 PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE

~~Applicability~~

~~Applies to the surveillance of the penetration room ventilation system.~~ A1

~~Objective~~

~~To verify an acceptable level of efficiency and operability of the penetration room ventilation system.~~

Specification

SR 3.7.11.2 & (LATER) (5.0)

~~4.11.1 At intervals not to exceed 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).~~ LATER

~~4.11.2 Initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.~~

SR 3.7.11.3

4.11.3 At intervals not to exceed 18 months, automatic initiation of the penetration room ventilation system shall be demonstrated.

SR 3.7.11.2 & (LATER) (5.0)

~~4.11.4a The tests and sample analysis of Specification 3.13.1a, b, & c. shall be performed at intervals not to exceed 18 months or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.~~ LATER

~~b. Cold DOP testing shall also be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.~~

~~c. Halogenated hydrocarbon testing shall also be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.~~

SR 3.7.11.1

4.11.5 Each circuit shall be operated at least 1 hour every month. ~~This test shall be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal.~~ L14 LA3 TRM

Bases

(A2)

The penetration room ventilation system is designed to collect and process potential reactor building penetration room leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of a sealed penetration room, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building pressure engineered safety features signal and initially requires no operator action.

Since the system is not normally operated, a periodic test is required to show that the system is available for its engineered safety features function. During this test the system will be inspected for such things as water, oil, or other foreign material, gasket deterioration in the HEPA units, and unusual or excessive noise or vibration when the fan motor is running.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per 18 months to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to ASTM D3803-1989.

Operation of the system each month for 1 hour will demonstrate operability of the active system components and the filter and adsorber system. If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

ANO-334

4.13 EMERGENCY COOLING POND

~~Applicability~~

~~Applies to the emergency cooling pond.~~

~~Objective~~

~~To verify the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.~~

A1

Specification

4.13.1 The emergency cooling pond shall be determined operable:

SR 3.7.8.1 1. At least once per 24 hours by verifying the pond's indicated water level is ≥ 5 feet.

3.7.10

SR 3.7.8.2 + Note 2. At least once per 24 hours during the period from June 1 through September 30 by verifying that the pond's average water temperature ~~at the point of discharge from the pond~~ is within its limit. LA1 Bases

SR 3.7.8.3 3. At least once per 12 months by ~~making soundings of the pond and~~ verifying ~~an average depth of 5 feet and that~~ the contained water volume of the pond is within its limit. LA1 Bases

3.7-11

SR 3.7.8.4 4. ~~At least once per 12 months~~ by ~~a visual inspection of the loose stone (riprap) placed on the banks of the pond and of the concrete slab spillway and verifying~~ that the earth portions of the stone covered embankments and the spillway: LA1 Bases

1. Have not been eroded or undercut by wave action, and

2. Do not show apparent changes in visual appearance or other abnormal degradation from their as built condition.

~~Bases~~

~~The requirements of Specification 4.13 provide for verification of a sufficient water inventory in the emergency cooling pond to handle a DBA with a concurrent failure of the Dardanelle Reservoir. This specification ensures that Specification 3.11.1 is met. Monitoring temperature only during the period June 1 through September 30 of each year ensures that, during the hot summer months, the pond temperature limit is not exceeded. During other periods of the year the pond temperature will not have the potential to reach the temperature limit. Soundings are performed to ensure the water volume is within limits and that the indicated level is indicative of an equivalent water volume for accident mitigation. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions. Visual inspections are performed to ensure any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation sh be performed by a qualified engineer of any apparent changes in visual appearance or other abnormal degradation to determine operability.~~

A2

3.7

3.7-21

LA4

TRM

4.14 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Test for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferrer indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed startup source shall be leak tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4. The periodic leak test does not apply to the boronometer source. This source shall be tested for leakage at least once per 18 months.

<Add SR 3.7.12.1> M25

4.17 FUEL HANDLING AREA VENTILATION SYSTEM SURVEILLANCE

A1

Applicability

Applies to the surveillance of the fuel handling area ventilation system.

Objective

To verify an acceptable level of efficiency and operability of the fuel handling area ventilation system.

Specification

SR 3.7.12.2 & <LATER> (5.0)

LATER

4.17.1 At intervals not to exceed 18 months, pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%).

4.17.2 Initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.

4.17.3 a. The tests and sample analysis of Specification 3.15.1.a, b, & c shall be performed within 720 system operating hours prior to irradiated fuel handling operations in the auxiliary building, and prior to irradiated fuel handling in the auxiliary building following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

L14

4.17.4 The system shall be operated for at least 10 hours prior to initiation of irradiated fuel handling operations in the auxiliary building if it has not been operated for at least 10 hours within the previous 30 days.

A2

Bases

Since the fuel handling area ventilation system may be in operation when fuel is stored in the pool but not being handled, its operability must be verified before handling of irradiated fuel. Operation of the system for 10 hours before irradiated fuel handling operations and performance of Specification 4.17.3 will demonstrate operability of the active system components and the filter and adsorber systems.

(A2)

~~Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop and air distribution should be determined once every 18 months to show system performance capability.~~

~~The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to ASTM D3803-1989.~~

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(LATER) (4.0)

~~assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks. [1]~~

LATER

~~5.2.2 Reactor Building Isolation System~~

~~Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. [2]~~

~~5.2.3 Penetration Room Ventilation System~~

~~This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. [3]~~

LA1 Bases

(LATER) (4.0)

~~REFERENCES:~~

~~(1) FSAR Section 5.1~~

~~(2) FSAR Section 5.2.5~~

~~(3) FSAR Section 6.5~~

LATER

LA1 Bases

3.7

6.12.5 Special Reports

<LATER> (5.0) Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification. LATER

a. Deleted A1

<LATER> (3.3D) b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LATER

c. Deleted A1

<LATER> (5.0) d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18. LATER

e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2. L11

f. Deleted

g. Deleted A1

h. Deleted

i. Deleted

<LATER> (3.8) j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H. LATER

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1

<LATER> (3.3D) l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1 LATER

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

<u>"A" - Administrative changes to requirements</u>:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.7: Plant Systems

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

3.7 L1

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The number of main steam safety valves (MSSVs) required to be OPERABLE is reduced based 3.7-17 on the number required to perform the safety function at specific power levels. In addition, separate condition entry for each inoperable MSSV is allowed. The MSSVs are considered as potential event initiators through inadvertent opening and depressurization of the secondary system. Current Technical Specifications only require 14 MSSVs to be OPERABLE. The control of inoperable MSSVs will be the same as controls for the currently allowed inoperable MSSV. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. The MSSVs also provide overpressurization protection for decreased heat removal events. Requirements are included to reduce reactor power well within the time frame during which the MSSVs were previously allowed to be inoperable with no action. Reducing the high flux trip setpoint provides assurance that sufficient MSSV capacity is available to mitigate the effects of an overpressure event during operation with less than 14 MSSVs. A reduced power reactor trip will result in consequences within those of previously analyzed 3.7-17 accidents. Allowing a separate Condition entry for each MSSV does not involve a significant increase in the consequences of an accident since appropriate compensatory measures are contained in the proposed ITS requirements. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are to be taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety for MSSVs is based on the capability to prevent an overpressurization event. The methodology for determination of the number of MSSVs includes a reduced reactor power trip setpoint to limit the thermal energy required to be relieved. Therefore, the change does not involve a significant reduction in the margin of safety.

3.7 L2

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The MSSVs are considered as potential event initiators through inadvertent opening and depressurization of the secondary system. The control of inoperable MSSVs will be the same as controls for the currently allowed inoperable MSSV. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. The main steam safety valves (MSSVs) are proposed to be allowed to be setpoint tested in MODE 3 during startup. Currently the MSSVs are required to have met the surveillance requirements, including setpoint testing, prior to heating the reactor above 280°F. The MSSVs will still be required to be OPERABLE with their setpoints properly adjusted (prior to heatup above 280°F). Plant experience with setpoint adjustment provides reasonable expectation that the MSSVs are capable of performing their safety function to prevent an overpressurization event. Also, performing this test at conditions closer to actual operating conditions minimizes any potential for inaccuracy due to differences between test conditions and operating conditions. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure compliance with the limiting condition for operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety for MSSVs is based on the capability to prevent an overpressurization event. The plant experience with MSSV setpoint adjustment is incorporated into procedures which provide assurance of proper adjustment, and if needed, confirmation of the setpoint early in the startup to maintain the capability of the MSSVs to perform their function. Therefore, the change does not involve a significant reduction in the margin of safety.

3.7 L3

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The change in the Required Action does not result in any hardware changes. The change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed). The change provides consistency between the Required Actions and Applicable conditions for the LCO. Further, the change of Required Actions does not significantly increase the consequences of an accident because the change does not affect the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters, from that resulting from the original analysis.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for unit conditions during which analysis assumes the equipment to function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The Required Actions are revised to be consistent with the Applicability for the equipment. Therefore, the change does not involve a significant reduction in the margin of safety.

3.7 L4

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

This change does not result in any hardware changes. The closed main steam isolation valves (MSIVs) or main feedwater isolation valves (MFIVs) are not assumed to be an initiator of any analyzed event. The consequences of any event occurring with the valves already closed will not be significantly increased since the isolation valves are already in their required position, and the closure time is zero which is less than the assumed closure time if the valves were open.

3.7-02

The Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve associated with each MFIV provide a redundant means of isolating the main feedwater flow in the event of a main steam line break (MSLB). Revising the allowed outage time to restore an inoperable MFIV to Operable status from 24 hours to a Completion Time of 72 hours is acceptable due to the presence of this redundant isolation capability, and due to the low probability of an MSLB occurring during any specific 72 hour period.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change allows continued operation in these conditions since the valves have already performed their safety function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The pertinent margin of safety associated with the isolation valve closure is provided by the time associated with the closure of the valves following an event. Since inoperable valves will be closed and maintained closed, the required closure time will be met and the change does not result in a significant reduction in a margin of safety.

3.7-02

Revising the allowed outage time from 24 hours to 72 hours does not involve a significant reduction in a margin to safety due to the presence of a redundant means to isolate the main feedwater flow in the event of an MSLB, and due to the presence of appropriate compensatory actions in the event an MFIV and any associated Main Feedwater Block Valve, Low Load Feedwater Control Valve or Startup Feedwater Control Valve is inoperable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L5

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs) are used to support mitigation of the consequences of an accident; however, they are not considered the initiator of any previously analyzed accident. As such the proposed revision of the Surveillance Frequency will not significantly increase the probability of any accident previously evaluated. Since the function of the isolation valves continues to be verified on a periodic basis, and the valves continue to be required to be OPERABLE, the change of the Surveillance Frequency will not reduce the capability of required equipment to mitigate the event. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety associated with the MSIV and MFIV is provided by their closure capability following an event. Since testing will continue to confirm the required parameters for these valves, this change does not involve a significant reduction in a margin of safety.

3.7 L6

Not Used.

3.7 L7

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters, from that of the analyses considering the original Completion Time.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

3.7 L8

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The current Technical Specifications require a shutdown if both emergency feedwater (EFW) pumps or their associated flow paths are inoperable, and the nonsafety related auxiliary feedwater (AFW) pump is available. However, the AFW pump is not required to be tested or verified to be available on any periodic basis. A change is proposed to not require the shutdown depending on the nonsafety related equipment, but rather leave this option to the licensee based on current knowledge of plant equipment and capability. Inoperable EFW equipment is not considered as an initiator of any previously evaluated accident. Therefore, the change does not increase the probability of an accident previously evaluated. Previously evaluated accidents do not depend on the nonsafety related AFW pump to mitigate consequences. However, as with any system, if it is available to mitigate an accident, it may be used. Therefore, the proposed change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Required Action to initiate restoration of reliable safety related equipment has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and the potential impact of failure of nonsafety related equipment. Therefore, the propose Required Action does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.7 L9

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The emergency feedwater (EFW) pumps are used to support mitigation of the consequences of an accident; however, EFW is not considered as the initiator of any event. Therefore, the proposed revision will not increase the probability of any accident previously evaluated. Since the function of the EFW pumps continues to be verified on a periodic basis, and the pumps continue to be required to be OPERABLE, the change of the Surveillance Frequency will not reduce the capability of required equipment to mitigate the event. As discussed in NUREG-1366, Section 9.1, industry studies indicate that EFW pump testing on a monthly basis may be contributing to equipment unavailability and that changing the test Frequency to quarterly is reasonably expected to increase the availability of the EFW system. Therefore, this change does not involve an increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety associated with the EFW pumps is provided by their flow capability following an event. Since testing will continue to confirm the required parameters for these pumps, this change does not involve a significant reduction in a margin of safety.

3.7 L10

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The emergency feedwater (EFW) pumps are used to support mitigation of the consequences of an accident; however, EFW is not considered as the initiator of any event. Therefore, the proposed revision will not significantly increase the probability of any accident previously evaluated. Since the function of the EFW pumps continues to be verified on a periodic basis, and the pumps continue to be required to be OPERABLE, the change of the Surveillance Frequency will not reduce the capability of required equipment to mitigate the event. This change also excludes requirements to perform functional testing of the motor driven EFW pump and it's associated train during MODE 4 when any steam generator is relied upon for heat removal. This presents requirements which are consistent with those proposed for the actuation system. During operation in this MODE, the time period for response to an event which requires emergency feedwater initiation is sufficient to allow for operator action. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety associated with the EFW system is provided by its capability to provide flow to the steam generators following an event. Since testing will continue to confirm the required parameters for the system, this change does not involve a significant reduction in a margin of safety.

3.7 L11

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any changes in hardware or methods of operation. The change in the submittal of "after the fact" information is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the administrative requirements for submittal of information and do not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

3.7 L12

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change in the conditions of the Frequency for the performance of a Surveillance Requirement does not result in any hardware changes. Neither the EFW system flow path verification, nor the EFW system flowpath configuration are considered as the initiator of any previously evaluated accident. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. The Surveillance will continue to provide timely recognition of EFW system impairment thus providing the operator an opportunity to provide system restoration. Further, the Surveillance will continue to be performed prior to operation that would result in sufficient core heat production that would require operation of the EFW System during a subsequent shutdown. Therefore, the proposed change to the conditions of the SR Frequency does not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Therefore, this change does not involve a significant reduction in the margin of safety.

3.7 L13

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

An extension of the Completion Time for a Required Action does not result in any hardware changes. The service water system is not considered as the initiator of a previously evaluated accidents. Therefore, this change does not significantly increase the probability of occurrence for initiation of any analyzed event. Further, neither the reason for the inoperability nor the Completion Time for performance of Required Actions significantly increases the consequences of an accident because the change does not change the assumed response of the equipment in performing its specified mitigation functions.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, neither the reason for the inoperability nor the short extension of the Completion Time interval involves a significant reduction in the margin of safety.

3.7 L14

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change in the Surveillance Requirement does not result in any hardware changes. The ventilation systems are not considered as the initiator of any previously evaluated accidents. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. The ventilation systems are considered in the mitigation of consequences of some accidents. However, the length of time for operation of the system during surveillances is still sufficient to verify proper functioning of the system. Therefore, the proposed change to the Applicability does not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper functioning of the system through surveillance. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will still ensure proper functioning of the system through surveillance. Therefore, this change does not involve a significant reduction in the margin of safety.

3.7 L15

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change in the Applicability and Required Actions does not result in any hardware changes. The analyses of concern are for a misloaded fuel assembly and a dropped fuel assembly. The spent fuel pool boron concentration is not considered as the initiator of either of these previously evaluated accidents. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. However, the spent fuel pool boron concentration is considered as an initial condition in the analysis of consequences of these accidents. Therefore, the Applicability will continue to include those conditions during which there is potential for these accidents, and the proposed Required Actions will initiate action to remove this potential. Therefore, the proposed change to the Applicability does not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not involve a significant reduction in the margin of safety.

3.7 L16

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The phrase "actual or simulated" in reference to the automatic initiation signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the function of the system functional test remains unaffected the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

Use of an actual signal instead of the existing requirement which limits use to a simulated signal, will not affect the performance of the surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between "actual" or "simulated." Therefore, the change does not involve a significant reduction in a margin of safety.

3.7 L17

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change in the Applicability and Required Actions does not result in any hardware changes. The analyses of concern are for a misloaded fuel assembly and a dropped fuel assembly. The penetration room ventilation system (PRVS) is not considered as the initiator of either of these previously evaluated accidents. Therefore, the change does not significantly increase the probability of occurrence for initiation of any previously evaluated accident. Also, the PRVS is not considered in the mitigation of consequences of these accidents. Therefore, the proposed change to the Applicability does not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for conditions during which there is potential for a fuel handling accident. Therefore, this change does not involve a significant reduction in the margin of safety.

3.7 L18

Not Used.

3.7 L19

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change will introduce the option to lock, seal, or otherwise secure the engineered safeguards (ES) valves for the service water system when OPERABILITY is required. Before this change, the only option was to lock the valves in the ES position. The method of verifying ES valve position is not an accident initiator and no hardware changes are proposed; therefore, the change does not significantly increase the probability of an accident. Expanding the methods available for verifying ES valve position does not significantly increase the consequences of a previously evaluated accident since the valves of interest are still placed in proper position for their safety function.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since expanding the methods of securing the ES valves in their actuated position has minimal impact on the availability of the systems. Furthermore, valve position surveillance, regardless of method of verification, is considered sufficient to provide system availability in the event of an accident.

ANO-290

3.7 L20

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

This change will allow the control room boundary to be opened intermittently under administrative controls, and will allow both trains of the control room ventilation system (CREVS) to be inoperable due to a control room boundary inoperability for a period of 24 hours. Neither CREVS nor the control room boundary are the initiator of any accident analyzed in the SAR. Therefore, this change does not result in a significant increase in the probability of an accident previously evaluated.

The CREVS and the control room boundary are intended to provide a habitable environment for the control room operators in the event of an accident that results in the release of radioactivity to the environment. The allowance to open the control room boundary intermittently is acceptable, because of the administrative controls that will be implemented to ensure that the opening can be rapidly closed when the need for control room isolation is indicated, restoring the control room habitability envelope. Allowing both CREVS trains to be inoperable for 24 hours due to an inoperable control room boundary is acceptable because of the low probability of an accident requiring control room isolation during any given 24 hour period, because entry into this Condition is expected to be an infrequent occurrence, and because preplanned compensatory measures to protect the control room operators from potential hazards are implemented. Therefore, this change will not result in a significant increase in the probability of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

This change does not involve a significant reduction in a margin of safety since: 1) administrative controls will be in place to ensure that an open control room boundary can be rapidly closed when a need for control room isolation is indicated; and 2) an inoperable control room boundary that renders both trains of CREVS inoperable is an infrequent occurrence, the probability of an accident requiring control room isolation during any given 24 hour period is low, and preplanned compensatory measures to protect the control room operators from potential hazards are implemented.

ANO-290

3.7 L21

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change will allow the penetration room ventilation system (PRVS) negative pressure boundary to be opened intermittently under administrative controls, and will allow both trains of the PRVS to be inoperable due to a PRVS negative pressure boundary inoperability for a period of 24 hours. Neither PRVS nor the PRVS negative pressure boundary are the initiator of any accident analyzed in the SAR. Therefore, this change does not result in a significant increase in the probability of an accident previously evaluated.

The PRVS and the PRVS negative pressure boundary are intended to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post-accident reactor building leaks. The allowance to open the PRVS negative pressure boundary intermittently is acceptable, because of the administrative controls that will be implemented to ensure that the opening can be rapidly closed when the need for PRVS negative pressure boundary isolation is indicated. Allowing both CREVS trains to be inoperable for 24 hours due to an inoperable PRVS negative pressure boundary is acceptable because of the low probability of an accident requiring PRVS negative pressure boundary isolation during any given 24 hour period, because entry into this Condition is expected to be an infrequent occurrence, and because preplanned compensatory measures to minimize environmental activity levels resulting from post-accident reactor building leaks are implemented. Therefore, this change will not result in a significant increase in the probability of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since: 1) administrative controls will be in place to ensure that an open PRVS negative pressure boundary can be rapidly closed when a need for PRVS negative pressure boundary isolation is indicated; and 2) an inoperable PRVS negative pressure boundary that renders both trains of PRVS inoperable is an infrequent occurrence, the probability of an accident requiring PRVS negative pressure boundary isolation during any given 24 hour period is low, and preplanned compensatory measures to minimize environmental activity levels resulting from post-accident reactor building leaks are implemented.

ANO-292

3.7 L22

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The change in the Condition and Required Action does not result in any hardware changes. The change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed). The change provides consistency between the Condition, Required Action and Applicable conditions for the LCO. Further, the change of Condition and Required Action does not significantly increase the consequences of an accident because the change does not affect the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters, from that resulting from the original analysis.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, for unit conditions during which analysis assumes the equipment to function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The Condition and Required Action are revised to be consistent with the Applicability for the equipment. Therefore, the change does not involve a significant reduction in the margin of safety.