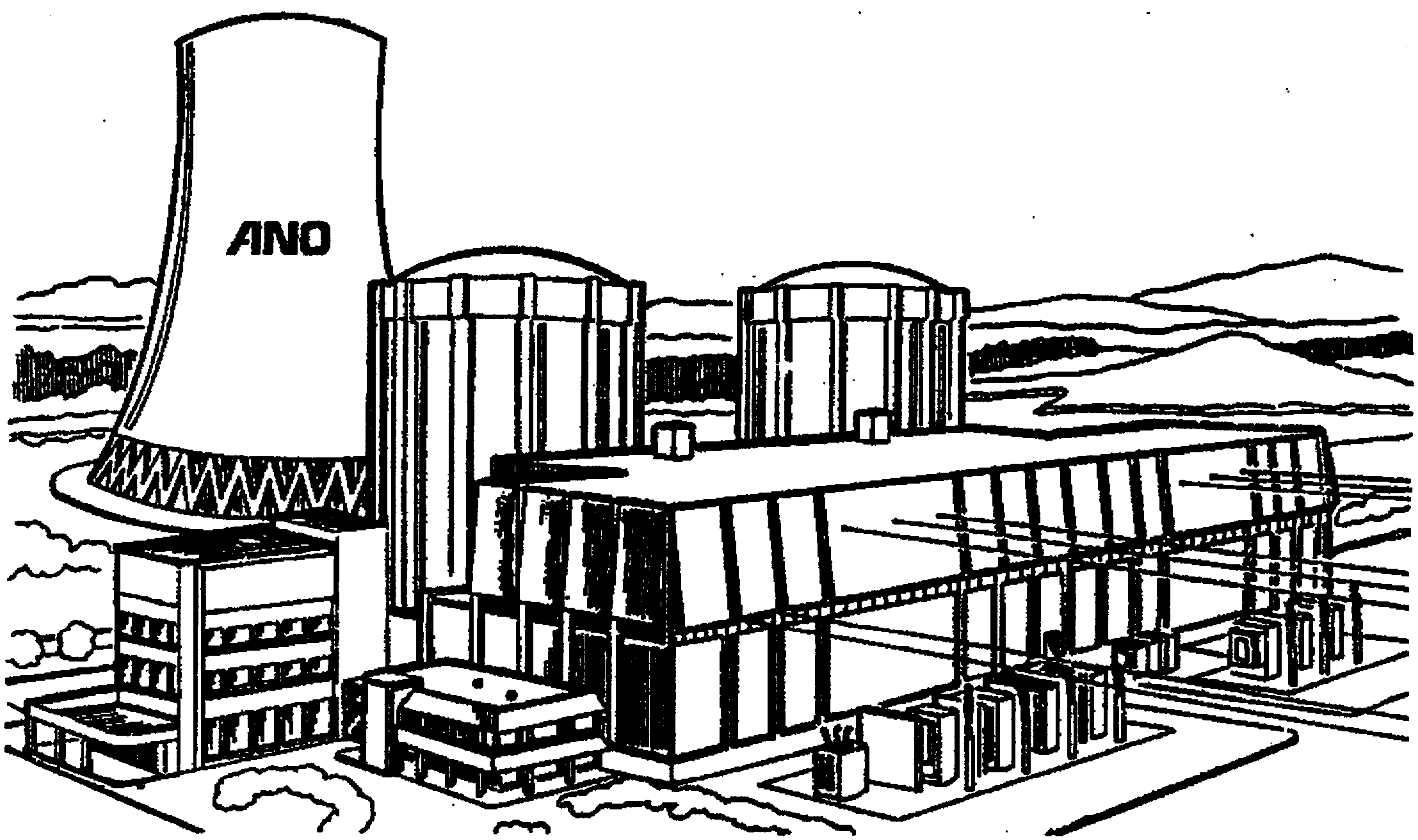
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED
TECHNICAL SPECIFICATIONS
SUBMITTAL



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This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
3.8.1	3.8.1	AC Sources-Operating
3.8.2	N/A	AC Sources-Shutdown
3.8.3	3.8.2	Diesel Fuel Oil, Lube Oil, and Starting Air
3.8.4	3.8.3	DC Sources-Operating.
3.8.5	N/A	DC Sources-Shutdown
3.8.6	3.8.4	Battery Cell Parameters
3.8.7	3.8.5	Inverters-Operating.
3.8.8	N/A	Inverters-Shutdown
3.8.9	3.8.6	Distribution Systems-Operating
3.8.10	N/A	Distribution Systems-Shutdown

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

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

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

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

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Restore required offsite circuit to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet LCO

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
		OR	
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	AND		
	B.4	Restore DG to OPERABLE status.	7 days AND 10 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One required offsite circuit inoperable. AND One DG inoperable.	--------------------NOTE-------------------- Enter applicable Conditions and Required Actions of LCO 3.8.6, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any train. --		
	D.1	Restore required offsite circuit to OPERABLE status.	12 hours
	OR		
	D.2	Restore DG to OPERABLE status.	12 hours
E. Two DGs inoperable.	E.1	Restore one DG to OPERABLE status.	2 hours
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1	Be in MODE 3.	12 hours
	AND		
	F.2	Be in MODE 5.	36 hours
G. Three or more required AC sources inoperable.	G.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	-----NOTE----- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. ---------- Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves "ready-to-load" conditions.	31 days
SR 3.8.1.3	-----NOTES----- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients below the required load range do not invalidate this test. 3. This SR shall be preceded by and follow, without shutdown, a successful performance of SR 3.8.1.2. ---------- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2475 kW.	31 days
SR 3.8.1.4	Verify each day tank contains ≥ 160 gallons of fuel oil.	31 days
SR 3.8.1.5	Verify the fuel oil transfer system operates to transfer fuel oil from storage tanks to the day tank.	31 days
SR 3.8.1.6	Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.	18 months

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. -- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes.	18 months

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. --- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected emergency loads through load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 Diesel Fuel Oil and Starting Air

LCO 3.8.2 The stored diesel fuel oil and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

-----------------------------NOTE-----------------------------

Separate Condition entry is allowed for each DG.

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

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	Verify each fuel oil storage tank contains $\geq$ 20,000 gallons of fuel.	31 days
SR 3.8.2.2	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.2.3	Verify each DG required air start receiver pressure is $\geq$ 175 psig.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 DC Sources - Operating

LCO 3.8.3 Both DC electrical power subsystems shall be OPERABLE.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify battery terminal voltage is ≥ 124.7 V on float charge.	7 days
SR 3.8.3.2	Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	18 months
SR 3.8.3.3	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 Battery Cell Parameters

LCO 3.8.4 Battery cell parameters shall be within limits.

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

ACTIONS

------------------------------------NOTE------------------------------------

Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Table 3.8.4-1 Category A or B limits.	A.1 Verify pilot cell electrolyte level and float voltage meet Table 3.8.4-1 Category C limits.	1 hour
	<u>AND</u>	
	A.2 Verify battery cell parameters meet Table 3.8.4-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>	
	A.3 Restore battery cell parameters to Table 3.8.4-1 Category A and B limits.	31 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with pilot cell or average electrolyte temperature of the representative cells < 60°F. <u>OR</u> One or more batteries with one or more battery cell parameters not within Table 3.8.4-1 Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery cell parameters meet Table 3.8.4-1 Category A limits.	7 days
SR 3.8.4.2	Verify electrolyte temperature of the pilot cell is ≥ 60°F.	31 days
SR 3.8.4.3	Verify battery cell parameters meet Table 3.8.4-1 Category B limits.	92 days AND Once within 24 hours after a battery discharge < 110 V AND Once within 24 hours after a battery overcharge > 145 V
SR 3.8.4.4	Verify average electrolyte temperature of representative cells is ≥ 60°F.	92 days

Table 3.8.4-1 (page 1 of 1)
Battery Cell Surveillance Requirements

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

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

(b) Corrected for electrolyte temperature.

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

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 Inverters - Operating

LCO 3.8.5 The required Red Train and Green Train inverters shall be OPERABLE.

-----NOTE-----

One inverter may be disconnected from its associated DC bus for ≤ 2 hours to perform load transfer to or from the swing inverter, provided:

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

b. All other 120 VAC vital buses are energized from their associated OPERABLE inverters.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.6, "Distribution Systems - Operating" with any 120 VAC vital bus de-energized. ---------- Restore inverter to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.5.1	Verify correct inverter voltage and alignment to required 120 VAC vital buses.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Distribution Systems - Operating

LCO 3.8.6 Two AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystem(s) inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One or more 120 VAC vital bus electrical power distribution subsystem(s) inoperable.	B.1 Restore 120 VAC vital bus electrical power subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
C. One or more DC electrical power distribution subsystem(s) inoperable.	C.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	12 hours 36 hours
E. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify correct breaker alignments to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	31 days

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternates) and the onsite standby power sources (emergency diesel generators (DGs)). As required by SAR, Section 1.4, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safeguards (ES) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

Offsite power is supplied to the unit switchyard from the transmission network by five transmission lines. From the switchyard, two electrically and physically separated offsite circuits provide AC power, through either the startup transformers or the unit auxiliary transformer, to the 4.16 kV ES buses. A detailed description of the offsite power network and the circuits to the Class 1E ES buses is found in the SAR, Chapter 8 (Ref. 2).

During typical on-line operation, power for unit equipment is provided from the unit auxiliary transformer. When the unit is off-line, unit equipment is typically powered from a startup transformer or from the unit auxiliary transformer back fed from the 500 kV switchyard. A unit trip (i.e., generator lockout) initiates an automatic transfer to an offsite power circuit (i.e., typically startup transformer No. 1). Startup transformer No. 2 is normally not selected for automatic transfer since it is the backup for both Unit 1 and Unit 2. In the event of a loss of offsite power to the startup transformer, an undervoltage condition trips its associated bus feeder breakers. When the startup transformer bus feeder breakers open, the bus feeder breakers for the alternate startup transformer automatically close (if available) provided the generator lockout relays have not been reset. If the power source is transferred to startup transformer No. 2, sufficient loads are automatically shed to avoid a degraded voltage condition (since startup transformer No. 2 is not sufficient to simultaneously provide power for full loading from both units.)

With an Engineered Safeguards Actuation System (ESAS) signal present, certain required unit loads are placed in service in a predetermined sequence. Within 1 minute after the initiating signal is received by the load sequencing timers, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are in service.

The onsite standby power source for each 4.16 kV ES bus is a dedicated DG. DGs 1 and 2 are dedicated to ES buses A3 and A4, respectively. A DG starts automatically on an applicable Engineered Safeguards Actuation System (ESAS) signal or on an ES bus degraded voltage or undervoltage signal (see LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation" and LCO 3.3.8, "Diesel Generator (DG) Loss of Power Start (LOPS)"). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ES bus undervoltage or degraded voltage, independent of or coincident with an ESAS signal. The DGs will also start and operate in the standby mode without tying to the ES bus on an ESAS signal alone. Following the trip of offsite power, an undervoltage signal strips nonpermanent loads from the ES bus. When the DG is tied to the ES bus, loads are then sequentially connected to their respective ES bus by the automatic load sequencing timers. The sequencing timers control the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In the event of a loss of preferred power, the ES electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a concurrent Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within 1 minute after the initiating signal is received by the load sequencing timers, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for emergency DGs 1 and 2 satisfy the guidance of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 2600 kW with 10% overload permissible for up to 2 hours in any 24 hour period. However, the "intended service" rating provided by the manufacturer is 2750 kW. This is the value used in postulated DG loading evaluations (Ref. 2). The ES loads that are powered from the 4.16 kV ES buses are listed in Reference 2.

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with SAR, Section 1.4, GDC 18 (Ref. 4). Periodic component tests are supplemented by extensive functional tests during outages (under simulated accident conditions) and maintenance inspections (per manufacturer's recommendations).

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the SAR, Chapter 14 (Ref. 5), assume ES systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ES systems so that the fuel, Reactor Coolant System (RCS), and reactor building design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution

Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Reactor Building Systems."

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit.

In MODES 1 and 2, the AC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 6). In MODES 3 and 4, the AC sources satisfy Criterion 4 of 10 CFR 50.36.

LCO

Two circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent DGs for each ES train (emergency DGs 1 and 2) ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormality or a postulated DBA.

Each required offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ES buses.

One offsite circuit consists of startup transformer No. 1, its supply from the switchyard bus tie autotransformer, either the 4160 V bus A1 or A2, and the feeder breaker providing power to either 4160 V ES bus A3 or A4. An alternative for this offsite circuit consists of the unit auxiliary transformer, its supply from the switchyard bus tie autotransformer and the overhead swing leads, either the 4160 V bus A1 or A2, and the feeder breaker providing power to either 4160 V ES bus A3 or A4. A second offsite circuit consists of startup transformer No. 2, its supply from the 161 kV switchyard ring bus, either the 4160 V bus A1 or A2, and the feeder breaker providing power to either 4160 V ES bus A3 or A4. If capable of automatic transfer, startup transformer No. 2 must have OPERABLE selective load-shed features for automatic shedding of loads to avoid a degraded voltage condition to be considered OPERABLE. An offsite circuit includes the necessary breakers and equipment to properly align the circuit and transmit power from the transmission line sources to a single 4160 V ES bus. If power is not being supplied to bus A3 or A4 from bus A1 or A2 respectively, one of the offsite circuits must be considered inoperable.

For the offsite AC sources, separation and independence are to the extent practical. An offsite circuit may be connected to more than one ES bus and not violate the separation criteria provided each OPERABLE required offsite circuit is capable of being aligned (manually or automatically) so that it is separate and independent of the other required offsite circuit. Only two of the possible offsite circuits are required to be OPERABLE.

When the main generator is synchronized to the 500 kV system, AC power for the ES loads may be taken from either the unit auxiliary transformer, either startup transformer, or a combination of these transformers concurrently sharing the load.

Power from the unit auxiliary transformer is not credited with meeting the requirements of LCO 3.8.1.a since it cannot function under all conditions (i.e., following a turbine trip) except when connected in the alternate configuration described above. However, powering the ES buses from the unit auxiliary transformer is permitted.

Each DG (DG1 and DG2) must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ES bus on detection of bus undervoltage. This will be accomplished within 15 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ES buses.

Proper sequencing of loads, including tripping of non-essential loads, is a required function for DG OPERABILITY.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

This LCO does not apply to the Alternate AC DG nor to the security DG.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormalities or abnormal transients; and

b. Adequate core cooling is provided and reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

ACTIONS

<u>A.1</u>

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

The Completion Time provides for a prompt confirmation of the OPERABILITY of the remaining offsite circuit. This is considered to be acceptable because of other indications, which are available in the control room for loss of the remaining offsite circuit.

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power available to supply its loads; and

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to both trains of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 7 days. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 7 days (for a total of 17 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on the time allowed in a 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 the 72 hour and 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

The Completion Time provides for a prompt confirmation of the OPERABILITY of the remaining offsite circuit. This is considered to be acceptable because of other indications, which are available in the control room for monitoring the status of the remaining offsite circuit.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single-failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on the other DG, the other DG would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the condition reporting program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DG(s) is not affected by the same problem as the inoperable DG.

B.4

Operation may continue in Condition B for a period that should not exceed 7 days.

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 13 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Condition A and Condition B are entered concurrently. The "AND" connector between the 7 day and 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that a Completion Time of 24 hours is allowed for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

With the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation would continue in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train (one or more trains), the Conditions and Required Actions for LCO 3.8.6, "Distribution Systems - Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.6 provides the appropriate restrictions for a de-energized train.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

E.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ES functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

With both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to 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 plant systems.

G.1

Condition G corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

Where the SRs discussed herein specify "ready-to-load" a minimum output voltage of 3750 V (~90% of the nominal 4160 V output voltage) is applicable. This value allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The required minimum frequency for loading of the DG is 58.8 Hz (derived from Safety Guide 9); however, this value is not routinely monitored to be within limit within 15 seconds. Meeting minimum frequency is expected prior to the DG voltage reaching the required minimum. This is administratively confirmed on an 18 month interval.

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, this SR is modified by a Note to indicate that DG starts for this Surveillance may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 testing with application of the Note, the DGs are started from standby conditions. Standby conditions for a DG means that the diesel engine oil is being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. The signal initiating the start of the

DG is varied from one test to another (start with handswitch at control room panel and at DG local control panel) to verify all starting circuits are OPERABLE.

SR 3.8.1.2 requires that the DG starts from standby conditions and achieves "ready-to-load" conditions (i.e., minimum voltage) within 15 seconds. The 15 second start requirement supports the assumptions of the design basis LOCA analysis in the SAR, Chapter 14 (Ref. 5).

The 31 day Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting full rated load. As discussed in Regulatory Guide 1.9 (Ref. 3), Section C.2.2, the load test is conducted at ≥ 90% of rated load, which is considered to be ≥ 90% of the intended service rating of 2750 kW, or 2475 kW. This parameter value (2475 kW) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. A minimum run time of 60 minutes ensures stabilized engine temperatures, while minimizing the time that the DG is connected to the offsite source.

The 31 day Frequency for this Surveillance provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

This SR is modified by three Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients (e.g., because of changing bus loads) do not invalidate this test. Note 3 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the engine mounted day tank is being properly maintained. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 40 minutes of DG operation at full load plus 10%.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, and the fuel delivery piping is not obstructed.

The design of the fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during DG monthly testing. Therefore, a 31 day Frequency is specified to correspond to the interval for DG testing.

SR 3.8.1.6

Transfer of each 4.16 kV ES bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. This Surveillance includes testing of the offsite power undervoltage and protective relaying interlocks associated with the required startup transformer power sources, and for startup transformer No. 2, this test also demonstrates the selective load shedding interlock function. (Note: This load shedding function is only required when startup transformer No. 2 is selected for automatic transfer.) These features provide protection of required equipment from a sustained degraded grid voltage situation.

The 18 month Frequency of the Surveillance takes into consideration the unit conditions required to perform the Surveillance (i.e., during refueling shutdown), and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.1.7

This Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the non-essential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve "ready-to-load" conditions (i.e., minimum required voltage) within the specified time.

The DG auto-start time of 15 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads, e.g., the running service water pump(s), is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue hardship or potential for undesired operation. In lieu of actual demonstration of connection and loading of loads during this test, separate testing that adequately shows the capability of the DG system to perform these functions is acceptable. This associated testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.8

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ES systems so that the fuel, RCS, and reactor building design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.7, during a loss of offsite power actuation test signal in conjunction with an ES actuation signal. This test is typically conducted by simulating an ESAS signal and either simultaneously or subsequently simulating a LOOP. In certain circumstances, many loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, DHR systems performing a DHR function are not desired to be interrupted from this mode of operation. In lieu of actual demonstration of connection and loading of loads during this test, separate testing that adequately shows the capability of the DG system to perform these functions is acceptable. This associated testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by a Note. The reason for the Note is to minimize wear and tear on the DGs during testing. For the purpose of this testing with application of the Note, the DGs are started from standby conditions, that is, with the engine oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs.

REFERENCES

1. SAR, Section 1.4, GDC 17.
2. SAR, Chapter 8.
3. Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants," Rev. 3, July 1993.
4. SAR, Section 1.4, GDC 18.
5. SAR, Chapter 14.
6. 10 CFR 50.36.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984 (0CNA078423).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 Diesel Fuel Oil and Starting Air

BASES

BACKGROUND

Each diesel generator (DG) is provided with fuel oil storage capacity sufficient to operate that diesel for a period of 3.5 days while the DG is supplying maximum post loss of coolant accident load demand discussed in the SAR, Section 8.3 (Ref. 1). The maximum load demand is calculated using the assumption that at least two DGs are initially available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time needed to replenish the onsite supply from outside sources.

Fuel oil is transferred from either storage tank to either day tank by either transfer pump (one pump is associated with each storage tank). Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG. All required outside tanks, pumps, and piping are located underground.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level. See Specification 5.5.13, "Diesel Fuel Oil Testing Program," for details.

Each DG has a designed air start system consisting of two redundant banks of two tanks (receivers) each. One bank of the two tanks contains adequate capacity (i.e., design margin) for five successive start attempts on the DG without recharging the air start receivers.

APPLICABLE SAFETY ANALYSES

The applicable Design Basis Accident (DBA) and transient analyses for the Diesel Fuel Oil and Starting Air systems are the same as for the DGs which they support. See the appropriate discussions in the Bases for LCO 3.8.1, "AC Sources – Operating."

Since diesel fuel oil and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Stored diesel fuel oil is required to have sufficient supply for 3.5 days of full load operation. It is also required to meet specific standards for quality. This requirement supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an abnormality or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources – Operating."

One bank of two tanks (receivers) with pressure ≥ 175 psig provides additional margin to that required for the assumed single DG start attempt without recharging the air start receivers.

APPLICABILITY

The AC sources (LCO 3.8.1) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormality or a postulated DBA. Since stored diesel fuel oil and the starting air subsystem support LCO 3.8.1, stored diesel fuel oil and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

<u>A.1</u>

In this Condition, the required fuel oil supply for a DG of 20,000 gallons (i.e., 138 inches) is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a 3 day supply of 17,140 gallons (i.e., 118 inches). These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity

(> 3 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

This Condition is entered as a result of a failure to meet the acceptance criterion of Specification 5.5.13. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

C.1

With the new fuel oil properties defined in the Bases for SR 3.8.2.2 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

D.1

With starting air receiver pressure < 175 psig in the required receivers, sufficient margin to that required for a single DG start attempt does not exist. However, as long as the receiver pressure is ≥ 158 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that the credited DG start is accomplished on the first attempt, and the low probability of an event during this brief period.

E.1

With a Required Action and associated Completion Time not met, or one or more DGs with fuel oil or required starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 3.5 days at full load. The 3.5 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location. An indicated tank level of 138 inches of fuel oil assures the required volume of 20,000 gallons for tanks T-57A and T-57B.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.2.2

The tests of fuel oil prior to addition to the storage tanks are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine operation. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between sampling (and associated results) of new fuel and addition of new fuel oil to the storage tank(s) to exceed 31 days. The tests, limits, and applicable ASTM Standards for the tests listed in Specification 5.5.13, "Diesel Fuel Oil Testing Program," are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-88 (Ref. 3); and

b. Verify in accordance with the tests specified in ASTM D975-81 (Ref. 3) that the sample has:

    1. an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27°, ≤ 39°,

    2. a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes,

3. a flash point of ≥ 125°F, and

4. water and sediment within limits.

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO since the fuel oil is not added to the storage tanks.

Following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-81 (Ref. 3) are met for new fuel oil when tested in accordance with ASTM D975-81 (Ref. 3), except that the analysis for sulfur may be performed in accordance with ASTM D1552-90 (Ref. 3) or ASTM D2622-87 (Ref. 3). These additional analyses are required by Specification 5.5.13, "Diesel Fuel Oil Testing Program," to be performed within 31 days following sampling and addition. This 31 days is intended to assure: 1) that the sample taken is not more than 31 days old at the time of adding the fuel oil to the storage tank, and 2) that the results of a new fuel oil sample (sample obtained prior to addition but not more than 31 days prior to) are obtained within 31 days after addition. For circumstances where multiple fuel oil additions are made within a short period of time, the samples taken for each batch added to the storage tank can be composited for a single follow-up analysis. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-88, Method A (Ref. 3). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each tank is considered and tested separately.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.2.3

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The pressure specified in this SR is intended to reflect margin above the minimum pressure necessary to start the DG utilizing both receiver tanks in one of the two banks.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

REFERENCES

1. SAR, Section 8.3.
2. 10 CFR 50 36.
3. ASTM Standards: D4057-88; D975-81; D4176-86; D1552-90; D2622-87; D2276-88, Method A.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and 120 VAC vital bus power (via inverters). As required by SAR, Section 1.4, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Red Train and Green Train). Each subsystem consists of one 125 VDC battery, the associated battery charger for each battery, and all the associated control equipment and interconnecting cabling.

Additionally, there is one spare battery charger per subsystem, which provides backup service in the event that a battery charger is out of service. If the spare battery charger is substituted, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, each 125 VDC subsystem is powered from the inservice battery charger with the battery floating on the system. In case of a loss of normal power to the battery charger, the DC load is automatically powered from the station battery. This results in a discharge of the associated battery (and may affect both the system and cell parameters).

The Red Train and Green Train DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 480 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the 120 VAC vital buses.

The DC power distribution system is described in more detail in Bases for LCO 3.8.6, "Distributions System – Operating."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours in addition to supplying power for the operation of momentary loads during the 2 hour period as discussed in the SAR, Chapter 8 (Ref. 4).

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries for Red Train and Green Train DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand.

Each subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger is also designed with sufficient capacity to restore the battery from the design minimum charge to its fully charged state while supplying normal steady state loads.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 5), assume that Engineered Safeguards (ES) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit.

In MODES 1 and 2, the DC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 6). In MODES 3 and 4, the DC sources satisfy Criterion 4 of 10 CFR 50.36.

LCO

The DC electrical power subsystems, each subsystem consisting of one battery, one of two battery chargers and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an abnormality or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires the associated battery to be OPERABLE and connected to the associated DC bus and one of its respective chargers to be OPERABLE and capable of being connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormalities; and

b. Adequate core cooling is provided, and reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

ACTIONS

A.1

Condition A represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 8 hour limit is consistent with the allowed time for an inoperable DC distribution system train.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery chargers, or inoperable battery chargers and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst- case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power subsystems with attendant loss of ES functions, continued power operation should not exceed 8 hours. The 8 hour Completion Time reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

B.1 and B.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to 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 3.8.3.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery (2.15 V per cell average) and are consistent with IEEE-450 (Ref. 7). The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 7).

SR 3.8.3.2

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

The Surveillance Frequency of 18 months is consistent with considerations that the battery service test should be performed during refueling outages, or at some other outage.

A modified performance discharge test may be performed in lieu of a service test.

The modified performance discharge test (Ref. 7) is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test and the test discharge rate must envelope the duty cycle of the service test if the modified performance discharge test is performed in lieu of a service test.

SR 3.8.3.3

A battery performance discharge test is a test of constant current capacity of a battery, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage (Ref. 7).

A battery modified performance discharge test is described in the Bases for SR 3.8.3.2. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.3.3; however, only the modified performance discharge test may be used to satisfy SR 3.8.3.3 while satisfying the requirements of SR 3.8.3.2 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 7), which recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 7), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is > 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 7).

REFERENCES

1. SAR, Section 1.4, GDC 17.
2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," March, 1971.
3. IEEE-308-1971, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."
4. SAR, Chapter 8.
5. SAR, Chapter 14.
6. 10 CFR 50.36.
7. IEEE-450-1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.3, "DC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 1), assume Engineered Safeguards systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit.

Battery cell parameters satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an abnormality or a postulated DBA. The limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

APPLICABILITY

The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery cell parameters are only required to be within limits when the DC power source is required to be OPERABLE. See the Applicability discussion in Bases for LCO 3.8.3.

ACTIONS

The Actions Table is modified by a Note, which indicates that separate Condition entry is allowed for each battery. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DC subsystem. Complying with the Required Actions for one inoperable DC subsystem may allow for continued operation, and subsequent inoperable DC subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1, A.2, and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the Category C limits specified in Table 3.8.4-1 in the accompanying LCO, the battery is degraded but there still is sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). These checks will provide a quick representative indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cell. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to within the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because parameter measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to within Category A and B limits. This periodic verification is consistent with the increased potential to exceed these battery cell parameter limits during these conditions.

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement may not be available. Therefore, the battery must be immediately declared inoperable and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as the Required Actions and associated Completion Time of Condition A not met or average electrolyte temperature of representative cell falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte level and temperature of pilot cells.

SR 3.8.4.2 and SR 3.8.4.4

This Surveillance verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450 (Ref. 3), which states that the temperature of electrolytes in the pilot cell should be determined at least once per month and that the temperature in representative cells (~10% of all connected cells) should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

SR 3.8.4.3

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). In addition, within 24 hours of a battery discharge < 110 V or a battery overcharge > 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

Table 3.8.4-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 3.8.4-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on a recommendation of IEEE-450 (Ref. 3), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195. This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 with the average of all connected cells > 1.195. These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limits of average specific gravity ≥ 1.190 is based on manufacturer recommendations. In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Footnotes (b) and (c) to Table 3.8.4-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.4-1 requires the above mentioned correction for electrolyte temperature.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.4-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

REFERENCES

1. SAR, Chapter 14.
2. 10 CFR 50.36.
3. IEEE-450-1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 Inverters - Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the 120 VAC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital bus. The inverters are normally powered from the 125 VDC Electrical Power System. An alternate AC source is also provided for the inverters which can be used during maintenance, during switchover to or from a swing inverter, or in case of inverter failure. The inverters provide an uninterruptible power source for the safety significant instrumentation and controls, including the Reactor Protection System (RPS), the Engineered Safeguards Actuation System (ESAS), and the Emergency Feedwater Initiation and Control (EFIC) system. Additionally, there are two swing inverters (one per train) which provide backup service in the event that an inverter is out of service. If the swing inverter is placed in service, requirements of independence and redundancy between trains are maintained. Specific details on inverters and their operating characteristics are found in SAR, Chapter 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 2), assume Engineered Safeguards systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the safety significant instrumentation and controls so that the fuel, Reactor Coolant System, and reactor building design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Reactor Building Systems."

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3) in MODES 1 and 2. In MODES 3 and 4, the inverters satisfy Criterion 4 of 10 CFR 50.36.

LCO

The inverters ensure the availability of AC electrical power for the instrumentation required to shut down the reactor and maintain it in a safe condition after an abnormality or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the safety significant instrumentation and controls is maintained. The four required inverters (two per train) ensure an uninterruptible supply of AC electrical power to the 120 VAC vital buses even if the 4.16 kV safety buses are de-energized.

OPERABLE inverters require the associated 120 VAC vital bus to be powered by the inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC Electrical Power System with associated OPERABLE station battery.

This LCO is modified by a Note that allows one required inverter to be disconnected from its associated DC bus for ≤ 2 hours to allow load transfer to or from a swing inverter, if the 120 VAC vital bus is powered from an alternate AC source during the period and all other inverters are OPERABLE.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormalities; and

b. Adequate core cooling is provided, and reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

ACTIONS

A.1

With a required inverter inoperable, its associated 120 VAC vital bus becomes inoperable unless it remains energized.

For this reason, a Note has been included in Condition A requiring entry into the Conditions and Required Actions of LCO 3.8.6, "Distribution Systems - Operating."

This ensures the vital bus is re-energized within 8 hours. Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit takes into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC vital bus is powered from its alternate AC source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the Required Actions and associated Completion Time are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation connected to the 120 VAC vital buses. The 31 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. SAR, Chapter 8.

2. SAR, Chapter 14.

3. 10 CFR 50.36.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Distribution Systems - Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are divided by train into two redundant and independent AC, DC, and 120 VAC vital bus electrical power distribution subsystems.

Each AC electrical power subsystem consists of an Engineered Safeguards (ES) 4.16 kV bus and 480 V buses. Each 4.16 kV ES bus has two offsite sources of power as well as a dedicated onsite diesel generator (DG) source as described in the Bases for LCO 3.8.1, "AC Sources - Operating." If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ES bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries.

The 120 VAC vital buses are arranged in two load groups per subsystem and are normally powered from the inverters. The alternate power supply for these vital buses is powered from the same Class 1E AC subsystem as the associated inverter, and its use is governed by LCO 3.8.5, "Inverters - Operating."

There are two independent 125 VDC electrical power distribution subsystems (one for each train).

The list of all required distribution buses, safety related load centers, motor control centers, and distribution panels is presented in Table B 3.8.6-1. Other ES buses are not required by this Specification, but may be required to support OPERABILITY of other required equipment. Inoperability of such buses would result in inoperability of the supported equipment.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the SAR, Chapter 14 (Ref. 1), assume ES systems are OPERABLE. The AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ES systems so that the fuel, Reactor Coolant System, and reactor building design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Reactor Building Systems."

The OPERABILITY of the AC, DC, and 120 VAC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit.

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

LCO

The required power distribution subsystems listed in Table B 3.8.6-1 ensure the availability of AC, DC, and 120 VAC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after abnormality or a postulated DBA. The AC, DC, and 120 VAC vital bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the AC, DC, and 120 VAC vital bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ES is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor. OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, and motor control centers to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE 120 VAC vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage or from its alternate AC source.

In addition, cross-tie breakers between redundant safety related AC, DC, and AC vital bus power distribution subsystems must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any cross-tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormalities; and

b. Adequate core cooling is provided, and reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

ACTIONS

A.1

With one or more required AC electrical power distribution subsystems inoperable, the remaining OPERABLE portions of the AC electrical power distribution subsystem(s) may be capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ES functions not being supported. Therefore, the required AC buses, load centers, and motor control centers must be restored to OPERABLE status within 8 hours.

Condition A worst case scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO.

If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one or more 120 VAC vital bus electrical power distribution subsystems inoperable, the remaining OPERABLE portions of the 120 VAC vital bus subsystem(s) may be capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum ES functions not being supported. Therefore, the 120 VAC vital bus subsystem(s) must be restored to OPERABLE status within 8 hours by powering the affected bus(es) from the associated inverter via inverted DC or from its alternate AC source.

Condition B represents one or more 120 VAC vital bus subsystem(s) without power; potentially both the DC source and the associated alternate AC source are nonfunctioning. In this situation the unit is significantly more vulnerable to a complete loss of all un-interruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital bus subsystem(s) and restoring power to the affected vital bus subsystem(s).

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 8 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 8 hour Completion Time takes into account the importance to safety of restoring the 120 VAC vital bus subsystem(s) to OPERABLE status, the redundant capability afforded by the other OPERABLE vital bus subsystem, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the 120 VAC vital bus subsystem(s). At this time, an AC train could again become inoperable, and 120 VAC vital bus subsystem(s) restored to OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With one or more DC subsystems inoperable, the remaining OPERABLE portions of the DC electrical power distribution subsystems may be capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ES functions not being supported. Therefore, the DC buses must be restored to OPERABLE status within 8 hours by powering the bus from the associated battery or one of the two associated chargers.

Condition C represents one or more DC subsystem(s) without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 8 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to 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 plant systems.

E.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This Surveillance verifies that the required AC, DC, and 120 VAC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained. The 31 day Frequency takes into account the redundant capability of the AC, DC, and 120 VAC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. SAR, Chapter 14.

2. 10 CFR 50.36.

Table B 3.8.6-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	RED TRAIN	GREEN TRAIN
AC Electrical Power Distribution Subsystems	4160 V 480 V 480 V	ES Bus A3 Load Center B5 Motor Control Center B51, B52, B53, B57	ES Bus A4 Load Center B6 Motor Control Center B61, B62, B63, B65, B56* and B55
DC Electrical Power Distribution Subsystems	125 V	Bus D01 Bus RA1 Distribution Panel D11	Bus D02 Bus RA2 Distribution Panel D21
120 VAC Vital Bus Subsystems	120 V	Bus RS1 Bus RS3	Bus RS2 Bus RS4

* Swing bus (normally associated with Green Train). Bus B55 is powered from Bus B56.

CTS DISCUSSION OF CHANGES

ITS Section 3.8: Electrical Power 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 CTS 3.0.5 provides guidance for supported system inoperabilities that result from loss of one of two required diverse electrical power sources (loss of DG or loss of normal {offsite} power). For other than MODES 1, 2, 3, and 4 (i.e., "Cold Shutdown conditions or Refueling Shutdown"), this guidance is not necessary since the ANO-1 assumed and credited functions for safety related systems do not rely on offsite AND DG power in these shutdown conditions. That is, during shutdown Modes the assumed and credited functions are not required to consider a concurrent loss of power source in order to perform their intended safety function. In accordance with the CTS definition of OPERABILITY, the "*necessary* … power sources" that are required are met with simply providing power from normal OR emergency sources during shutdown Modes. Therefore, the CTS 3.0.5 reference to non-applicability in Cold Shutdown or Refueling Shutdown is simply a clarification of the natural intent. The ITS presentation that incorporated CTS 3.0.5 into ITS 3.8.1, captures the necessary and applicable limitations without the need for a specific reference to non-applicability in shutdown Modes. (Refer also to 3.8 Discussion of Difference (DOD) #17 for related discussion on electrical power sources during shutdown conditions).

A4 This information is out of date and no longer represents a requirement or a relaxation of the requirements. Omitting this information has no impact on the assumptions of the safety analysis or any equipment used to mitigate any design basis event.

A5 Not used.

A6 Not used.

A7 CTS 4.6.1.1 is revised to include a Note with ITS SR 3.8.1.2 (NUREG SR 3.8.1.2) that allows for DG starts to be preceded by an engine prelube period and followed by a warmup period prior to loading. CTS 4.6.1.2 is revised to include a Note with ITS SR 3.8.1.7 and SR 3.8.1.8 (NUREG SR 3.8.1.11 and SR 3.8.1.19) that allows for DG starts to be preceded by an engine prelube period and followed by a warmup period prior to loading. This is consistent with current application of the CTS (which neither requires nor prohibits prelube and warmup periods), with the recommendations of Generic Letter 84-15, and with NUREG-1430. Also, SR 3.8.1.3, Notes 1, 2, & 3, are included to reflect current practice that are not specifically addressed in the current requirements. These proposed Notes do not change the actual CTS requirements as shown in CTS 4.6.1.1 and CTS 4.6.1.2 and are considered administrative. All of these Notes are consistent with allowances provided in NUREG-1430.

A8 Not used.

A9 CTS 4.6.1.1 requires load testing the DGs at "full rated load." The DGs ratings include: 2600 kW continuous service, 2750 kW intended service, 2850 kW for 2000 hours, and 3000 kW for 4 hours. Regulatory Guide 1.9, Section C.2.2 indicates that the full load testing is to be conducted between 90% to 100% of the continuous load rating. Since the intended service rating is used for load rating of the ANO-1 DGs, the minimum test loading is based on 90% of the intended service full load, or 2475 kW. The CTS is not considered to include an upper limit for this test. Further, even when considering the "intended service rating," the DG is clearly capable of operating at up to 3000 kW for the one hour test without affecting OPERABILITY. The loading of the DG can easily be controlled between 2475 kW and 3000 kW; therefore, no upper limit is necessary. Further, as indicated in the NUREG Bases, the upper limit is only to prevent routine overloading which may result in more frequent inspections. Again, this does not affect OPERABILITY, but only availability which is licensee controlled under the maintenance program. The upper limit for loading of the DGs is therefore not adopted. Such an upper limit is unnecessarily restrictive for a one hour run when set at ≥ 90% of the rated load. This is considered to be consistent with the CTS and therefore, the change is administrative.

A10 Not used.

A11 Not used.

A12 CTS 3.7.1 established an Applicability of > 200°F for the Auxiliary Electrical Systems Specifications. This is equivalent to ITS MODES 1, 2, 3 and 4 and is consistent with the stated Applicability established for ITS 3.8.1 and 3.8.6.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 Specific requirements are included for inverters during operation in MODES 1, 2, 3, and 4. These inverters provide emergency power for AC vital buses which in turn power the reactor protection system and engineered safeguards actuation system instrumentation and controls. These are additional restrictions on unit operation. These changes are consistent with NUREG-1430.

M2 Not used.

M3 The default actions in CTS 3.7.2.A for not meeting the requirements of any of the other conditions in CTS 3.7.2 are revised to omit the allowed delay of 24 hours in Hot Shutdown. This change revises the default conditions for proposed LCO 3.8.1 to be consistent with other default conditions, e.g., LCO 3.0.3, for failure to meet the ACTIONS and/or a loss of function. This change represents an additional restriction on unit operation.

M4 An additional Required Action (ITS 3.8.1 RA A.1) is included for an inoperable required offsite circuit, which requires prompt verification of the appropriate breaker alignment and indicated power availability for the remaining OPERABLE required offsite circuit. The prompt verification is consistent with CTS (in order to determine that only one offsite circuit is inoperable). This change provides additional knowledge of the status of the remaining offsite circuit in order to ensure a highly reliable power source is available should the unit trip. This is consistent with NUREG-1430.

M5 In the event of concurrent diesel generator and offsite circuit inoperabilities, the existing CTS Actions appear to allow independent application of allowed repair times. When a subsequent inoperability occurs just prior to restoration of the previous inoperability and close to the expiration of the allowed 72 hours, when taken to extreme, this independent application can provide an unlimited time of operation with an inoperable AC source. While these simultaneous inoperabilities are expected to be rare, a maximum restoration time limit is not imposed in the CTS. The proposed ITS 3.8.1 RA A.3 and RA B.4 format presents a maximum restoration time as an additional Completion Time of "10 days from discovery of failure to meet the LCO." A similar Completion Time of "16 hours from discovery of failure to meet the LCO" is also incorporated for the various electrical power distribution subsystems in proposed ITS 3.8.6 (refer to ITS 3.8.6 RA A.1, B.1 and C.1). These additional Completion Times represent additional restrictions on unit operation. This change is consistent with NUREG-1430.

M6 Not used.

M7 Not used.

M8 CTS 3.7.1.G requires the selective load-shed features associated with startup transformer No. 2 (ST2) to be OPERABLE, but only "if selected for auto transfer." It is noted for reference that ST2 is considered the secondary offsite circuit, which is allowed to be manually connected if desired, and still be considered OPERABLE. Auto connection of ST2 is also allowed but only with the selective load-shed feature OPERABLE. CTS 3.7.2.H provides actions for inoperability of this feature with two options: Option (1) requires that the ST2 feeder breakers be placed in "pull-to-lock" within one hour; and that the interlock be restored within 30 days, or a Special Report be submitted within 30 days. Option (2) allows 72 hours for restoration and then requires hot shutdown in the next 6 hours and cold shutdown in the following 30 hours per Note 14 of Table 3.5.1-1. Option (2) is essentially equivalent to considering the offsite circuit inoperable in ITS.

To evaluate Option (1), "if selected for auto transfer" is first understood to refer to being "capable of auto transfer." With ST2 feeder breakers not in pull-to-lock, it is continuously capable of auto transfer – whether or not ST2 is "selected" ("selected" only affects the order in which auto transfer is attempted). As such, the CTS action of 3.7.2.H Option (1) (which requires ST2 feeder breakers be placed in pull-to-lock) removes the capability for auto transfer and effectively eliminates the requirement for any further completion of the Action (to submit a Special Report). It also allows ST2 to be considered OPERABLE (based on the capability of manual connection after manual load shedding).

Given these discussions, ST2 is considered to be an inoperable offsite circuit during the time that the selective load-shed features are inoperable and the feeder breakers are available for auto transfer. CTS 3.7.2.H allows 1 hour to establish the feeder breakers in pull-to-lock before imposing a 72 hour restoration time. The ITS action for an inoperable offsite circuit allows only 72 hours. Therefore, this change eliminates the additional CTS allowed 1 hour time. This is a more restrictive change with no impact on safety since 72 hours remains a reasonable time to reestablish the OPERABILITY of ST2 (which, in this scenario, can be accomplished by placing the feeder breakers in pull-to-lock if desired).

M9 A new surveillance of offsite power circuits is included to provide additional assurance of power availability. A weekly verification of proper breaker alignment and indicated power availability for each required offsite circuit is being included as proposed ITS SR 3.8.1.1. This is an additional restriction on unit operation consistent with NUREG-1430.

M10 Not used.

M11 CTS 4.6.1.4.e is supplemented to explicitly include new fuel oil testing in ITS SR 3.8.2.2. The sampling of new fuel oil prior to addition to the storage tanks provide a means of determining whether the new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate detrimental impact on diesel engine operation. Additionally, Action C is included to provide a limited restoration time in the event new fuel oil is added and subsequent tests of the new fuel oil are discovered to be out of limits. This is an additional restriction on unit operation consistent with NUREG-1430.

M12 Specific acceptance criteria for proper operation following a loss of offsite power and following a concurrent loss of offsite power and engineered safety features actuation are included in the proposed SRs for the diesel generator tests, proposed ITS SR 3.8.1.7 and SR 3.8.1.8. CTS 4.6.1.2 provides only requirements that the diesel generator be tested under these conditions. Specific acceptance criteria will assure that the parameters are evaluated against the assumptions used for the parameters in the safety analysis. These acceptance criteria represent an additional restriction on unit operation consistent with NUREG-1430.

M13 Not used.

M14 A periodic verification of proper breaker alignment and indicated power availability on safeguards and instrument buses is being added to the ITS, as SR 3.8.6.1. This verification provides a specific surveillance to verify compliance with CTS 3.7.1.B. This is an additional restriction on unit operations based on NUREG-1430 which considers the installed instrumentation.

M15 CTS 4.6.1.1 requires that the diesel generator be run following the monthly start test "until diesel generator operating temperatures have stabilized." Proposed SR 3.8.1.3 dictates that DG operation continue for one hour. This is a more explicit presentation, but generally consistent with the current requirement. Experience has shown that 60 minutes is the maximum time the diesel generator requires to reach operating temperature. Since shorter times may be acceptable under the CTS, this change is considered more restrictive on unit operations. This change is consistent with NUREG-1430.

M16 CTS 3.0.5 identifies additional system, subsystem, train, component or device OPERABILITY criteria when either its emergency AC power or normal AC power source is inoperable. Once a system, subsystem, train, component or device redundant to one associated with the inoperable AC source is determined to be concurrently inoperable, the CTS allows 2 hours to initiate a required shutdown which is equivalent to the CTS 3.0.3 times. If ITS LCO 3.0.3 is required to be entered (see DOC L1), the resulting actions would be more restrictive than the CTS as described in Section 3.0 DOC M2. This is considered to be an additional restriction on unit operation consistent with NUREG-1430.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 CTS 3.0.5 describes how system, subsystem, train, component or device OPERABILITY is determined when either its emergency AC power or normal AC power source is inoperable. The CTS determination is generally expected to be a "prompt" determination, even though no specific limit on the time frame is identified. When compared to CTS 3.0.3, there would appear to be a one hour allowance to determine if a loss of function exists, before entering the CTS 3.0.3 time frames of 1, 6, 6, & 24 hours. The proposed ACTIONS for inoperable AC Sources specify time limits longer than that provided by the CTS for completing this initial determination and declaring the system, subsystem, train, component or device associated with the inoperable AC source inoperable. Twenty-four hours has been provided if an offsite circuit is inoperable, 12 hours has been provided if both unit-specific circuits are inoperable, and 4 hours if one diesel generator is inoperable. These times provide a reasonable time to restore the feature or AC source to OPERABLE status commensurate with the level of degradation of unit systems. This change is consistent with NUREG-1430.

The ACTIONS in CTS 3.0.5 ultimately require the unit be placed in a MODE in which the Specification is not applicable, down to and including Cold Shutdown within a set Completion Time. A similar requirement is also imposed in the proposed ITS 3.8.1 Required Actions for an inoperable offsite circuit, for two inoperable offsite circuits, and for an inoperable diesel generator, for the Required Actions not met. However, the intermediate step to be in Hot Standby within 6 hours is omitted. This minor change allows the operations staff additional flexibility in determining the preferred rate and method of performing the shutdown. This change is also consistent with NUREG-1430.

The ITS does not always require a shutdown if a loss of function is identified. Rather, it requires that both redundant components be declared inoperable and the corresponding ACTIONS of the LCO applicable for those components be entered. These ACTIONS may provide for other compensatory measures that have been determined to be appropriate for the condition. Therefore, CTS 3.7.2.C and 3.7.3.A.2 statements regarding opposite train component verification of OPERABILITY are unnecessary and duplicative of the CTS 3.0.5 requirements (which become ITS 3.8.1 Required Actions A.2 and B.2 requirements) and are shown as deleted. This change is consistent with NUREG-1430.

L2 Actions for ITS LCO 3.8.2 will allow time for restoration of parameters that do not result in immediate inability of the diesel generators to perform their function. These parameters, while supporting diesel generator OPERABILITY, contain substantial margin in addition to the limits, which would be absolutely necessary for diesel generator OPERABILITY. Therefore, certain levels of degradation in these parameters are justified to extend the allowances for restoration. During the proposed extended periods for restoration of these parameters, the diesel generator would still be capable of performing its intended function. For example, the fuel oil volume may be allowed to be less than the CTS 20,000 gal for up to 48 hours, provided 6/7 of the required supply continues to be available. The reduced limits provide a high level of assurance that the AC Sources will be available when needed. These changes are consistent with NUREG-1430.

L3 The ITS would allow several new Conditions which allow concurrent inoperabilities of the equipment required by CTS 3.7.1. CTS 3.7.2.A allows only one of the subordinate conditions (i.e., CTS 3.7.2.B through 3.7.2.H) to exist and several of the allowed outage times are dependent on the OPERABILITY of other electrical equipment. These are discussed separately below.

CTS 3.7.2.B for one inoperable offsite circuit requires verification of the diesel generators and an implied verification that the other offsite circuit is available. Proposed ITS 3.8.1 Condition C will allow two inoperable offsite circuits for 24 hours, and proposed ITS 3.8.1 Condition D will allow a combination of one offsite circuit and one diesel generator to be inoperable for 12 hours. CTS 3.7.2.C similarly allows for only one diesel generator to be inoperable and only if both offsite circuits are verified available. In addition to the combination proposed in ITS 3.8.1 Condition D, Condition E will allow both diesel generators to be inoperable for 2 hours. These allowed Completion Times are consistent with Regulatory Guide 1.93. Further, inoperability of the AC distribution systems is totally separated from dependence on the OPERABILITY of the diesel generators. These conditions of concurrent inoperabilities have been generically determined to be acceptable temporary conditions partially due to the high improbability of such a condition existing concurrently with the need for the equipment to perform its safety functions and partially due to the abilities of the remaining equipment. This improbability is supported by allowing only short duration for such conditions. However, concurrent inoperabilities beyond those identified will continue to result in unit shutdown because application of proposed ITS 3.8.1 Condition G and ITS 3.8.6 Condition E will result in unit shutdown in accordance with LCO 3.0.3. This change is consistent with NUREG-1430.

CTS 3.7.2.C and 3.7.2.D also require demonstrations of OPERABILITY of diesel generators when other AC Sources are inoperable. These demonstrations are no longer required if the inoperability is other than a diesel generator, and the demonstration is not required if the inoperability is a diesel generator if it can be determined that the inoperability is not caused by a common mode failure. These conditions present no basis for questioning the OPERABILITY of a diesel generator for which the surveillances are current. Therefore, these requirements for demonstration of

OPERABILITY are omitted. This change is also consistent with NUREG-1430 and with several previously approved changes for other plant's Technical Specifications.

L4 The CTS 3.7.2.B Completion Time for restoration of an inoperable offsite circuit is extended from 24 hours to 72 hours consistent with NUREG 3.8.1 RA A.3. The proposed Completion Time is based on a reasonable time for repairs of an offsite circuit, the capacity and capability of the remaining sources, and the low probability of a design basis event occurring during this period. This change is also consistent with the completion time of other CTS Specifications (e.g., Table 3.5.1-1, Note 14 referenced from CTS 3.7.2.H(2)) also pertaining to offsite power (switchyard) component inoperability. This change is consistent with NUREG-1430.

Additionally, the allowed Completion Time provided by CTS Table 3.5.1-1, Note 14, (referenced from CTS 3.7.2.H(2)) to reach hot shutdown following a 72 hour restoration period is increased from 6 hours to 12 hours consistent with NUREG 3.8.1 RA G.1. This is a reasonable time to achieve the required unit conditions from full power considering the degraded power sources. This change is consistent with NUREG-1430.

L5 NUREG 3.8.6 contains and ACTIONS Note that allows separate condition entry for each battery. The CTS requirements do not explicitly allow the separate condition entry, nor do they prohibit separate condition entry. ITS 3.8.4 provides for separate condition entry. This recognizes the fact that although both batteries may be in a degraded condition, they still have sufficient capacity to perform their function. This change is consistent with NUREG-1430.

L6 The proposed LCO 3.8.1 Required Action B.3.1 provides an allowance to avoid unnecessary testing of the OPERABLE diesel generator when a diesel generator is declared inoperable. This change is consistent with that approved on the River Bend Station docket (Amendment #64, dated 9/29/92). The intent of the current actions is to confirm no common-mode failure has rendered more than one diesel generator inoperable. This assurance can be ascertained in many cases by means other than the existing requirement for a diesel generator start and load. If an assessment can determine no common-mode failure exists on the remaining OPERABLE diesel generators, the proposal allows for not requiring an unnecessary diesel generator start. Minimizing diesel generator starts is recommended to avoid unnecessary diesel wear, thereby enhancing overall diesel generator reliability (refer to Generic Letter 84-15). Furthermore, if a diesel generator start is necessary (e.g., common-mode failure can not be ruled out), the requirement to load the diesel generator (requiring paralleling with offsite power) is eliminated. Sufficient assurance of continued diesel generator OPERABILITY is provided with a single start-only test, rather than repeated starting and loading. These changes are consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

L7 CTS 4.6.1.2.d requires that the diesel generators operate "for ≥ 1 hour after operating temperatures have stabilized" following the loss of offsite power testing. The proposed ITS SR 3.8.1.7 and SR 3.8.1.8 require that the diesel generator operate for only 5 minutes. Capability for the diesel generator extended operation is adequately demonstrated by SR 3.8.1.3. Any additional operation following the loss of offsite power tests is for maintenance purposes only and is therefore, not required by the surveillance. This change is consistent with NUREG-1430.

L8 CTS 4.6.1.2.c is not included in ITS. This Surveillance was originally requested to be included in CTS by a June 3, 1977 "generic letter" to ANO from NRC. Later, a second letter (December 17, 1979), which again identified this item, misquoted the originally requested surveillance. The original intent was to simulate a DG trip, restart, and subsequent reconnection of the DG to the bus. (Note that such a requirement was included as SR 4.8.1.1.2.c.6 in the 1980 version (Revision 4) of NUREG-0103, the B&W STS.) This was misquoted to require interruption of the offsite power source and subsequent reconnection of the onsite power source (the DG) to the bus. The misquote was not recognized and the latter Surveillance Requirement was incorporated as CTS 4.6.1.2.c via Amendment 60. However, neither of these tests simulate an activity assumed in the safety analysis beyond the loss of offsite power at the initiation of the event. Further, connection of the DG to the bus following a loss of offsite power, i.e., initial interruption of offsite power, is tested every 18 months, both alone and in conjunction with an engineered safeguards signal. Therefore, an additional test is either duplicative (if done for offsite power) or beyond the assumptions of the safety analysis (if done for onsite power), and is not included in ITS. This change is also consistent with NUREG-1430.

L9 The air compressors in the diesel generator air start system (CTS 3.7.1.C.4 and 4.6.1.4.a) are not specifically assumed to operate to mitigate any design basis accident. Rather the air start system receiver tank is assumed to contain sufficient air to start the diesel generator when necessary. Additional margin is provided by increasing the air supply 17 psig above that sufficient to start the DG. Typically, the diesel generator is assumed to start on the first attempt, or it is considered to be the single failure. The safety analysis does not require the compressor to run to recharge the receiver, only that the air start system be prepared to provide the start attempt when called upon, regardless of the source of the air pressure. This change is consistent with NUREG-1430.

L10 Not used

L11 CTS 4.6.1.4 requires that several tests be conducted during the monthly diesel generator start test. This testing is not required to be conducted in conjunction with the DG monthly start test in ITS. Although testing in conjunction with the DG start test may be convenient, it is not required. Testing separately from the start test is sufficient to verify capability of the support systems to perform their respective safety functions. This change is consistent with NUREG-1430.

L12 CTS 3.7.1.B requires that the ESAS AC distribution systems be powered from either the startup transformers or the unit auxiliary transformer in order to be considered OPERABLE (i.e., not allowing the distribution system to be considered OPERABLE if powered solely by the DG). The requirement for powering bus A3 and A4 from bus A1 and A2 is relocated to ITS 3.8.1, AC source OPERABILITY versus the CTS association with distribution system OPERABILITY. The specific power source is not related to the capability of the distribution system to perform its required safety function; an OPERABLE emergency diesel generator is also an acceptable power source for the AC distribution system to be considered OPERABLE. This change is consistent with NUREG-1430, and its definition of OPERABILITY, which, as applied to the distribution system, requires only the normal or emergency power source, but not both. Placing the requirement for bus A3 and A4 to be powered from bus A1 and A2 with the OPERABILITY of offsite sources, continues to impose the requirement and limitations if/when not met.

L13 Not used.

L14 CTS Table 4.1-1, item 32 requires a Monthly channel check and Quarterly channel test of the diesel generator protective relaying starting interlocks and circuitry. These are proposed to be incorporated into the 18 month start and load testing for loss of offsite power (NUREG SR 3.8.1.11), and for a loss of offsite power in conjunction with an engineered safeguards actuation signal (NUREG SR 3.8.1.19). This Frequency is consistent with the recommendations of Regulatory Guide 1.108, and takes into consideration unit conditions preferred for the performance of such testing.

CTS DISCUSSION OF CHANGES

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 The description of the equipment to which the requirements are applicable has been moved to the licensee controlled Bases. This information provides details of the method of implementation, which are not directly pertinent to the actual requirement. 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 process identified in Chapter 5 of the proposed ITS. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
3.7.1.A	Bases 3.8.1, LCO
3.7.1.F	Bases 3.8.1, SR 3.8.1.6
3.7.1.G	Bases 3.8.1, SR 3.8.1.6
Table 4.1-1, item 32	Bases 3.8.1, SR 3.8.1.7, SR 3.8.1.8
Table 4.1-1, item 33	Bases 3.8.1, SR 3.8.1.6
4.6.1.1	Bases 3.8.1, SR 3.8.1.2
4.6.1.3	Bases 3.8.1, BACKGROUND

LA2 The ambiguous limitation for inoperability of a diesel generator of "7 days in any month" in CTS 3.7.2.C is removed from the proposed Completion Time for an inoperable diesel generator. Such limitations on total time of inoperability are based on reliability concerns and are not addressed in the RSTS. This limitation will be addressed by the maintenance program in accordance with 10 CFR 50.63. The programmatic controls on diesel generator unavailability are sufficient to ensure the diesel generator receives adequate attention to maintain high reliability. Removal of these details from the Technical Specifications will have no significant effect on diesel generator OPERABILITY. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to the program and procedures will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

LA3 The description of the equipment to which the requirements are applicable has been moved to the licensee controlled Technical Requirements Manual (TRM). This information provides details of the method of implementation, which are not directly pertinent to the actual requirement. 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 will be controlled by 10 CFR 50.59. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
Table 4.1-1, item 33	TRM
4.6.1.1	TRM

LA4 The CTS 4.6.1.5 Surveillance is not specifically detailed in the proposed ITS. Programmatic controls on the Inservice Testing Program (IST) are sufficient to ensure the diesel generator fuel oil transfer pumps receive the required testing. Removal of these details from the Technical Specifications will have no effect on diesel generator OPERABILITY. The testing will be maintained in the IST and procedures. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to the IST and the procedures will be controlled by 10 CFR 50.55a and 10 CFR 50.59. This change is consistent with NUREG-1430.

LA5 CTS 4.6.2.4 requires that any battery "which has not been loaded while connected to its 125 VDC distribution system" to be loaded for 30 minutes each quarter. The associated Bases provide the added confirmation that this loading is simply "supplying the connected loads while maintaining the battery fully charged." This requirement is obviously being met for any connected battery charger by virtue of satisfying CTS 4.6.2.1, "Verify battery terminal voltage is $\geq$ 124.7 V on float charge," which is retained as ITS SR 3.8.3.1. As such, this single Surveillance adequately and completely encompasses CTS 4.6.2.4 (30 minute loading). The remaining purpose of CTS 4.6.2.4 is to imply alternating each battery charger with the spare charger each quarter. This operational maintenance practice is relocated from the Technical Specifications to the TRM. Since these details are not necessary to maintain or confirm OPERABILITY of the in-service charger, it can be moved to a licensee controlled document without a significant impact on safety. The TRM will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

LIMITING CONDITION FOR OPERATION (continued)

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE; or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in an OPERATING CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

3.8.1 RA A.2
3.8.1 RA B.2
3.8.1 RA C.1

L1

M16

1. ~~At least HOT STANDBY within the next 6 hours,~~
2. At least ~~HOT SHUTDOWN~~ MODE 3 within ~~the following 6~~ 12 hours, and
3. At least ~~COLD SHUTDOWN~~ MODE 5 within ~~the subsequent 24~~ 36 hours.

3.8.1 RA F.1
3.8.1 RA F.2

A1

This Specification is not applicable in ~~Cold Shutdown or Refueling Shutdown.~~ MODE 5 or 6

A3

BASES

A2

~~3.0.1 through 3.0.4 Establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):~~

~~"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shutdown the reactor or follow any remedial Action permitted by the Technical Specification until the condition can be met."~~

~~3.0.1 Establishes the Applicability statement within each individual Specification as the requirement for when (i.e., in which operational modes or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The Action requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.~~

~~There are two basic types of Action requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the Action requirements. In this case, conformance to the Action requirements provides an acceptable level of safety for unlimited continued operation as long as the Action requirements continue to be met. The second type of Action requirement specifies a time limit in which conformance to the Limiting Condition for Operation must be met. This time limit is the allowable outage time to~~

3.8.1

A2

BASES (continued)

initiated or that higher modes of operation are not entered when corrective action is being taken to obtain compliance with a Specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with Action requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a mode change. Therefore, in this case, if the requirements for continued operation have been met in accordance with the requirements of the specification, then entry into that mode of operation is permissible. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with Action requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower mode of operation. For the purpose of compliance with this specification the term 'shutdown' is defined as a required reduction in the REACTOR OPERATING CONDITION.

3.0.5 Delineates what additional conditions must be satisfied to permit operation to continue when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of the Limiting Condition for Operation for the normal or emergency power source, not the individual Limiting Condition for Operation statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.2.C provides for a 7 day out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to

3.8.1

A2

be consistent with the Limiting Condition for Operation statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.7.1.A requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. Specification 3.7.2.B provides a 24 hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits would also be inoperable. This would dictate invoking the applicable Limiting Condition for Operation statements for each of the applicable LCOs. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the Limiting Condition for Operation statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, shutdown is required in accordance with this specification.

During Cold Shutdown and Refueling Shutdown, Specification 3.0.5 is not applicable and thus the individual Action statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

{NOTE 14 referenced from CTS 3.7.2.H(2), PAGE 57}

TABLE 3.5.1-1 (Cont'd)

<LATER> (3.3 D)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the ~~electromatic relief valve power supply within the following 12 hours.~~ — LATER

<LATER> (3.3 D)

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, ~~otherwise, Note 14 applies.~~ — LATER

3.8.1 RA A.3, F.1, F.2 <LATER> (3.3 D)

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in HOT SHUTDOWN within the next 6 12 hours and in COLD SHUTDOWN within the following 30 hours. — & LATER — L4

<LATER> (3.3.A & 3.3 C)

15. This trip function may be bypassed at up to 10% reactor power. — LATER

<LATER> (3.3 A)

16. This trip function may be bypassed at up to 45% reactor power. — LATER

<LATER> (3.3 D)

17. ~~With no channel operable, within 1 hour initiate and maintain operation of the control room~~ emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation ~~system in the recirculation mode of operation.~~ — LATER

<LATER> (3.3 C)

19. ~~This function may be bypassed below 750 psig OTSG pressure.~~ Bypass is automatically removed when pressure exceeds 750 psig. — LATER

<LATER> (3.3 D)

20. ~~With one channel inoperable, (1) either restore the inoperable channel to operable status within 7~~ days, or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above. — LATER

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in HOT SHUTDOWN within 12 hours.

3.8.1

3.8.1
3.8.2
3.8.6

3.7 Auxiliary Electrical Systems

Applicability

Applies to the auxiliary electrical power systems.

Objectives

To specify conditions of operation for plant station power necessary to ensure safe reactor operation and combined availability of the engineered safety features.

A1

Specifications

MODES 1, 2, 3, & 4

3.8.1 APPL
3.8.2 APPL
3.8.6 APPL

3.7.1 The reactor shall not be heated or maintained above 200°F unless the following conditions are met (except as permitted by Paragraph 3.7.2) A12

3.8.1.a LCO

A. Any one of the following combinations of power sources operable:

1. Startup Transformer No. 1 and Startup Transformer No. 2.

2. Startup Transformer No. 2 and Unit Auxiliary Transformer provided that the latter one is connected to the 22KV line from the switchyard rather than to the generator bus. LA1 BASES

3.8.6 LCO

B. All 4160 V switchgear, 480 V load centers, 480 V motor control centers and 120 V AC distribution panels in both of the ESAS distribution systems are operable and are being powered from either one of the two startup transformers or the unit auxiliary transformer. L12

3.8.1.b LCO
3.8.2 LCO

C. Both diesel generator sets are operable each with:

SR 3.8.1.4

1. a separate day tank containing a minimum of 160 gallons of fuel,

SR 3.8.2.1

2. a separate emergency storage tank containing a minimum of 138 inches (20,000 gallons) of fuel, L2

SR 3.8.1.5

3. a separate fuel transfer pump, and

3.8.2 LCO

4. a separate starting air ~~compressor.~~ subsystem L9

D. DELETED

E. DELETED

3.8.1.a LCO

F. The off-site power undervoltage and protective relaying interlocks associated with required startup transformer power sources shall be operable per Table 3.5.1-1.

3.8.1.a LCO

G. The selective load-shed features associated with Startup Transformer No. 2 shall be operable if selected for auto transfer. LA1 BASES

<INSERT CTS 56A>

<CTS INSERT CTS56A>

Add ITS 3.8.2 Actions & Actions Note	Diesel Fuel Oil and Starting Air	L2
Add ITS 3.8.5	Inverters - Operating	M1

3.8.1
3.8.6

<INSERTS CTS 57A & 57B>

L3

3.8.1 APPL
3.8.6 APPL
3.8.6 RA D.1, D.2, E.1
3.8.1 RA F.1, F.2
3.8.1 RA G.1

MODES 1-4

3.7.2 A. The specifications in 3.7.1 may be modified to allow one of the following conditions ~~to exist after the reactor has been heated above 200F.~~ Except as indicated in the following conditions, if any of these conditions are not met, ~~a hot shutdown shall be initiated within 12 hours. If the condition is not cleared within 24 hours,~~ the reactor shall be brought to ~~cold shutdown~~ MODE 5 within an additional 24 hours. (A1) (M3) (A1)

3.8.1 RA A.3 — B. In the event that one of the offsite power sources specified in 3.7.1.A (1 or 2) is inoperable, reactor operation may continue for up to ~~24~~ 72 hours ~~if the availability of the diesel generators is immediately verified.~~ (L4) (L3)

3.8.1 RA B.4 — C. Either one of the two diesel generators may be inoperable for up to 7 days ~~in any month~~ provided that during such 7 days the operability of the remaining diesel generator is demonstrated ~~immediately~~ within 24 hours and daily thereafter, ~~there are no inoperable ESF components associated with the operable diesel generator,~~ and provided that the two sources of off-site power specified in 3.7.1.A(1) or 3.7.1.A(2) are available. (LA2) (L3) (L1) (L3)

3.8.1 RA B.3.2
3.8.1 RA B.2

3.8.6 RA A.1, B.1 — D. Any 4160V, 480V, or 120V switchgear, load center, motor control center, or distribution panel in one of the two ESF distribution systems may be inoperable for up to 8 hours, ~~provided that the operability of the diesel generator associated with the operable ESF distribution system is demonstrated immediately and all of the components of the operable distribution system are operable.~~ If the ESF distribution system is not returned to service at the end of the 8 hour period, Specification 3.7.2.A shall apply. (L3)

3.8.6 RA D.1, D.2

E. DELETED

F. DELETED

G. DELETED

H. ~~If the requirements of Specification 3.7.1.G cannot be met, either:~~

~~(1) place all Startup Transformer No. 2 feeder breakers in "pull-to-lock" within 1 hour, restore the inoperable interlocks to operable status within 30 days, or submit within 30 days a Special Report pursuant to Specification 6.12.5 outlining the cause of the failure, proposed corrective action and schedule for implementation; or~~ (L15)

~~(2) apply the action requirements of Table 3.5.1-1, Note 14.~~ (A1)

3.8.1 RA F.1, F.2
<see CTS pg 45f>

<CTS INSERT CTS57A>

for ITS 3.8.1 AC Sources - Operating

Add Required Action A.1 (M4)

Add "10 day" Completion Time for Required Action A.3 (M5)
and for Required Action B.4

Add Required Action B.3.1 (L6)

Add Required Action C.2 and Conditions D and E (L3)

<CTS INSERT CTS57B>

for ITS 3.8.6 Distribution Systems - Operating

Add "16 hour" Completion Time for Required Action A.1 (M5)
and for Required Action B.1
and for Required Action C.1

3.8.1
3.8.2
3.8.3
3.8.4
3.8.6

⟨INSERT 3.8.4 ACTIONS NOTE⟩ L5

3.8.3 LCO, 3.8.3 APPL, 3.8.4 APPL — 3.7.3 Both 125 VDC electrical power subsystems shall be operable ~~when the unit is above the cold shutdown condition.~~ MODES 1, 2, 3 & 4 A1

A. With one 125 VDC electrical power subsystem inoperable:

⟨LATER⟩ (5.0) — 1. ~~verify that there are no inoperable safety related components associated with the operable 125 VDC electrical subsystem which are redundant to the inoperable 125 VDC electrical power subsystem.~~ LATER

3.8.1 RA B.2 — 2. verify the operability of the diesel generator associated with the operable 125 VDC electrical subsystem immediately, and L1

3.8.3 RA A.1 — 3. restore the 125 VDC electrical subsystem to operable status within 8 hours.

3.8.3 RA B.1, B.2 — B. With one 125 VDC electrical power subsystem inoperable, and unable to satisfy the requirements or allowable outage times of 3.7.3.A.1, 3.7.3.A.2, or 3.7.3.A.3, the unit shall be placed in ~~hot shutdown~~ MODE 3 within 12 hours and in ~~cold shutdown~~ MODE 5 within an additional 24 hours. A1

3.8.4 LCO — 3.7.4 Battery cell parameters shall be within limits when the associated DC electrical power subsystems are required to be operable.

3.8.4 COND A — A. With one or more batteries with one or more battery cell parameters not within Table ~~4.6-1~~ 3.8.4-1 Category A or B limits:

3.8.4 RA A.1 — 1. Within 1 hour, verify pilot cell electrolyte level and float voltage meet Table ~~4.6-1~~ Category C limits, A1

3.8.4 RA A.2 — 2. Within 24 hours and once per 7 days thereafter, verify battery cell parameters meet Table ~~4.6-1~~ 3.8.4-1 Category C limits, and

3.8.4 RA A.3 — 3. Within 31 days, restore battery cell parameters to Table ~~4.6-1~~ Category A and B limits.

3.8.4 COND B & RA B.1 — B. With one or more batteries with one or more battery cell parameters not within Table ~~4.6-1~~ Category A or B limits and unable to satisfy the requirements or allowable outage times of 3.7.4.A.1, 3.7.4.A.2, or 3.7.4.A.3, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A. A1

C. With one or more batteries with electrolyte temperature of the pilot cell not within the limits of Specification 4.6.2.8, electrolyte temperature of representative cells not within the limits of Specification 4.6.2.6 or with one or more batteries with one or more battery cell parameters not within Table ~~4.6-1~~ 3.8.4-1 Category C limits, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

~~Bases~~

~~The electrical system is designed to be electrically self-sufficient and provide adequate, reliable power sources for all electrical equipment during startup, normal operation, safe shutdown and handling of all emergency situations. To prevent the concurrent loss of all auxiliary power, the various sources of power are independent of and isolated from each other.~~ A2

3.8.1
3.8.2
3.8.3
3.8.4
3.8.6

In the event that the offsite power sources specified in 3.7.1.A (1 or 2) are inoperable, the required capacity of one emergency storage tank plus one day tank (20, 160 gallons) will be sufficient for not less than three and one-half days operation for one diesel generator loaded to full capacity. (ANO-1 FSAR 8.2.2.3) The underground emergency storage tanks are gravity fed from the bulk storage tank and are normally full, while the day tanks are fed from transfer pumps which are capable of being cross connected at their suction and discharges and automatically receive fuel oil when their inventory is less than 180 gallons. Thus, at least a seven day total diesel oil inventory is available onsite for emergency diesel generator operation during complete loss of electric power conditions.

Technical Specification 3.7.2 allows for the temporary modification of the specifications in 3.7.1 provided that backup system(s) are operable with safe reactor operation and combined availability of the engineered safety features ensured. A2

Technical Specifications 3.7.1.F and 3.7.1.G provide assurance that the Startup Transformer No. 2 loads will not contribute to a sustained degraded grid voltage situation. This will protect ESF equipment from damage caused by sustained undervoltage.

The 125 VDC electrical power system consists of two independent and redundant safety related class 1E DC electrical subsystems. Each subsystem consists of one 100% capacity 125 VDC battery, its associated battery charger, and its distribution network. Additionally, there is one spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, no operable battery charger, or inoperable battery and no operable associated battery charger), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power subsystems with attendant loss of ES functions, continued power operation should not exceed 8 hours.

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational event or a postulated design basis accident. Cell parameter limits are conservatively established, allowing continued DC electrical system function even with Table 4.6-1 Category A and B limits not met.

With one or more cells in one or more batteries not within limits (i.e., Table 4.6-1 Category A limits not met, or Category B limits not met, or Category A and B limits not met) but within the Table 4.6-1 Category C limits, the battery is degraded but has sufficient capacity to perform its intended function. Therefore, the battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period of time. The pilot cell electrolyte level and float voltage are required to be verified to meet the Table 4.6-1 Category C limits within 1 hour (TS 3.7.4.A.1). These checks will provide a quick representative status of the remainder of the battery cells. Verification that the Table 4.6-1 Category C limits are met (TS 3.7.4.A.2) provides assurance that during the time needed to restore the parameters to within the Category A and B limits, the battery will still be capable of performing its intended function. This verification is repeated at 7 day intervals until the parameters are restored to within Category A and B limits. This periodic verification is consistent with the increased potential to exceed these battery parameter limits during these conditions.

With one or more batteries with one or more battery cell parameters outside the Table 4.6-1 Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured. Therefore, the battery must be immediately declared inoperable and the corresponding DC electrical power subsystem must be declared inoperable.

Additionally, other potentially extreme conditions, such as electrolyte temperature of the pilot cell falling below 60°F, average electrolyte temperature of representative cells falling below 60°F or battery terminal voltage below the limit are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

A2

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	
30. Decay heat removal system isolation valve automatic closure and interlock system	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.
31. Deleted				
32. Diesel generator protective relaying starting interlocks and circuitry	M	Q (R)	NA	
33. Off-site power undervoltage and protective relaying interlocks and circuitry	M	R(1)	R(1)	(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.
34. Borated water storage tank level indicator	W	NA	R	
35. Reactor trip upon loss of main feedwater circuitry	M	PC	R	

<LATER> (3.3 D) LATER

<LATER> (3.4 B) LATER

A1

SR 3.8.1.7 SR 3.8.1.8 L14 LA1 Bases LA3 TRM

SR 3.8.1.6 LA1 Bases

<LATER> (3.3 D) LATER

<LATER> (3.3 A) LATER

3.8.1

3.8.1
3.8.2

4.6 AUXILIARY ELECTRICAL SYSTEM TESTS

A1

~~Applicability~~

~~Applies to the periodic testing and surveillance requirements of the auxiliary electrical system to ensure it will respond promptly and properly when required.~~

~~Specification~~

LA3 TRM

4.6.1 Diesel Generators

SR 3.8.1.2 1. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 15 seconds. ~~The signal initiating the start of the diesel shall be varied from one test to another (start with handswitch at control room panel and at diesel local control panel) to verify all starting circuits are operable.~~ LA1 Bases

SR 3.8.1.3 The generator shall be synchronized from the control room and loaded to full rated load and allowed to run ~~until diesel generator operating temperatures have stabilized.~~ ≥ 1 hour A9 M15

SR 3.8.1.7 SR 3.8.1.8 2. A test shall be conducted once every 18 months to demonstrate the ability of the diesel generators to perform as designed by: M12

SR 3.8.1.7 a. simulating a loss of off-site power,

SR 3.8.1.8 b. simulating of loss of off-site power in conjunction with an ESF signal,

~~c. simulating interruption of off-site power and subsequent reconnection of the on-site power source to their respective busses, and~~ L8

SR 3.8.1.7 SR 3.8.1.8 d. operating the diesel generator for ~~≥1 hour after operating temperatures have stabilized.~~ 5 minutes L7

~~3. Each diesel generator shall be given an inspection once every 18 months following the manufacturer's recommendations for this class of standby service.~~ ~~(A one-time extension of this interval is allowed so that these may be performed during the 1R9 refueling outage, and completed no later than December 1, 1990.)~~ LA1 Bases A4

& (LATER) (5.0) 4. ~~During the~~ monthly ~~diesel generator test specified in paragraph 1 above,~~ the following shall be performed: L11 & LATER

SR 3.8.2.3 a. The diesel generator starting ~~air compressors shall be checked for operation and their ability to recharge the~~ air receivers. L9

SR 3.8.1.5 b. The diesel oil transfer pumps shall be checked for operability and their ability to transfer oil to the day tank.

SR 3.8.1.4 c. The day tank fuel level shall be verified.

SR 3.8.2.1 d. The emergency storage tank fuel level shall be verified.

3.8.2
3.8.3
3.8.4

<ADD LCO 3.8.2 ACTION B,C & SR 3.8.2.2 for New fuel oil> M11

SR 3.8.2.2 & <LATER>(5.0) e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment. LATER

SR 3.8.2.3 5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.

~~Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.~~ LA4

4.6.2 DC Sources and Battery Cell Parameters

SR3.8.3.1 1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.

SR 3.8.3.2 2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.

SR 3.8.3.3 3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.

~~4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.~~ LA5

SR3.8.4.1 5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.

SR3.8.4.4 6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.

SR3.8.4.3 7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.

SR3.8.4.2 8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.

~~4.6.3 Emergency Lighting~~

~~The correct functioning of the emergency lighting system shall be verified once every 18 months.~~ R TRM

<INSERT CTS100aA>

for ITS 3.8.1 AC Sources - Operating

Add SR 3.8.1.1 M9

Add SR 3.8.1.2 NOTE A7

Add SR 3.8.1.3 NOTE 1 A7

Add SR 3.8.1.3 NOTE 2 A7

Add SR 3.8.1.3 NOTE 3 A7

Add SR 3.8.1.7 NOTE A7

Add SR 3.8.1.8 NOTE A7

<INSERT CTS100aB>

for ITS 3.8.6 Distribution Systems - Operating

Add SR 3.8.6.1 M14

3.8.4

TABLE 3.8.4-1

3.8.4-1 A1

Table ~~1.6-1~~ (page 1 of 1)
Battery Cell Surveillance Requirements

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

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

(b) Corrected for electrolyte temperature.

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

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The emergency power system provides power requirements for the engineered safety features in the event of a DBA. Each of the two diesel generators is capable of supplying minimum required engineered safety features from independent buses. This redundancy is a factor in establishing testing intervals. The monthly tests specified above will demonstrate operability and load capacity of the diesel generator. The fuel supply and diesel starter motor air pressure are continuously monitored and alarmed for abnormal conditions. Starting on complete loss of off-site power will be verified by simulated loss-of-power tests once every 18 months.

The SR 4.6.2.1 verification of battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the battery charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery (2.15 V per cell average) and are consistent with the battery vendor allowable minimum volts per cell. The inability to meet this requirement constitutes an inoperable battery.

The SR 4.6.2.2 battery service test is a special test of the battery capability as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements. A modified performance discharge test may be performed in lieu of a service test. The inability to meet this requirement constitutes an inoperable battery.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the battery. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test and the test discharge rate must envelope the duty cycle of the service test if the modified performance discharge test is performed in lieu of a service test.

The SR 4.6.2.3 battery performance discharge test is a test of constant current capacity of a battery after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage. The inability to meet this requirement constitutes an inoperable battery.

Either the battery performance discharge test or the modified performance discharge test, described above, is acceptable for satisfying SR 4.6.2.3; however, only the modified performance discharge test may be used to satisfy SR 4.6.2.3 while satisfying the requirements of SR 4.6.2.2 at the same time.

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The acceptance criteria for this surveillance are consistent with IEEE-450. This reference recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The frequency for this test is normally 60 months. If the battery shows signs of degradation, or if the battery has reached 85% of its service life and capacity is < 100% of the manufacturer's rating, the frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its service life, the frequency is only reduced to 24 months for batteries that retain ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450, when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. A2

SR 4.6.2.4 requires that each required battery charger be capable of supplying the connected loads while maintaining the battery fully charged. This is based on the assumption that the batteries are fully charged at the beginning of a design basis accident, and on the safety function of providing adequate power for the design basis accident loads.

SR 4.6.2.5 verifies that the Table 4.6-1 Category A battery cell parameters are consistent with vendor recommendations and IEEE-450, which recommend regular battery inspections (at least once per month) including voltage, specific gravity, and electrolyte level of pilot cells.

The SR 4.6.2.6 verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450, which states that the temperature of electrolytes in representative cells (~10% of all connected cells) should be determined on a quarterly basis. Lower than normal temperatures act to inhibit or reduce battery capacity. This surveillance ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

SR 4.6.2.7 verifies that the Table 4.6-1 Category B battery cell parameters are consistent with vendor recommendations and IEEE-450, which recommend regular battery inspections (at least once per quarter) including voltage, specific gravity, and electrolyte level of each connected cell. In addition, within 24 hours after a battery discharge to < 110 V or a battery overcharge to > 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450, which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

The SR 4.6.2.8 verification that the temperature of the pilot cell is ≥ 60°F is consistent with a recommendation of IEEE-450, which states that the temperature of electrolytes in pilot cells should be determined on a monthly basis. Lower than normal temperatures act to inhibit or reduce battery capacity. This surveillance ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

Table 4.6-1 delineates the limits on electrolyte level, cell float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

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The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450, with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 4.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

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The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the battery vendor allowable minimum cell voltage and on a recommendation of IEEE-450, which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195. This value is characteristic of a charged cell with adequate capacity. According to IEEE-450, the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that is jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 with the average of all connected cells > 1.195. These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for float voltage is based on IEEE-450, which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limits of average specific gravity ≥ 1.190 is based on manufacturer recommendations. In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

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Footnotes (b) and (c) to Table 4.6-1 are ~~applicable to Category A, B, and C~~ specific gravity. Footnote (b) to Table 4.6-1 requires the above mentioned correction for electrolyte temperature. The value of 2 amps used in footnote (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition. This current provides, in general, an indication of overall battery condition.

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Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450. Footnote (c) to Table 4.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

The SR 4.6.3 testing of the emergency lighting is scheduled every 18 months and is subject to review and modification if experience demonstrates a more effective test schedule.

<u>REFERENCE</u>

FSAR, Section 8

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

<LATER> (3.3 D) b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LATER

c. Deleted

<LATER> (5.0) d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18. LATER

<LATER> (3.7) e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2. LATER

f. Deleted

g. Deleted

h. Deleted

i. Deleted

j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H. M8

<LATER> (3.3 D)

~~k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1~~

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.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.8: Electrical Power 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:

<u>**3.8 L1**</u>

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The AC Sources are used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. Equipment powered by the AC Sources, which may be considered as an initiator, continues to be evaluated for loss of function and previously determined appropriate ACTIONS for such inoperabilities continue to be required. Experience with the reliability of the AC sources indicates that the proposed increase in the Completion Time will not significantly increase the probability of a loss of electric power accident or of any other accident previously evaluated. The proposed ACTION continues to provide adequate assurance of OPERABLE required equipment and therefore, 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 corrective actions are taken to restore plant systems to OPERABLE status, as assumed 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?**

This change does not involve a significant reduction in a margin of safety since the OPERABILITY of the equipment and loss of function continue to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the equipment to OPERABLE status, rather than requiring a shutdown transient.

3.8 L2

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The AC Sources are used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. Equipment powered by the AC Sources, which may be considered as an initiator, continues to be assured of electrical power. The proposed increased restoration time involves parameters unrelated to initiating the failure of the AC Sources. As such the proposed time allowance for restoration will not increase the probability of any accident previously evaluated. The proposed changes allow additional time for restoration of parameters that have been identified as not immediately affecting the capability of the power source to provide its required safety function. The identified parameters are capable of being replenished during operation of the diesel generators and batteries, and the short additional Completion Time continues to provide adequate assurance of OPERABLE required equipment. 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 continue to ensure operable safety equipment is available. 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 parameter limits provide substantial margin to the parameter values that would be absolutely necessary for diesel generator and battery operability. When the parameters are less than their limits this margin is reduced. However, the availability of AC Sources continues to be assured since the allowed time for parameters to be less than their limits is short and the allowed levels for the parameters are adequate to provide the immediately needed power availability. Further, the parameters can be restored to within limits during the time provided by the reduced level of the parameter should they be required. Therefore, the reduction in margin is not a significant one.

3.8 L3

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The diesel generators (DGs) and offsite circuits are used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. They provide multiple sources of power to multiple trains of mitigating systems and components. The proposed conditions of concurrent inoperabilities retain sufficient sources of power for the necessary mitigating systems and components and do not initiate the loss of the remaining sufficient sources of power. As such the proposed conditions of concurrent inoperabilities will not increase the probability of any accident previously evaluated. Neither will the change allow continued operation without sufficient AC Sources to power the necessary safety equipment, or with any complete loss of safety function. The consequences of an event occurring during the proposed conditions of concurrent inoperabilities are the same as the consequences of an event occurring under the current ACTIONS. 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 proper actions are required, consistent with applicable regulatory guidance. 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 Completion Times to restore multiple inoperable AC Sources to OPERABLE status will minimize the potential for plant transients that can occur during the process of a plant shutdown by providing some time to restore the affected AC Sources to OPERABLE status prior to requiring a unit shutdown. Any reduction in the margin provided by having multiple sources of power to multiple trains of mitigating systems and components will be non-risk significant because of the proven reliability of the remaining sources of power. This reliability, indicated by experience, leads to a high improbability of the proposed conditions of concurrent inoperabilities existing concurrently with the need for the equipment to perform it safety functions and concurrent with the failure of the remaining equipment. This improbability is supported by allowing only short durations for such conditions. Further, component inoperability unrelated to the DGs provides no basis for questioning the OPERABILITY of the opposite train components for which the surveillances are current. Therefore, the reduced requirements for verification of opposite train component OPERABILITY will not result in a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.8 L4

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The offsite circuits can be used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. Equipment powered by the offsite circuits, which may be considered as an initiator, continues to be evaluated for loss of function and previously determined appropriate ACTIONS for such inoperabilities continue to be required. Experience with the reliability of the offsite circuits and the diesel generators which back them up indicates that the proposed increase in the Completion Time will not significantly increase the probability of a loss of electric power accident or of any other accident previously evaluated. As such the proposed increases in the Completion Times will not increase the probability of any accident previously evaluated. The consequences of an event occurring during the proposed Completion Times are the same as the consequences of an event occurring under the current ACTIONS. Therefore, the proposed change does not involve a significant increase to 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 actions are required, consistent with applicable regulatory guidance. 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 extended Completion Time to restore a required offsite circuit to OPERABLE status prior to requiring a unit shutdown is acceptable based on the overall probability of an event requiring the inoperable AC Sources during this time period. The extended Completion Time will minimize the potential for plant transients that can occur during shutdown by providing some time to restore the affected AC Sources to OPERABLE status prior to requiring a unit shutdown. Any reduction in the margin provided by having multiple sources of power to multiple trains of mitigating systems and components will be insignificant because of the proven reliability of the remaining sources of power. This reliability, indicated by experience, leads to a high improbability of the need for the equipment to perform it safety functions concurrent with an inoperable offsite circuit and concurrent with the failure of the remaining power source equipment. This improbability is still supported by allowing only short durations for such conditions.

3.8 L5

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Although the batteries may be in a degraded condition, allowing separate condition entry does not alter accident initiation assumptions nor does it alter component availability for accident mitigation as both batteries are capable of fulfilling their function. In the event a battery is determined to be inoperable, Actions exist to ensure that the appropriate activities occur. Therefore, this change does not involve a significant increase in the probability or 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 ACTIONS are taken for battery inoperability. 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?

Due to the conservative nature of battery calculations, operation with two batteries in a degraded condition does not result in a significant reduction in the margin of safety. The margin provided by this conservatism is reduced by the degraded condition but such temporary reduction is already permitted for one battery at a time. The change proposed is to allow the possibility of both trains to experience this non-significant margin reduction at the same time rather than only one at a time. The two trains provide a redundancy margin for single failure. Since both trains still have sufficient capacity to perform their function, there is no reduction in the redundancy margin by allowing separate condition entry. Should a battery be determined to be inoperable, proper guidance is provided to ensure the appropriate actions are taken. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.8 L6

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The diesel generators (DGs) are used to support mitigation of the consequences of an accident, and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. Minimizing diesel generator starts is recommended to avoid unnecessary diesel wear. Therefore, the elimination of an ACTION which requires starting and/or loading the DGs due to the inoperability of another DG power source will decrease the probability of the loss of electric power accident. The proposed ACTION continues to provide adequate assurance of OPERABLE DGs and therefore, 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. 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 OPERABILITY of the DGs continues to be determined in the same manner as before, whether another DG is inoperable or not. Since the elimination of redundant DG testing is only allowed when it is known that a common inoperability does not exist, the proposed change provides an equivalent assurance of the capability of the DGs to perform their safety function. The margin of safety is enhanced by reducing unnecessary DG starts, and by minimizing the reduced reliability that occurs while operating in test mode connected to offsite power and non-vital loads. Therefore, this change does not involve a significant reduction in a margin of safety.

3.8 L7

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The diesel generators are used to support mitigation of the consequences of an accident; and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. Demonstration of the capability of the diesel generator for extended operation is still provided by other surveillances. The proposed revision of the method of performing the Surveillance on the diesel generator will reduce diesel generator run time and, therefore, reduce accumulation of run time generated wear. Therefore, the proposed revision of the method of performing the Surveillance on the diesel generator will reduce the probability of the occurrence of the loss of electric power accident. Since the function of the diesel generator continues to be verified, and continues to be required to be OPERABLE, the change of the method of performance of the Surveillance 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 for diesel generator is based on availability and capacity of OPERABLE sources. The availability and capacity of the diesel generator continue to be confirmed with the required Surveillances. The revision of the method of performance of the Surveillance still provides assurance that the diesel generator will perform its required function when needed. Therefore, this change does not involve a significant reduction in a margin of safety.

3.8 L8

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The diesel generators (DGs) are used to support mitigation of the consequences of an accident, and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. Eliminating a test that is either duplicative (if done for offsite power) or beyond the assumptions of the safety analysis (if done for onsite power) results in eliminating electrical perturbations that could lead to a loss of power. As such, this elimination will not significantly increase the probability of any accident previously evaluated. The remaining Surveillances continue to provide adequate assurance of OPERABLE DGs. Therefore, this change does not involve a significant increase to 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 adequate surveillances are performed to assure availability of the DGs. 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 OPERABILITY of the DGs continues to be determined in the same manner as before. Since the elimination of a test that is either duplicative (if done for offsite power) or beyond the assumptions of the safety analysis (if done for onsite power) results in eliminating electrical perturbations that could lead to a loss of power, this change will increase the availability of the DGs. Furthermore, there is no impact on the assurance of the capability of the DGs to perform their safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

3.8 L9

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The diesel generators (DGs) are used to support mitigation of the consequences of an accident, and require adequate starting air to provide this function. However, the source of the air is not designated in the safety analysis. Therefore, the DG air start compressors are not assumed to operate in response to a design basis event. Neither are they considered as the initiator of any previously analyzed accident. As such, the removal of the air compressors from the required equipment does not involve a significant increase in the probability or 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 air start capability is available for the DGs. 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 source of the DG starting air is not pertinent to the safety analysis, only that adequate starting air is available. This continues to be required. Therefore, this change does not involve a significant reduction in a margin of safety.

3.8 L10 Not Used

3.8 L11

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The diesel generators (DGs) are used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. The efficacy of the testing of the diesel generator support systems is not reduced by performing the tests separately from the diesel generator start test. Since the function of the DG, and associated support components, continues to be verified, and continues to be required to be OPERABLE, the change in the grouping of these surveillances will not reduce the capability of required equipment to mitigate a postulated event nor increase the likelihood that the required equipment will fail as part of a loss of electric power event. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated nor 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 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 for the DG is based on the OPERABILITY of the DG which requires OPERABLE support systems. The capacity of the DG support systems continues to be confirmed with the required Surveillances. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.8 L12

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The Engineered Safeguards Actuation System (ESAS) AC distribution system is used to support mitigation of the consequences of an accident; however, it is not considered the initiator of any previously analyzed accident. As such the proposed revision of the requirements for determining OPERABILITY of the distribution system will not significantly increase the probability of any accident previously evaluated. Since the function of the system continues to be available regardless of the AC Source providing the power which is being distributed, the system continues to be capable of performing its safety function and therefore, OPERABLE. The change 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 not revise the requirements for OPERABLE offsite circuits, but will only revise the dependency of the distribution systems on the offsite circuit for OPERABILITY. 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 AC Distribution systems is based on capability of the system to provide adequate power to the components required to operate for prevention and mitigation of a transient or accident. This capability is not affected by the choice of AC Source which is providing the power. Thus, the choice of power source should not affect the administrative determination of OPERABILITY of the distribution system. Since the distribution system will perform its required function when powered from any required AC Source, this change does not involve a significant reduction in a margin of safety.

3.8 L13 **Not Used.**

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.8 L14

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The diesel generator starting and protective circuitry are used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR section 14.1.2.8. Experience with the reliability of the diesel generator starting and protective circuitry indicates that the proposed increase in the surveillance interval will not significantly increase the probability of a loss of electric power accident or of any other accident previously evaluated. Since the function of the diesel generator continues to be verified, and continues 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 for diesel generator is based on availability and capacity of OPERABLE sources. The availability and capacity of the diesel generator continue to be confirmed with the required Surveillances. The revision of the Surveillance Frequency still provides assurance that the diesel generator will perform its required function when needed. Therefore, this change does not involve a significant reduction in a margin of safety.

ITS DISCUSSION OF DIFFERENCES

ITS Section 3.8: Electrical Power Systems

1 NUREG LCO 3.8.1.c, NUREG 3.8.1 Condition F, & NUREG SR 3.8.1.11, SR 3.8.1.18, & SR 3.8.1.19 - The automatic load sequencing design for this unit is provided by individual timers in the circuitry for each component which are initiated by a loss of power, and subsequent restoration of power, to the circuit. The design does not include an overall "train" sequencer which controls all loading to the diesel generator or engineered safety features bus. Therefore, there is no need to separately identify this component in LCO 3.8.1, nor in a separate Condition if inoperable (RSTS Condition F). Additionally, the diesel generator may be directly, adversely affected by an inoperable sequencing timer. Therefore, the sequencing timers are addressed with operability of the diesel generators.

2 NUREG LCO 3.8.1, Required Action A.3, second Completion Time, and Required Action B.4 Completion Times - The CTS 3.7.2.C Completion Time of 7 days for the diesel generators (DGs) is retained. Similarly, the overall Completion Time for "failure to meet the LCO" is extended to 10 days. The overall Completion Time is the additive time for an inoperable DG and an inoperable offsite feed as though they occur back-to-back. The 7 day Completion Time has been previously found acceptable and the plant specific risk assessment has not identified the allowed outage time for the AC Sources to represent an unacceptable risk. This change is consistent with current license basis.

3 NUREG SR 3.8.1.2, SR 3.8.1.2, Notes 1 & 3, SR 3.8.1.3, Note 4, and SR 3.8.1.7 - The diesel generator design does not provide for gradual acceleration. Consequently, all starts are fast starts and subject to the 15 second acceptance criteria. Therefore, SR 3.8.1.2 is revised to be equivalent to NUREG SR 3.8.1.7, and SR 3.8.1.7 and the Notes related to use of SR 3.8.1.7 are unnecessary. These changes are consistent with current license basis.

Further, NUREG SR 3.8.1.2, Note 1 would be inconsistent with the remainder of the NUREG. Any test which satisfies the requirements of another test may be credited to satisfy both. This is standard practice and fully satisfies the requirements without specific identification. A specific note here would present confusion for other SRs which may be satisfied by alternate testing but for which the specific SR does not contain a similar note. This change incorporates TSTF-253.

4 NUREG SR 3.8.1.2 & SR 3.8.1.3 Frequency, & Table 3.8.1-1 - The variable DG test Frequency requirements are not included per Generic Letter 94-01. Implementation of the provisions of the maintenance rule for the DGs, including the applicable regulatory guidance, provides the program to assure DG reliability and performance. Further, a monthly Frequency is consistent with CTS SR 4.6.1.1. This change is consistent with current license basis.

5 NUREG SR 3.8.1.3, Note 3 - The requirement to restrict the performance of this SR to only one DG at a time is not included since such a restriction is inconsistent with the content of the remainder of the NUREG and is unnecessary. Additionally, such a Note does not allow simultaneous starts such as may be conducted under NUREG SR 3.8.1.20 to fulfill the test requirements of NUREG SR 3.8.1.3 even though the start would otherwise fulfill all test criteria. The NRC Staff has previously concluded (see Generic Letter 91-04) that the TS need not restrict surveillances as only being performed during shutdown. The logic for that conclusion is readily extended to other similar restrictions such as this.

Administrative controls and risk insights have, to-date, provided adequate restriction such that when surveillances are performed, proper regard is considered for their effect on the safe operation of the plant. If the simultaneous performance of a surveillance on redundant components would adversely affect safety, the SR is postponed on at least one of the components. This administrative control will continue to be imposed, and the surveillance postponed until the unit is in a condition or mode that is consistent with the safe conduct of that surveillance. This requirement has been adequately performed under administrative control in the past (i.e., it is not in the current TS) and is proposed to continue to be administratively controlled. This change is consistent with current license basis.

6 NUREG SR 3.8.1.4 & SR 3.8.1.6 - The design of the DG fuel oil system includes an engine mounted day tank; but not a day tank separate from the engine mounted tank. This change is consistent with current license basis.

7 NUREG SR 3.8.1.8, Note, SR 3.8.1.11, Note 2, SR 3.8.1.19, Note 2, SR 3.8.4.7, Note 2, & SR 3.8.4.8, Note - The requirement to restrict the performance of this SR from MODES 1 and 2 (or MODES 1 through 4, as appropriate for the respective SRs) is not included since such a restriction is inconsistent with the remainder of the NUREG and is unnecessary. The NRC Staff has previously concluded (see Generic Letter 91-04) that the TS need not restrict surveillances as only being performed during shutdown. Administrative controls and risk insights have, to-date, provided adequate restriction such that when surveillances are performed during power operation, proper regard is considered for their effect on the safe operation of the plant. If the performance of a surveillance during plant operation would adversely affect safety, EOI/ANO-1 has postponed, and will continue to postpone, the surveillance until the unit is in a condition or mode that is consistent with the safe conduct of that surveillance. This requirement has been adequately performed under administrative control in the past (i.e., it is not in the current TS) and is proposed to continue to be administratively controlled. This is consistent with current license basis.

8 NUREG SR 3.8.1.8, Note, SR 3.8.1.11, Note 2, SR 3.8.1.19, Note 2, SR 3.8.4.7, Note 2, & SR 3.8.4.8, Note - Incorporated TSTF-008, Rev 2.

ITS DISCUSSION OF DIFFERENCES

9 NUREG SR 3.8.1.9 and SR 3.8.1.10 - These SRs for testing full and partial load rejection capability of the DGs are not adopted. Failure of one DG is assumed to occur during a design basis accident without specifying the mechanism for the failure. While capability to operate following these events (i.e., DG load rejection) is desirable, it is not required by the safety analysis. Further, such surveillance requirements do not exist in the CTS. This change is consistent with current license basis.

10 NUREG SR 3.8.1.12 & SR 3.8.1.20 - SR 3.8.1.12 is not included since it does not confirm the capability of the tested components to perform any function required or assumed by the safety analysis. As such, it is not appropriate for a TS requirement. The safety analysis assumes/requires loading of the DG only following a loss of offsite power. If offsite power is available, it is utilized, and if it is lost, the DG is started and loaded as tested by proposed SR 3.8.1.7 or SR 3.8.1.8. Therefore, the standby start of the DG is not utilized by the safety analysis.

SR 3.8.1.20 is not included since it confirms only that the unit design controls have been appropriately implemented. This is not typical for Technical Specifications and is not proposed to be incorporated. The design controls for the unit have been adequately confirmed through post modification testing (i.e., a similar SR is not included in the CTS), and are proposed to continue to be confirmed in this manner. These changes are consistent with current license basis.

11 NUREG SR 3.8.1.13 - The ANO-1 design does not include automatic bypassing of the emergency diesel generator trips during emergency operation following an engineered safeguards actuation signal. This change is consistent with current license basis.

12 Not used.

13 NUREG 3.8.4, Condition A, NUREG 3.8.9, Conditions B and C, and NUREG 3.8.7 Bases - The NUREG 3.8.4 and 3.8.9 Conditions are revised to retain the CTS 3.7.3 allowed time for continued operation with an inoperable battery, battery charger, or DC electrical power distribution subsystem. As long as there is no "loss of function" identified, the CTS time frame of 8 hours has been previously determined to be acceptable and the plant specific risk assessment has not identified the allowed outage time for the DC Sources to represent an unacceptable risk. The "loss of function" will continue to be determined in accordance with the SFDP and if identified, appropriate actions will be taken in accordance with the Specification for the lost function. NUREG LCO 3.8.9, Condition B is also revised to allow 8 hours for one 120 VAC vital bus electrical power distribution subsystem inoperable. This time period is consistent with the proposed ITS 3.8.9 Condition A and C Completion Times for an AC or DC subsystem inoperability. The Bases provided for a 2 hour Completion Time also support an 8 hour Completion Time. The NUREG 3.8.7 Bases were revised to incorporate reference to the correct Completion Time. These changes are consistent with current license basis.

14 NUREG 3.8.4 and 3.8.5 - NUREG SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, and SR 3.8.4.5 are not proposed to be adopted. NUREG SR 3.8.4.2, SR 3.8.4.3 and SR 3.8.4.4 are omitted since visible corrosion does not necessarily mean the battery is inoperable (as indicated in the Bases for NUREG SR 3.8.4.4). Also, the bracketed values of resistance specified in the NUREG are vendor recommended values; that is, values at which some action should be taken, not necessarily when the OPERABILITY of the battery is in question. Therefore, NUREG SR 3.8.4.5 is also proposed to be omitted. The safety analyses do not assume a specific battery resistance value, but typically assume the batteries will supply adequate power. Therefore, the key issue is the overall battery resistance. Between surveillances, the resistance of each connection varies independently from all the others. Some of these connection resistances may be higher or lower than others, and the battery may still be able to perform its function and should not be considered inoperable solely because one connector's resistance is high. Overall resistance is a direct impact on OPERABILITY, however, it is adequately determined as acceptable through completion of the battery service and discharge tests. Finally, acceptable resistance is currently determined through the battery service and discharge tests since the CTS does not include a surveillance equivalent to NUREG SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, and SR 3.8.4.5. Similarly, visual indication of physical damage or abnormal deterioration is cause for investigation, but does not necessarily mean that the battery could not perform its function if called upon. Such indication would be documented under the station's condition reporting program and evaluated consistent with the extent of the damage or deterioration. Therefore, these parameters are proposed to be administratively controlled under the maintenance program.

15 NUREG SR 3.8.1.14, SR 3.8.1.15, and SR 3.8.1.16 - These SRs for testing a twenty-four hour run, hot restart capability, and synchronization capability of the DGs are not adopted. The capability to run for an extended duration is satisfactorily demonstrated by the one hour run which allows the DG to reach a stable engine temperature. Operation beyond this time frame does not provide significant additional information with regard to the DGs capability to perform for long periods as designed. Also, a hot restart capability is considered to be a requirement beyond the current licensing basis. Failure of one DG is assumed to occur during a design basis accident without specifying the mechanism for the failure. While capability to operate following a trip is desirable, it is not required by the safety analysis. Finally, synchronization with offsite power sources is regularly conducted during testing of the DGs, and an additional test of this capability provides no significant additional information. Further, such surveillance requirements do not exist in the CTS.

16 Not used.

ITS DISCUSSION OF DIFFERENCES

17 NUREG LCO 3.8.2, LCO 3.8.5, LCO 3.8.8, & LCO 3.8.10: Electrical requirements during shutdown MODES are not included in the ANO ITS. This is consistent with the presentation of these support systems in the CTS. Rather than, in accordance with NUREG-1430, explicit electrical power source requirements, which are not completely tied to the supported equipment, the ITS (as well as the CTS) rely solely on the definition of OPERABILITY for the necessary OPERABILITY of electrical power sources and distribution subsystems. This is consistent with the current licensing basis.

18 Not used.

19 NUREG 3.8.4 - The NUREG SR 3.8.4.7 Note 1 restriction to use the substitution of the modified performance discharge test for the service test only "once per 60 months" is not adopted. As indicated in the Bases, the modified performance discharge test is required to envelope the duty cycle of the service test. It is therefore, a conservative test and should be allowed at any time. Further, IEEE-450, on which the service test and substitution of a modified performance discharge test are based, contains no such limitation. Finally, the CTS for ANO-1 does not contain this limitation, nor the limitation for use of a "modified" performance discharge test. This change is consistent with current license basis as it will exist following NRC approval of the April 9, 1999 (1CAN049902) license amendment request associated with the 125 VDC electrical system (DOC-A14).

20 NUREG SR 3.8.1.5, SR 3.8.3.5 & SR 3.8.3.6 - These SRs are not proposed to be adopted. These SRs are omitted since their inclusion would be inconsistent with the remainder of the NUREG (and the philosophy of 10 CFR 50.36 as revised to incorporate the Policy Statement) which does not include "preventive maintenance" requirements. The Bases for each of these SRs indicates that they are "preventive maintenance" requirements and that their failure does not necessarily mean that the equipment is not capable of meeting the definition of OPERABLE. The safety analyses do not specifically address these requirements and they are not required to assure the equipment's capability to perform its respective safety function. The preventive maintenance program, along with the Diesel Generator Fuel Oil Testing Program and the routine start testing of the DGs, will adequately identify detrimental fuel oil parameters. Therefore, providing additional surveillances is an unnecessary deviation from the normal scope of TSs, and these SRs are not included. The deletion of NUREG SR 3.8.3.6 is consistent with Generic Traveler TSTF-002, Rev. 1.

21 NUREG SR 3.8.1.17 - The design of the diesel generator logic does not include an ESAS override of the test mode as described in this SR. As such, a similar SR is not included in the CTS and is not proposed to be included in the ITS. This change is consistent with current license basis.

22 NUREG 3.8.6 - The LCO is revised to omit the "Train A and Train B" terminology which is not used, with respect to batteries, at ANO-1. This change is administrative in that the Bases are adequate to describe the batteries to which the LCO is applicable.

23 NUREG SR 3.8.1.11 and SR 3.8.1.19 - The details of the acceptance criteria for these SRs are revised to match CTS 4.6.1.2 acceptance criteria of "demonstrate the ability of the DGs to perform as designed" for a loss of offsite power and for a loss of offsite power in conjunction with an ES signal, and to be consistent with the changes made to NUREG SR 3.8.1.2. Steady state voltage and frequency requirements are not included in the current licensing basis (see DOD 40). This change is consistent with CTS.

24 Not used.

25 NUREG LCO 3.8.3 - The lube oil requirements are not included in this Specification since the design does not provide for a measurable indication of the amount of lube oil available. The design provides only a dip stick which indicates "sufficient" lube oil available, i.e., above the minimum mark. When lube oil is below the maximum, but above the minimum, action is initiated to add lube oil to the system. Thus, sufficient lube oil is always available. The administrative controls and maintenance practices have, to-date, provided adequate restriction such that when surveillances are performed, proper regard is considered for their effect on the safe operation of the plant. If the lube oil is below the minimum mark, the DG is considered inoperable and if the lube oil is between the maximum and minimum marks, prompt actions is initiated to restore the lube oil level. These administrative controls will continue to be imposed, and this requirement, which is not in the CTS, is proposed to continue to be administratively controlled. This change is consistent with current license basis.

26 NUREG Bases - This change is editorial in nature and incorporated only to provide nomenclature consistent with that used in other plant related documents.

27 NUREG Bases - This change provides plant specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures.
B 3.8.1 BACKGROUND - Added discussion of DG "intended service" rating, moved information regarding definition of an offsite circuit to the LCO section, and deleted incorrect information regarding the capability of the offsite circuit transformers.
B 3.8.1 BACKGROUND, LCO & NUREG SR 3.8.1.8- Revised discussion of offsite circuits design.
B 3.8.1 LCO - Revised offsite circuits automatic transfer design discussion since only one circuit is normally available for automatic transfer, and neither circuit is the normal ESF bus power source. The normal power source is the unit auxiliary transformer, and startup transformer No. 2 is normally unavailable for automatic transfer.
B 3.8.1 ACTIONS - Removed references to non-applicable guidance documents. These guidance documents were not used in the development of the original surveillance requirements, and are not applicable to the proposed SRs.
B 3.8.1 ACTIONS A.1, A.2, B.1, B.2, C.1 and C.2 - The turbine driven emergency feedwater (EFW) pump is redundant to the motor driven pump. There are only two EFW pumps.
B 3.8.1 Required Actions A.1 & B.1 - Added missing Completion Time Bases.

B 3.8.1 Required Action A.2 – Clarified (consistent with the Required Action presentation) that the Required Action applies to conditions when offsite power is not "available to supply." The ANO-1 design is such that offsite power is not normally connected, but available for automatic transfer to the ES buses.

B 3.8.1 Required Actions D.1 & D.2 - Omitted discussion of susceptibility to a single failure that is not be true for some situations that would result in entry into this condition.

B 3.8.1 SRs - Removed references to non-applicable guidance documents. These guidance documents were not used in the development of the original surveillance requirements, and are not applicable to the proposed SRs.

B 3.8.1 NUREG SR 3.8.1.2, SR 3.8.1.11, & SR 3.8.1.19 - The DG engine coolant is not a forced circulation while the engine is not running; natural circulation provides the cooling.

B 3.8.1 NUREG SR 3.8.1.3 - The discussions of DG power factors omitted since these are not included in the SR or in the CTS.

B 3.8.1 NUREG SR 3.8.1.4 - The DG fuel oil day tank volume is revised as appropriate to reflect unit specific design.

B 3.8.1 NUREG SR 3.8.1.6 (ITS SR 3.8.1.5) - The Bases are revised to omit discussion of automatic fuel oil transfer since this is not required.

B 3.8.1 NUREG SR 3.8.1.8 (ITS SR 3.8.1.6) - The Frequency discussion is revised to match the DG testing Frequency.

B 3.8.1 NUREG SR 3.8.1.11 (ITS SR 3.8.1.7) - The loss of offsite power testing is revised to reflect unit specific design.

B 3.8.1 NUREG SR 3.8.1.19 (ITS SR 3.8.1.8) - Discussion of Regulatory Guide revised so that the recommendations are not stated to be "requirements" since Regulatory Guides are not requirements, but rather provide only guidance on an acceptable method to implement the requirements.

B 3.8.1 NUREG SR 3.8.1.19 (ITS SR 3.8.1.8) - The loss of offsite power testing in conjunction with an ES signal is revised to reflect unit specific operating restrictions necessary to preserve decay heat removal during the conduct of the surveillance.

28 Not used.

29 NUREG Bases - This change provides plant specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures.
B 3.8.3 BACKGROUND, LCO, ACTION A.1 & NUREG SR 3.8.3.1 - The DG fuel oil storage tank volume is sufficient for 3.5 days of operation at full load.
B 3.8.3 BACKGROUND - The DG fuel oil system description is revised to reflect the interconnections of storage tanks and day tanks.
B 3.8.3 BACKGROUND, ACTIONS and SRs - Removed references to non-applicable guidance documents. These guidance documents were not used in the development of the original surveillance requirements, and are not applicable to the proposed SRs.
B 3.8.3 BACKGROUND & NUREG SR 3.8.3.3- The DG fuel oil properties discussion is revised for consistency with the ANO-specific design of the fuel oil storage system.
B 3.8.3 BACKGROUND, LCO, and SR 3.8.3.4 - The DG air start system is revised to reflect the unit specific design.
B 3.8.3 APPLICABLE SAFETY ANALYSES - Revised to refer to the Applicable Safety Analyses for LCO 3.8.1.

30 NUREG Bases - This change provides plant specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures.
B 3.8.4 BACKGROUND & LCO - The Battery System description is revised to reflect plant design and nomenclature and eliminate non-applicable discussions.
B 3.8.4 BACKGROUND, ACTIONS and SRs - Removed references to non-applicable guidance documents. These guidance documents were not used in the development of the original surveillance requirements, and are not applicable to the proposed SRs
B 3.8.4 NUREG SR 3.8.4.8 - "as found" reference for performance discharge testing is not per recommendations of IEEE-450-1995, and is deleted.
B 3.8.4 ACTIONS B.1 and B.2 & NUREG SR 3.8.4.7 - Removed references to non-applicable guidance documents. These guidance documents were not used in the development of the original surveillance requirements, and are not applicable to the proposed SRs.
B 3.8.4 SR 3.8.4.8 – The definition of battery degradation is revised from ≥ 10% to > 10% to be consistent with IEEE-450, 1995.
B 3.8.6 APPLICABLE SAFETY ANALYSIS - unit specific nomenclature and references provided.
B 3.8.6 SURVEILLANCE REQUIREMENTS - Battery parameters were revised to reflect plant procedure acceptance criteria.

31 NUREG Bases - This change provides plant specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures.
B 3.8.7 BACKGROUND, LCO, ACTION A.1 & NUREG SR 3.8.7.1 - The Inverters description is revised to reflect plant design and nomenclature.

32 NUREG Bases - This change provides plant specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures.
B 3.8.9 BACKGROUND, LCO, ACTIONS B.1 & C.1, & Table B 3.8.9-1 - The distribution system description is revised to reflect plant design and nomenclature.

33 NUREG 3.8.1, 3.8.4, 3.8.6, 3.8.7, and 3.8.9 Bases, Applicable Safety Analysis – removed inaccurate statement of analysis assumptions. The implication of the statement being deleted is that a single failure is always assumed and that a loss of offsite power is required to be applied to all accidents. There are accidents that do not assume a loss of offsite power in the ANO-1 SAR. There are also accidents in the ANO-1 SAR that does not assume a single failure. Elimination of the inaccurate statement has no impact on the remainder of the Bases.

34 Not used.

35 NUREG LCO 3.8.9 Condition B - This Condition and the Required Action are revised from an individual bus basis to a subsystem basis. The LCO is written on a subsystem basis because the redundancy is provided on a subsystem basis. Each subsystem contains two 120 V AC vital buses, but even with the loss of both of the buses in one subsystem, the other subsystem is adequate to provide for safe shutdown. A subsystem basis is also consistent with the Bases description of the need for the Required Action.

36 NUREG LCO 3.8.4 Required Action B.1 and LCO 3.8.9 Required Action D.1 Completion Times are revised from 6 hours to 12 hours consistent with CTS 3.7.3.B. NUREG LCO 3.8.7 (for Inverters) Required Action B.1 Completion Time is also revised from 6 hours to 12 hours for consistency with other electrical section shutdown actions. Since the CTS contains no explicit requirements for Inverters, this change is not directly related to current license basis, however, it is consistent with the current license basis for the remainder of the electrical section shutdown actions.

37 NUREG LCO 3.8.1 Required Actions A.1 and B.1 - The periodic Completion Time for these Required Actions is eliminated (the initial performance remains as per the NUREG) to be consistent with the CTS. This resultant Completion Time continues to provide the verification of an OPERABLE highly reliable power source on an increased Frequency as indicated in the Bases for these Required Actions. Since, these periodic verifications are not in the CTS, this is consistent with the current license basis.

38 NUREG LCO 3.8.1 - The term "qualified" is omitted from this LCO since the term is not used to define the acceptable offsite circuits in the Bases. The acceptable circuits are identified in the Bases for the LCO without indication of what would make them qualified. Therefore, omitting this term is an editorial change with no impact on the requirements. This change is consistent with current license basis.

39 NUREG SR 3.8.1.8 - This SR revised to reflect the unit design for utilization of offsite circuits. When power is supplied from the Unit Auxiliary Transformer during normal operation, a unit trip detected by the main generator lockout relays would initiate a fast transfer to the selected preferred power supply. (ST1 is typically the selected preferred offsite circuit power supply. ST2 is typically disabled from auto transfer by placing the bus feeder breakers control switches in a pull-to-lock position, because ST2 is shared by both units and has limited capacity. This is to ensure ST2 is available to supply ES bus loads.) The transfer would open the Unit Auxiliary Transformer supply breaker and simultaneously close the preferred power supply breaker (i.e., the selected startup transformer's breaker) in a few cycles provided that the preferred power supply had

acceptable voltage and was ready to accept the load. If the fast transfer to the startup transformer fails, then there will be an attempt to automatic slow transfer to the same startup transformer initiated by 4.16 KV bus A1 (A2) undervoltage auxiliary relays (after loads are shed from the bus.) If both fast and slow transfers fail to take place, then the transfer would have to be manual to the alternate offsite circuit, i.e., typically ST2. If a load is transferred to ST2 manually upon a unit trip, sufficient load is shed to ensure degraded voltage isolation does not occur. Thus, this SR is written to test the automatic transfer to the selected offsite circuit and the manual transfer to the remaining offsite circuit. As indicated in the Bases, this test maintains the requirements included in CTS Table 4.1-1, item 33. This change is consistent with current license basis.

40 NUREG SR 3.8.1.2 - This SR is revised to omit the specific acceptance criteria for steady state voltage and frequency. CTS 4.6.1.1 requires only the DG be demonstrated "ready for loading." This acceptance criteria is retained in the ITS. The Bases indicate that this acceptance criteria is met if the DG exceeds the minimum voltage of 3750 V within the time allowed. All acceptance criteria have been administratively controlled to-date and are proposed to continue to be so controlled. This change is consistent with current license basis.

41 NUREG SR 3.8.1.3 - The upper limit for loading of the DGs is not adopted. Such an upper limit is unnecessarily restrictive for a one hour run when set at 100% of the rated load. The DGs ratings include: 2600 kW continuous service, 2750 kW intended service, 2850 kW for 2000 hours, and 3000 kW for 4 hours. Even when considering the "intended service rating," the DG is clearly capable of operating at up to 3000 kW for the one hour test without affecting OPERABILITY. The loading of the DG can easily be controlled between 2750 kW and 3000 kW; therefore, no upper limit is necessary. Further, as indicated in the NUREG Bases, the upper limit is only to prevent routine overloading which may result in more frequent inspections. Again, this does not affect OPERABILITY, but only availability which is licensee controlled under the maintenance program. This change is consistent with current license basis.

The lower limit of 2475 kW is based on the CTS 4.6.1.1 requirement for testing at "full rated load" and the Regulatory Guide 1.9 Section C.2.2 discussion that the full load testing is to be conducted between 90% to 100% of the continuous load rating. Since the intended service rating is used for load rating, the minimum test loading is based on 90% of the intended service full load. Additional information has been added to the Bases to discuss the ANO-1 treatment of instrument uncertainty for this parameter. This change is consistent with current license basis.

42 NUREG LCO 3.8.3 - NUREG Conditions E and F and NUREG SR 3.8.3.4 are revised to refer to the "required" air start receivers. The air start system for each DG consists of two redundant banks of two tanks each. One bank of two tanks is sufficient to provide margin to the required start attempt. Hence, only one bank is "required" by the LCO. This is consistent with current license basis (CTS 4.6.1.5).

43 NUREG LCO 3.8.9 - SR 3.8.9.1 is revised to omit voltage from the acceptance criteria. The installed instrumentation does not provide for direct voltage readings for each required subsystem. Rather, voltage is generally presumed to be sufficient if there is no indication of an undervoltage on the 4160 V buses. Therefore, specific requirements for verification of voltage are not included. This change is consistent with current license basis.

44 NUREG LCO 3.8.9 - The Frequency for SR 3.8.9.1 is revised from 7 days to 31 days. Incorrect breaker alignments are readily identified by either trouble alarms or other indications of a "dead" bus. Therefore, a 31 day Completion Time is sufficient for this Surveillance.

45 NUREG LCO 3.8.7 - The LCO is revised to allow switching between the inservice inverter and a swing inverter without entering an ACTION. Both inverters are typically aligned to the alternate AC source prior to the load transfer. While this is not a frequent operation, it is unnecessarily restrictive to require entry into an ACTION to implement a design feature. An allowed time of 2 hours provides ample time to perform the transfer or return to the original configuration if the transfer can not be completed for some unforeseen reason.

46 NUREG LCO 3.8.7 - The Frequency for SR 3.8.7.1 is revised from 7 days to 31 days. Trouble alarms are provided to identify both low inverter voltage and incorrect alignment. Therefore, a 31 day Frequency is sufficient for this Surveillance.

47 NUREG LCO 3.8.9 - The ACTIONS for ITS 3.8.9 are revised to be applicable for more than one subsystem inoperable. Condition F identifies the appropriate ACTION for two or more electrical power distribution subsystems inoperable that result in a loss of function. However, no Condition is applicable if the inoperability of two or more subsystems does not result in a loss of function. This change allows continued operation if no loss of function exists. In addition, the Bases statements of "and a loss of function has not yet occurred" are not incorporated since this is inconsistent with usage and with the remainder of the NUREG. Such wording is inconsistent with usage since Conditions A, B, and C would still be applicable and entered, but moot since Condition E would require a shutdown. Should the lost function be restored, the Completion Time for Condition A, B, or C, as applicable, would have begun at the time of initial entry into the Condition, not at the time of restoration of the lost function. Further, the Bases wording is inconsistent with numerous other such Conditions in the NUREG which do not identify applicability based on no loss of function.

48 NUREG SR 3.8.4.6 is not adopted. This NUREG Surveillance demonstrates the design capability of the charger and is not directly related to verification of the lowest functional level required to confirm the assumed safety related function, and is not a CTS (current licensing basis) requirement. Adopting this NUREG Surveillance at ANO-1 would result in an outage impact and/or non-trivial temporary test setup.

The ANO-1 battery charger design includes a fully redundant spare charger for each of the two inservice chargers. It is standard practice to alternate chargers to equalize run times. This results in continuous verification of the charger's capability to carry nominal DC loads while maintaining the battery fully charged (which satisfies ITS SR 3.8.3.1). Furthermore, each outage (once per 18 months) one charger will be used to recharge the battery after its required service or performance discharge test. This test will confirm the charger capability to function at its current-limit, and continue to fully recharge the battery. This capability will be demonstrated by alternating chargers such that each charger is utilized once per 36 months (nominally). Together, the required ITS SR 3.8.3.1 and the other licensee-controlled performance tests and monitoring will continue to adequately verify the necessary safety function of the chargers. This is consistent with the current license basis.

49 NUREG Bases 3.8.6 - This editorial change adds Bases for the ITS 3.8.4 Actions Note where none were previously provided. NUREG 3.8.6 Actions are preceded by a Note that states that separate entry is allowed for each battery. No Bases were provided for this Note, which is inconsistent with the format of other sections of the NUREG Bases.

50 Not used

51. NUREG Bases - ANO-1 was designed and licensed to the AEC's General Design Criteria (GDC) which was published in the Federal Register on July 11, 1967 [32FR10213]. Appendix A to 10 CFR 50 effective in 1971 [36FR3256] and subsequently amended, is somewhat different from the proposed 1967 criteria. SAR Section 1.4 includes an evaluation of ANO with respect to the 1967 criteria. The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the SAR.

52. NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

The 10 CFR 50.36 Criterion satisfied by the ITS LCOs was modified to preserve consistency with the ANO-1 license basis. The NUREG Criterion specified were modified to be consistent with the analysis assumptions regarding equipment availability and operating condition (i.e., MODE).

53. NUREG 3.8.6 – Incorporated TSTF-278.

54 NUREG Table 3.8.6-1, Footnote b requires level correction of specific gravity. This correction is not incorporated in the ANO-1 ITS Table 3.8.4-1, Footnote b.

If the electrolyte level is between the high and low level marks and the temperature corrected specific gravity is within the manufacturer's nominal specific gravity, IEEE-450-1995, Annex A, Paragraph A.3, states that it is not necessary to correct the specific gravity of the battery for electrolyte level. The vendor of the ANO-1 batteries (C&D Technologies, Inc) has stated that, in essence, the level correction factor allows the battery user to estimate what the specific gravity would be after the cell has been topped up with water. Thus if cells have been determined to be in a fully charged state, but electrolyte levels are below the high level indicator, the user can determine if the cells require water or electrolyte (of the same specific gravity as originally provided) for the topping up process, by applying the level correction to his "as found" specific gravity readings. Although the correction for specific gravity can provide useful information, C&D considers this correction to be meaningless as a means of determining battery operability. C&D's Operating Instructions do not recommend correcting electrolyte specific gravity for level. Therefore, a level correction is not applied to the Category A, B or C specific gravity measurement. This change is consistent with current license basis.

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE: 38

a. Two ~~qualified~~ circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; ~~[and]~~ 3.7.1.A 3.7.1.F 3.7.1.G

b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System~~; and~~. 3.7.1.C

~~c. Automatic load sequencers for Train A and Train B].~~ 1

APPLICABILITY: MODES 1, 2, 3, and 4. 3.7.1 3.7.2.A

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ~~[required]~~ offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE ~~[required]~~ offsite circuit.	1 hour ~~AND~~ ~~Once per 8 hours thereafter~~
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND		
			(continued)

N/A 37 3.0.5

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. (continued)	A.3 Restore ~~[required]~~ offsite circuit to OPERABLE status.	72 hours AND ~~6~~ 10 days from discovery of failure to meet LCO 2	3.7.2.B Table 3.5.1-1 Note 14 NA
B. One ~~[required]~~ DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE ~~[required]~~ offsite circuit(s).	1 hour ~~AND~~ ~~Once per 8 hours thereafter~~ 37	3.7.2.C NA
	AND B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)	3.0.5 3.7.2.C
	AND B.3.1 Determine OPERABLE DG~~(s)~~ is not inoperable due to common cause failure.	~~[~~24~~]~~ hours	NA edit
	OR B.3.2 Perform SR 3.8.1.2 for OPERABLE DG~~(s)~~.	~~[~~24~~]~~ hours	3.7.2.C edit
	AND		

(continued)

 CTS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
B. (continued)	B.4	Restore ~~[required]~~ DG to OPERABLE status.	~~72 hours~~ 7 days 2 AND ~~6~~ 10 days from discovery of failure to meet LCO	3.7.2.C NA
C. Two ~~[required]~~ offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)	3.0.5
	AND C.2	Restore one ~~[required]~~ offsite circuit to OPERABLE status.	24 hours	NA

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
D. One {required} offsite circuit inoperable. **AND** One ~~[required]~~ DG inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.~~9~~ 6, "Distribution ~~Systems—Operating,~~" when Condition D is entered with no AC power source to any train. ----------------------------		NA
	D.1 Restore {required} offsite circuit to OPERABLE status.	12 hours	NA
	OR		
	D.2 Restore ~~[required]~~ DG to OPERABLE status.	12 hours	NA
E. Two ~~[required]~~ DGs inoperable.	E.1 Restore one ~~[required]~~ DG to OPERABLE status.	2 hours	NA

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
~~F. ---REVIEWER'S NOTE--- This Condition may be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads following a loss of offsite power independent of, or coincident with, a Design Basis Event. --- One [required] [automatic load sequencer] inoperable.~~	~~F.1 Restore [required] [automatic load sequencer] to OPERABLE status.~~	~~[12] hours~~	1
F ~~G~~. Required Action and Associated Completion Time of Condition A, B, C, D, ~~[or]~~ or E~~, or F~~ not met.	F ~~G~~.1 Be in MODE 3.	12 hours	3.7.2.A & H 3.0.5
	AND		
	F ~~G~~.2 Be in MODE 5.	36 hours	3.7.2.A & H 3.0.5
G ~~H~~. Three or more ~~[required]~~ AC sources inoperable.	G ~~H~~.1 Enter LCO 3.0.3.	Immediately	3.7.2.A

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each [required] offsite circuit.	7 days	NA
SR 3.8.1.2	-----NOTES----- 1. ~~Performance of SR 3.8.1.7 satisfies this SR.~~ 3 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. ~~3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met.~~ 3 ----- Verify each DG starts from standby conditions , in ≤ 15 seconds, and achieves ~~steady state voltage ≥ [3740] V and ≤ [4580] V, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.~~ "ready-to-load" conditions. 3	~~As specified in Table 3.8.1-1~~ 4 31 days	4.6.1.1 40

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
SR 3.8.1.3	-------------------NOTES------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients ~~outside the~~ below the required load ~~range~~ do not invalidate this test. ~~3. This Surveillance shall be conducted on only one DG at a time.~~ 3 ~~4~~. This SR shall be preceded by and ~~[immediately]~~ follow, without shutdown, a successful performance of SR 3.8.1.2. ~~or SR 3.8.1.7~~ --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ ~~[4500]~~ 2475 kW ~~and ≤ [5000] kW~~.	~~As specified in Table 3.8.1-1.~~ 31 days	NA 41 NA 5 NA edit 3 4 4.6.1.1 41 6
SR 3.8.1.4	Verify each day tank ~~[and engine mounted tank]~~ contains ≥ ~~[220]~~ 160 gal~~. of fuel oil~~lons.	31 days	4.6.1.4.c 3.7.1.C edit
~~SR 3.8.1.5~~	~~Check for and remove accumulated water from each day tank [and engine mounted tank].~~	~~[31] days~~	20
SR 3.8.1.~~6~~ 5	Verify the fuel oil transfer system operates to ~~[automatically]~~ transfer fuel oil from storage tank~~[s]~~ to the day tank ~~[and engine mounted tank]~~.	~~[92]~~ 31 days	4.6.1.4.b 3.7.1.c.3 6

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.8.1.7~~	~~NOTE: All DG starts may be preceded by an engine prelube period.~~ ~~Verify each DG starts from standby condition and achieves, in ≤ [10] seconds, voltage ≥ [3740] V and ≤ [4580] V, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.~~	~~184 days~~
SR 3.8.1.~~8~~ 6	~~NOTE: This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR.~~ Verify ~~[~~automatic ~~[and manual]~~ transfer of AC power sources ~~from the normal offsite circuit to each alternate [required] offsite circuit.~~ to the selected offsite circuit and manual transfer to the alternate required offsite circuit.	~~[~~18 months~~]~~

3 7 8 39

Table 4.1-1, item 33 edit

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.9	----------NOTES---------- 1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ [0.9]. -------------------------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ [63] Hz; b. Within [3] seconds following load rejection, the voltage is ≥ [3740] V and ≤ [4580] V; and c. Within [3] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz.	[18 months]
SR 3.8.1.10	----------NOTE---------- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. -------------------------- Verify each DG operating at a power factor ≤ [0.9] does not trip, and voltage is maintained ≤ [5000] V during and following a load rejection of ≥ [4500] kW and ≤ [5000] kW.	[18 months]

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(continued)

 CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY	
SR 3.8.1.~~14~~ 7 ---NOTE~~S~~---		NA
~~1.~~ All DG starts may be preceded by an engine prelube period.		7
~~2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.~~		8
Verify on an actual or simulated loss of offsite power signal:	~~[~~18 months~~]~~	4.6.1.2.a
a. De-energization of emergency buses;		4.6.1.2 & T4.1-1 item 32
b. Load shedding from emergency buses; and		
c. DG auto-starts from standby condition and:		
1. achieves "ready-to-load" conditions		
2 ~~1~~. energizes permanently connected loads, in ≤ ~~[10]~~ 15 seconds,		1
3 ~~2~~. energizes auto-connected shutdown load through ~~[~~automatic load sequencer~~]~~ timers, and		
~~3. maintains steady-state voltage ≥ [3740] V and ≤ [4580] V,~~		23
~~4. maintains steady-state frequency ≥ [58.8] Hz and ≤ [61.2] Hz, and~~		
4 ~~5~~. supplies ~~permanently~~ connected ~~and auto-connected shutdown~~ loads for ≥ 5 minutes.		4.6.1.2.d edit

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------NOTES------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. --- Verify on an actual or simulated [Engineered Safety Feature (ESF)] actuation signal each DG auto-starts from standby condition and: a. In ≤ [12] seconds after auto-start and during tests, achieves voltage ≥ [3740] V and ≤ [4580] V; b. In ≤ [12] seconds after auto-start and during tests, achieves frequency ≥ [58.8] Hz and ≤ [61.2] Hz; c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are energized [or auto-connected through the automatic load sequencer] from the offsite power system.	[18 months]

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(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	-----NOTE----- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. --------------- Verify each DG automatic trip is bypassed on [actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESP actuation signal] except: a. Engine overspeed; [and] b. Generator differential current[; c. Low lube oil pressure; d. High crankcase pressure; and e. Start failure relay].	[18 months]

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(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	------------------------------NOTES------------------------------ 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. --- Verify each DG operating at a power factor ≤ [0.9] operates for ≥ 24 hours: a. For ≥ [2] hours loaded ≥ [5250] kW and ≤ [6000] kW; and b. For the remaining hours of the test loaded ≥ [4500] kW and ≤ [5000] kW.	[18 months]
SR 3.8.1.15	------------------------------NOTES------------------------------ 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ [2] hours loaded ≥ [4500] kW and ≤ [5000] kW. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. --- Verify each DG starts and achieves, in ≤ [10] seconds, voltage ≥ [3740] V and ≤ [4580] V, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.	[18 months]

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(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.16	------------------NOTE------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. --- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	[18 months]
[SR 3.8.1.17	------------------NOTE------------------ This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. --- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to ready-to-load operation[; and b. Automatically energizing the emergency load from offsite power].	[18 months]]

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(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.18	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. -- Verify interval between each sequenced load block is within ± [10% of design interval] for each emergency [and shutdown] load sequencer.	[18 months]

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.~~19~~ 8 -------------------NOTE~~S~~------------------- ~~1.~~ All DG starts may be preceded by an engine prelube period. ~~2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.~~ --	NA 7 8
Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions 2 ~~1~~. energizes ~~permanently~~ connected loads ~~in ≤ [10] seconds,~~ 15 3 ~~2~~. energizes auto-connected emergency loads through ~~[~~load sequencer~~s]~~ timing timers, and ~~3. achieves steady-state voltage ≥ [3740] V and ≤ [4580] V,~~ ~~4. achieves steady-state frequency ≥ [58.8] Hz and ≤ [61.2] Hz, and~~ 4 ~~5~~. supplies ~~permanently~~ connected ~~and auto-connected emergency~~ loads for ≥ ~~[5]~~ 60 minutes.	~~[~~18 months~~]~~ 4.6.1.2.b 4.6.1.2 & T4.1-1 item 32 edit 1 23 4.6.1.2.d edit

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	---NOTE--- All DG starts may be preceded by an engine prelube period. --- Verify, when started simultaneously from standby condition, each DG achieves, in ≤ [10] seconds, voltage ≥ [3740] V and ≤ [4580] V, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.	10 years

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Table 3.8.1-1 (page 1 of 1)
Diesel Generator Test Schedule

NUMBER OF FAILURES IN LAST 25 VALID TESTS[a]	FREQUENCY
≤ 3	31 days
≥ 4	7 days[b] (but no less than 24 hours)

(a) Criteria for determining number of failures and valid tests shall be in accordance with Regulatory Position C.2.1 of Regulatory Guide 1.9, Revision 3, where the number of tests and failures is determined on a per DG basis.

(b) This test frequency shall be maintained until seven consecutive failure free starts from standby conditions and load and run tests have been performed. This is consistent with Regulatory Position [ ], of Regulatory Guide 1.9, Revision 3. If, subsequent to the 7 failure free tests, 1 or more additional failures occur such that there are again 4 or more failures in the last 25 tests, the testing interval shall again be reduced as noted above and maintained until 7 consecutive failure free tests have been performed.

[Note: If Revision 3 of Regulatory Guide 1.9 is not approved, the above table will be modified to be consistent with the existing version of Regulatory Guide 1.108, GL 84-15, or other approved guidance.]

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3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

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

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

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

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

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	---NOTE--- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A.	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	
	A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required DG inoperable.	B.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	B.2	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	B.3	Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND		
	B.4	Initiate action to restore required DG to OPERABLE status.	Immediately

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.2.1	-------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.13 through SR 3.8.1.16, [SR 3.8.1.18,] and SR 3.8.1.19. -- For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources—Operating," except SR 3.8.1.8, SR 3.8.1.17, and SR 3.8.1.20, are applicable.	In accordance with applicable SRs

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CTS

3.7.1.C

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 2 Diesel Fuel Oil~~, Lube Oil,~~ and Starting Air

LCO 3.8.3 2 The stored diesel fuel oil~~, lube oil,~~ and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE. 3.7.1.A & N/A

ACTIONS

NA

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

edit

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~DGs~~ fuel oil storage tank(s) with fuel ~~level~~ volume < ~~[33,000] gal~~ 20,000 gallons and > ~~[28,285] gal~~ 17,140 gallons ~~in the storage tank.~~	A.1 Restore fuel oil ~~level~~ volume to within limits.	48 hours
~~B. One or more DGs with lube oil inventory < [500] gal and > [425] gal.~~	~~B.1 Restore lube oil inventory to within limits.~~	~~48 hours~~
B ~~C~~. One or more DGs with stored fuel oil total particulates not within limit.	B ~~C~~.1 Restore fuel oil total particulates to within limits.	7 days

NA

edit

25

NA

(continued)

25
CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
C ~~D~~. One or more DGs with new fuel oil properties not within limits.	C ~~D~~.1 Restore stored fuel oil properties to within limits.	30 days	NA
D ~~E~~. One or more required DGs with starting air receiver pressure < ~~[225]~~ 175 psig and ≥ ~~[125]~~ 158 psig.	D ~~E~~.1 Restore required starting air receiver pressure to ~~≥ [225]~~ psig. within limits	48 hours	42 NA edit
E ~~F~~. Required Action and associated Completion Time not met. OR One or more required DGs with diesel fuel oil ~~Lube oil,~~ or starting air subsystem not within limits for reasons other than Condition A, B, C, ~~D,~~ or ~~E~~ D.	E ~~F~~.1 Declare associated DG inoperable.	Immediately	NA 25 42

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.8.~~3~~ 2.1	Verify each fuel oil storage tank contains ≥ ~~[33,000]~~ 20,000 gallons of fuel.	31 days	4.6.1.4.d 3.7.1.c edit

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
~~SR 3.8.3.2~~	~~Verify lube oil inventory is ≥ [500] gal.~~	~~31 days~~	25
SR 3.8.~~3~~.~~3~~ 2 2	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program	4.6.1.4.e NA
SR 3.8.~~3~~.~~4~~ 2 3	Verify each DG required air start receiver pressure is ≥ ~~[225]~~ 175 psig.	31 days	42 4.6.1.4.a 4.6.1.5
~~SR 3.8.3.5~~	~~Check for and remove accumulated water from each fuel oil storage tank.~~	~~[31] days~~	20
~~SR 3.8.3.6~~	~~For each fuel oil storage tank: a. Drain the fuel oil; b. Remove the sediment; and c. Clean the tank.~~	~~10 years~~	

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.~~4~~ 3 DC Sources—Operating

LCO 3.8.~~4~~ 3 Both ~~The Train A and Train B~~ DC electrical power subsystems shall be OPERABLE. 3.7.3 edit

APPLICABILITY: MODES 1, 2, 3, and 4. 3.7.3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	~~2~~ 8 hours (13) 3.7.3.A.3
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	~~6~~ 12 hours (36) 3.7.3.B 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.~~4~~ 3.1 Verify battery terminal voltage is ≥ ~~[129/258]~~ 124.7 V on float charge.	7 days 4.6.2.1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.2	~~Verify no visible corrosion at battery terminals and connectors.~~ ~~OR~~ ~~Verify battery connection resistance [is ≤ [1E-5 ohm] for inter-cell connections, ≤ [1E-5 ohm] for inter-rack connections, ≤ [1E-5 ohm] for inter-tier connections, and ≤ [1E-5 ohm] for terminal connections].~~	~~92 days~~ 14
SR 3.8.4.3	~~Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration.~~	~~[12] months~~ 14
SR 3.8.4.4	~~Remove visible terminal corrosion and verify battery cell to cell and terminal connections are [clean and tight, and are] coated with anti-corrosion material.~~	~~[12] months~~ 14
SR 3.8.4.5	~~Verify battery connection resistance [is ≤ [1E-5 ohm] for inter-cell connections, ≤ [1E-5 ohm] for inter-rack connections, ≤ [1E-5 ohm] for inter-tier connections, and ≤ [1E-5 ohm] for terminal connections].~~	~~[12] months~~ 14

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.8.4.6~~	~~-----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.~~ ~~Verify each battery charger supplies ≥ [400] amps at ≥ [125/250] V for ≥ [8] hours.~~	~~[18 months]~~ 48
SR 3.8.~~4~~ 3 .~~7~~ 2	~~-----NOTES----- 1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 once per 60 months.~~ 19 ~~2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.~~ 7 8 Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test. 19	~~[18 months]~~ 6 6 4.6.2.2

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.8.~~8~~.~~8~~ 3 3	~~-----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.~~ Verify battery capacity is ≥ ~~[~~80~~]~~% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	7 8 60 months AND 12 months when battery shows degradation, or has reached ~~[~~85~~]~~% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached ~~[~~85~~]~~% of the expected life with capacity ≥ 100% of manufacturer's rating	4.6.2.3

17

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	

(continued)

17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-----NOTE----- The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. ----- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7, SR 3.8.4.8.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.~~6~~ 4 Battery Cell Parameters 3.7.4

LCO 3.8.~~6~~ 4 Battery cell parameters ~~for the Train A and Train B batteries~~ shall be within ~~the~~ limits ~~of Table 3.8.6-1~~. 22 53

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE. 3.7.4

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One or more batteries with one or more battery cell parameters not within Category A or B limits. Table 3.8.4-1	A.1	Verify pilot cell~~s~~ electrolyte level and float voltage meet Table 3.8.~~6~~ 4-1 Category C ~~values~~ limits.	1 hour	3.7.4.A.1 edit 53
	AND			
	A.2	Verify battery cell parameters meet Table 3.8.~~6~~ 4-1 Category C ~~values~~ limits.	24 hours AND Once per 7 days thereafter	3.7.4.A.2 edit
	AND			
	A.3	Restore battery cell parameters to Category A and B limits ~~of~~ ~~Table 3.8.6-1~~. 4	31 days	3.7.4.A.3 53

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with pilot cell or average electrolyte temperature of the representative cells < ~~[60]~~ 60 °F. OR One or more batteries with one or more battery cell parameters not within Category C values. Table 3.8.4-1	B.1 Declare associated battery inoperable.	Immediately	3.7.4.B 30 3.7.4.C 3.7.4.C 53

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.8.~~6~~ 4.1	Verify battery cell parameters meet Table 3.8.~~6~~ 4-1 Category A limits.	7 days	4.6.2.5
SR 3.8.4.2	Verify electrolyte temperature of the pilot cell is ≥ 60 °F.	31 days	4.6.2.8 30

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY	
SR 3.8.~~6~~.~~2~~ 4 3 Verify battery cell parameters meet Table 3.8.~~6~~-1 4 Category B limits.	92 days AND Once within 24 hours after a battery discharge < [110] V AND Once within 24 hours after a battery overcharge > ~~[150]~~ 145 V	4.6.2.7
SR 3.8.~~6~~.~~3~~ 4 4 Verify average electrolyte temperature of representative cells is ≥ ~~[60]~~ 60 °F.	92 days	4.6.2.6

4

Table 3.8.~~6~~-1 (page 1 of 1)
Battery Cell Surveillance Requirements

CTS

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL	
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	Above top of plates, and not overflowing	Table 4.6-1
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V	Table 4.6-1
Specific Gravity[b][c]	≥ ~~[1.200]~~ 1.195	≥ ~~[1.195]~~ 1.190 AND Average of all connected cells > ~~[1.205]~~ 1.195	Not more than 0.020 below average connected cells AND Average of all connected cells ≥ ~~[1.195]~~ 1.190	Table 4.6-1

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

(b) Corrected for electrolyte temperature ~~and level. Level correction is not required, however, when battery charging is < [2] amps when on float charge.~~ 54

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

CTS

NA

3.8 ELECTRICAL POWER SYSTEMS

3.8 [7] 5 Inverters—Operating

LCO 3.8 [7] 5 The required Red Train [A] and Green Train [B] inverters shall be OPERABLE.

---NOTE---

[One/two] inverter[s] may be disconnected from ~~[its/their]~~ associated DC bus for ≤ 2[4] hours to perform ~~an equalizing charge on [its/their] associated [common] battery,~~ load transfer to or from the swing inverter, provided: 45

a. The associated 120 V AC vital bus[es] [is/are] energized from ~~[its/their] [Class 1E constant voltage source]~~ alternate ~~transformers] [inverter using internal AC source]~~; and

b. All other 120 V AC vital buses are energized from their associated OPERABLE inverters.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~[required]~~ inverter inoperable.	A.1 ---NOTE--- Enter applicable Conditions and Required Actions of LCO 3.8 [9] 6 "Distribution Systems - Operating" with any 120 VAC vital bus de-energized. --- Restore inverter to OPERABLE status.	24 hours

edit

(continued)

CTS
N/A

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	~~6~~ 12 hours 36 hours

36

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.~~7~~ 5.1 Verify correct inverter voltage~~,~~ and ~~[frequency,]~~ and alignment to required ~~AC~~ 120 VAC vital buses.	31 ~~7~~ days

46
edit

17

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters—Shutdown

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more [required] inverters inoperable.	A.1 Declare affected required feature(s) inoperable. OR A.2.1 Suspend CORE ALTERATIONS. AND A.2.2 Suspend movement of irradiated fuel assemblies. AND A.2.3 Initiate action to suspend operations involving positive reactivity additions. AND	Immediately Immediately Immediately Immediately
		(continued)

17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.8.1 Verify correct inverter voltage, [frequency,] and alignments to required AC vital buses.	7 days

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 6 Distribution Systems—Operating

edit 3.7.1.B

LCO 3.8.9 Two ~~Train A and Train B~~ AC, DC, and 120 VAC ~~AC~~ vital bus electrical power distribution subsystems shall be OPERABLE.

3.7.1
3.7.2.A

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One or more AC electrical power distribution subsystem(s) inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours **AND** 16 hours from discovery of failure to meet LCO	47 3.7.2.D NA
B. One or more 120 VAC ~~AC~~ vital bus electrical power distribution subsystems inoperable.	B.1 Restore 120VAC ~~AC~~ vital bus electrical power distribution subsystem(s) to OPERABLE status.	8 ~~2~~ hours **AND** 16 hours from discovery of failure to meet LCO	13 edit 3.7.2.D 35 NA 47
C. One or more DC electrical power distribution subsystem(s) inoperable.	C.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 ~~2~~ hours **AND** 16 hours from discovery of failure to meet LCO	47 3.7.2.F 13 NA

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	~~6~~ 12 hours	36 3.7.2.A 3.7.2.D 3.7.2.F
	AND			
	D.2	Be in MODE 5.	36 hours	3.7.2.A 3.7.2.D 3.7.2.F
E. Two or more electrical power inoperable distribution subsystems that result in a loss of function.	E.1	Enter LCO 3.0.3	Immediately	3.7.2.A 47

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.8.~~9~~.1 6	Verify correct breaker alignments ~~and voltage~~ to ~~required~~ AC, DC, and ~~AC~~ 120 VAC vital bus electrical power distribution subsystems.	31 ~~7~~ days	44 NA edit 43

17

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems—Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and AC vital bus ~~electrical~~ power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	

(continued)

17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status. **AND** A.2.5 Declare associated required decay heat removal subsystem(s) inoperable and not in operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

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

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources—Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate~~(s)~~) and the onsite standby power sources (~~[Train A and Train B]~~ emergency diesel generators (DGs)). As required by ~~[10 CFR 50, Appendix A]~~, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered ~~Safety Feature~~ Safeguards (ES~~F~~) systems.

SAR, Section 1.4

edit

51

edit

INSERT B 3.8-1A

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

27

Offsite power is supplied ~~to~~ five the unit switchyard~~(s)~~ from the transmission network by ~~[two]~~ transmission lines. From the switchyard~~(s)~~, two electrically and physically separated offsite circuits provide AC power, through ~~[step down station]~~ either the startup transformers or the unit auxiliary transformer~~(s)~~, to the 4.16 kV ES~~F~~ buses. A detailed description of the offsite power network and the circuits to the Class 1E ES~~F~~ buses is found in the ~~F~~SAR, Chapter ~~[8]~~ (Ref. 2).

27

~~An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).~~

27

With an engineered safeguards actuation system (ESAS) signal present, ~~C~~certain required unit loads are ~~returned to~~ placed in service in a predetermined sequence ~~in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System~~. Within ~~[~~1 minute~~]~~ after the initiating ~~signal is received~~, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are ~~returned to~~ in service by the load sequencing timers ~~via the load sequencer~~.

1

The onsite standby power source for each 4.16 kV ES~~F~~ bus is a dedicated DG. DGs ~~[1]~~ and ~~[2]~~ are dedicated to ES~~F~~ buses ~~[11]~~ A3 and ~~[12]~~ A4, respectively. A DG starts

(continued)

<INSERT B3.8-1A>

During typical on-line operation, power for unit equipment is provided from the unit auxiliary transformer. When the unit is off-line, unit equipment is typically powered from a startup transformer or from the unit auxiliary transformer back fed from the 500 kV switchyard. A unit trip (i.e., generator lockout) initiates an automatic transfer to an offsite power circuit (i.e., typically startup transformer No. 1). Startup transformer No. 2 is normally not selected for automatic transfer since it is the backup for both Unit 1 and Unit 2. In the event of a loss of offsite power to the startup transformer, an undervoltage condition trips its associated bus feeder breakers. When the startup transformer bus feeder breakers open, the bus feeder breakers for the alternate startup transformer automatically close (if available) provided the generator lockout relays have not been reset. If the power source is transferred to startup transformer No. 2, sufficient loads are automatically shed to avoid a degraded voltage condition (since startup transformer No. 2 is not sufficient to simultaneously provide power for full loading from both units.)

<INSERT B3.8-2A below>

However, the "intended service" rating provided by the manufacturer is 2750 kW. This is the value used in postulated DG loading evaluations (Ref. 2).

BASES

BACKGROUND (continued)

an applicable Engineered Safeguards Actuation System (ESAS)

automatically on ~~a Reactor Coolant System (RCS) pressure~~ signal or on an ESØ bus degraded voltage or undervoltage signal ~~(refer to~~ see LCO 3.3.5, "Engineered ~~Safety Feature~~ Safeguards Actuation System (ESFAS) Instrumentation" and LCO 3.3.8, "~~Emergency~~ Diesel Generator (EDG) Loss of Power Starts (LOPS)"). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESØ bus undervoltage or degraded voltage, independent of or coincident with an ~~Safety Injection (SI)~~ ESAS signal. The DGs will also start and operate in the standby mode without tying to the ESØ bus on an ~~SI~~ signal alone. Following the trip of offsite power, ~~a sequencer~~ an undervoltage signal strips nonpermanent loads from the ESØ bus. When the DG is tied to the ESØ bus, loads are then sequentially connected to ~~its~~ their respective ESØ bus by the automatic load sequencing timers. The sequencing ~~logic~~ timers control the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

26

edit

26

1

In the event of a loss of preferred power, the ESØ electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA). concurrent

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within ~~[1]~~ 1 minute after the initiating signal is received, by the load sequencing timers, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

1

Ratings for ~~Train A and Train B~~ emergency DGs satisfy the ~~requirements~~ guidance of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is ~~[7000]~~ 2600 kW with ~~[10]~~ 10% overload permissible for up to 2 hours in any 24 hour period. INSERT B3.8-2A from above The ESØ loads that are powered from the 4.16 kV ESØ buses are listed in Reference 2. (Ref. 2)

edit

27

INSERT B3.8-2B from B3.8-15

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the ~~F~~SAR, ~~Chapter [6] (Ref. 4) and~~ Chapter ~~[15]~~ 14 (Ref. 5), assume ESØ systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESØ systems so that the fuel, ~~RCS~~ Reactor Coolant System (RCS), and

edit

edit

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

reactor building ~~Containment~~ design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "~~Containment~~ Reactor Building Systems."

edit

edit

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. ~~This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during accident conditions in the event of:~~

33

~~a. An assumed loss of all offsite power or all onsite AC power; and~~

~~b. A worst-case single failure.~~

In MODES 1 and 2, ~~T~~the AC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 6). ~~NRC Policy Statement.~~

In MODES 3 and 4, the AC sources satisfy Criterion 4 of 10 CFR 50.36.

52

38

LCO

Two ~~qualified~~ circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent (emergency DGs 1 and 2) DGs for each ES train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an ~~anticipated operational occurrence (AOO)~~ abnormality or a postulated DBA.

edit

26

~~Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.~~

27

~~In addition, one required automatic load sequencer per train must be OPERABLE.~~

1

Each required offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ES~~F~~ buses.

INSERT B3.8-3A

~~Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. Offsite circuit #2 consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF~~

27

(continued)

<INSERT B3.8-3A>

One offsite circuit consists of startup transformer No. 1, its supply from the switchyard bus tie autotransformer, either the 4160 V bus A1 or A2, and the feeder breaker providing power to either 4160 V ES bus A3 or A4. An alternative for this offsite circuit consists of the unit auxiliary transformer, its supply from the switchyard bus tie autotransformer and the overhead swing leads, either the 4160 V bus A1 or A2, and the feeder breaker providing power to either 4160 V ES bus A3 or A4. A second offsite circuit consists of startup transformer No. 2, its supply from the 161 kV switchyard ring bus, either the 4160 V bus A1 or A2, and the feeder breaker providing power to either 4160 V ES bus A3 or A4. If capable of automatic transfer startup transformer No. 2 must have OPERABLE selective load-shed features for automatic shedding of loads to avoid a degraded voltage condition to be considered OPERABLE. An offsite circuit includes the necessary breakers and equipment to properly align the circuit and transmit power from the transmission line sources to a single 4160 V ES bus. If power is not being supplied to bus A3 or A4 from bus A1 or A2 respectively, one of the offsite circuits must be considered inoperable.

For the offsite AC sources, separation and independence are to the extent practical. An offsite circuit may be connected to more than one ES bus and not violate the separation criteria provided each OPERABLE required offsite circuit is capable of being aligned (manually or automatically) so that it is separate and independent of the other required offsite circuit. Only two of the possible offsite circuits are required to be OPERABLE.

When the main generator is synchronized to the 500 kV system, AC power for the ES loads may be taken from either the unit auxiliary transformer, either startup transformer, or a combination of these transformers concurrently sharing the load. Power from the unit auxiliary transformer is not credited with meeting the requirements of LCO 3.8.1.a since it cannot function under all conditions (i.e., following a turbine trip) except when connected in the alternate configuration described above. However, powering the ES buses from the unit auxiliary transformer is permitted.

BASES

LCO (continued)

~~transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.~~ 27

Each DG (DG1 and DG2) must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ES~~F~~ bus on detection of bus undervoltage. This will be accomplished within [10] 15 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ES~~F~~ buses. ~~These capabilities are required to be met from a variety of initial conditions, such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions.~~ 15 ~~Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.~~ 21

Proper sequencing of loads, (including tripping of non-essential loads,) is a required function for DG OPERABILITY.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

~~For the offsite AC sources, separation and independence are to the extent practical. [A circuit may be connected to more than one ESF bus, with fast transfer capability to the other circuit OPERABLE, and not violate separation criteria.~~ 27 ~~A circuit that is not connected to an ESF bus is required to have OPERABLE fast transfer interlock mechanisms to at least two ESF buses to support OPERABILITY of that circuit.]~~ 27

This LCO does not apply to the Alternate AC DG nor to the security DG.

APPLICABILITY

The AC sources ~~[and sequencers]~~ are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that: 1

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

(continued)

BASES

APPLICABILITY (continued)

b. Adequate core cooling is provided and ~~containment~~ reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA. edit

The AC power requirements for MODES 5 and 6 are ~~covered in LCO 3.8.2, "AC Sources—Shutdown."~~ addressed by the definition of OPERABILITY for each required supported load. 17

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit ~~on a more frequent basis~~. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered. 37 edit 27

<INSERT B3.8-5A>

~~Reviewer's Note: The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~ 27

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train. ~~This includes motor driven emergency feedwater pumps. Single train systems, such as turbine driven emergency feedwater pumps, may not be included.~~ 27

(continued)

<INSERT B3.8-5A>

The Completion Time provides for a prompt confirmation of the OPERABILITY of the remaining offsite circuit. This is considered to be acceptable because of other indications which are available in the control room for loss of the remaining offsite circuit.

BASES

ACTIONS

A.2 (continued)

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power available to ~~supplying~~ its loads; and 27

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to ~~Train A and Train B~~ both trains of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. edit

A.3

~~According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours.~~ 27 With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the

(continued)

BASES

ACTIONS

A.3 (continued)

potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 7 days ~~72 hours~~. This could lead to a total of ~~144 hours~~ 10 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional ~~72 hours~~ 7 days (for a total of ~~8~~ 17 days) allowed prior to complete restoration of the LCO. The ~~8~~ 10 day Completion Time provides a limit on the time allowed in a 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 the 72 hour and ~~8~~ 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met. 2

As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of

(continued)

BASES

ACTIONS

B.1 (continued)

the offsite circuits ~~on a more frequent basis~~. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

37

27

INSERT B3.8-8A

~~Reviewer's Note: The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~

27

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. ~~This includes motor driven emergency feedwater pumps. Single train systems, such as turbine driven emergency feedwater pumps, are not included.~~ Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG.

27

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train is inoperable.

(continued)

<INSERT B3.8-8A>

The Completion Time provides for a prompt confirmation of the OPERABILITY of the remaining offsite circuit. This is considered to be acceptable because of other indications, which are available in the control room for monitoring the status of the remaining offsite circuit.

BASES

ACTIONS

B.2 (continued)

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single-failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid
unnecessary testing of OPERABLE DG(s). If it can be
determined that the cause of the inoperable DG does not
exist on the OPERABLE DG, SR 3.8.1.2 does not have to be the edit
performed. If the cause of inoperability exists on other
DG~~(s)~~, the other DG~~(s)~~ would be declared inoperable upon
discovery and Condition E of LCO 3.8.1 would be entered.
Once the failure is repaired, the common cause failure no
longer exists and Required Action B.3.1 is satisfied. If
the cause of the initial inoperable DG cannot be confirmed
not to exist on the remaining DG~~(s)~~, performance of
SR 3.8.1.2 suffices to provide assurance of continued
OPERABILITY of that DG.

(continued)

BASES

ACTIONS

B.3.1 and B.3.2 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the ~~[plant corrective action]~~ program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

condition reporting

26

According to Generic Letter 84-15 (Ref. 7), [24] hours is reasonable to confirm that the OPERABLE DG(s) is not affected by the same problem as the inoperable DG.

B.4

~~According to Regulatory Guide 1.93 (Ref. 6),~~ operation may continue in Condition B for a period that should not exceed ~~72 hours.~~ 7 days.

2

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The ~~72 hour~~ 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of ~~144 hours,~~ 10 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of ~~9~~ days) allowed prior to complete restoration of the LCO. The ~~6~~ day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Condition A and Condition B are entered concurrently. The "AND" connector between the ~~72 hour~~ 7 day and ~~6~~ 10 day Completion Times

13

10

2

(continued)

BASES

ACTIONS

B.4 (continued)

means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that ~~Regulatory Guide 1.93 (Ref. 6) allows~~ a Completion Time of 24 hours is allowed for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. ~~This includes motor driven auxiliary feedwater pumps. Single train features, such as turbine driven auxiliary pumps, are not included in the list.~~ 27 27

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

If at any time during the existence of Condition C (two offsite circuits inoperable) and a required feature becomes inoperable, this Completion Time begins to be tracked. edit

~~According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours.~~ 27 This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria,

~~According to Reference 6,~~ With the available offsite AC sources, two less than required by the LCO, operation may 27

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation would continue in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, (one or more trains) the Conditions and Required Actions for LCO 3.8.~~9~~ 6 "Distribution Systems—Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.~~9~~ 6 provides the appropriate restrictions for a de-energized train. edit edit

~~According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours.~~ 27

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. ~~Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure.~~ 27 The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

E.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

~~According to Reference 6,~~ With both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

~~F.1~~

~~The sequencer(s) is an essential support system to [both the offsite circuit and the DG associated with a given ESF bus]. [Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus.] Therefore, loss of an [ESF bus sequencer] affects every major ESF system in the [division]. The [12] hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.~~

~~This Condition is preceded by a Note that allows the Condition to be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads under any conditions. Implicit in this Note is the concept that the Condition must be retained if any sequencer failure mode results in the inability to start all or part of the safety~~

1

(continued)

BASES

ACTIONS

~~F.1 (continued)~~

~~loads when required, regardless of power availability, or results in overloading the offsite power circuit to a safety bus during an event thereby causing its failure. Also implicit in the Note is that the Condition is not applicable to any train that does not have a sequencer.~~ 1

~~G~~F.1 and ~~G~~F.2

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to 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 plant systems. 12 edit

~~H~~G.1

Condition ~~H~~ C corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

Move to INSERT B3.8-2B

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with ~~10 CFR 50, Appendix A,~~ SAR Section 1.4 GDC 18 (Ref. 4). Periodic component tests are supplemented by extensive functional tests during ~~refueling~~ outages (under simulated accident conditions) and by maintenance inspections (per manufacturer's recommendations). 51

~~The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the FSAR.~~ 27

(continued)

"ready to load," a

is applicable

3750

BASES

SURVEILLANCE REQUIREMENTS (continued)

Where the SRs discussed ~~herein~~ specify ~~voltage and frequency tolerances, the following is applicable.~~ ~~The~~ minimum ~~steady state~~ output voltage of ~~[3740]~~ V ~~is~~ 90% of the nominal 4160 V output voltage. This value~~, which is specified in ANSI C84.1 (Ref. 11),~~ allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. ~~The specified maximum steady state output voltage of [4756] V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).~~ 40

INSERT B 3.8-16A

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 ~~and SR 3.8.1.7~~ 3

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do ~~not~~ get lubricated when the engine is not running, ~~these~~ this SR~~s~~ ~~are~~ is modified by a Note ~~(Note 2 for SR 3.8.1.2)~~ to indicate that ~~all~~ DG starts for ~~these~~ this Surveillance~~s~~ may be preceded by an edit

(continued)

<INSERT B3.8-16A>

The required minimum frequency for loading of the DG is 58.8 Hz (derived from Safety Guide 9); however, this value is not routinely monitored to be within limit within 15 seconds. Meeting minimum frequency is expected prior to the DG voltage reaching the required minimum. This is administratively confirmed on an 18 month interval.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 ~~and SR 3.8.1.7~~ (continued) 3

edit

engine prelube period and followed by a warmup period prior to loading ~~by an engine prelube period~~. with application of the Note

For the purposes of SR 3.8.1.2 ~~and SR 3.8.1.7~~ testing, the DGs are started from standby conditions. Standby conditions for a DG means that the diesel engine ~~coolant and~~ oil ~~are~~ is being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. 3 27

INSERT B3.8-17A

~~In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. This is the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.~~ 3

SR 3.8.1.~~7~~ 2 requires that, ~~at a 184 day Frequency,~~ the DG starts from standby conditions 15 "ready-to-load" conditions (i.e. minimum voltage) and achieves ~~required~~ voltage ~~and frequency~~ within ~~10~~ 15 seconds. The ~~10~~ 15 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter ~~15~~ 14 (Ref. 5). 3 40

~~The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, the 10 second start requirement of SR 3.8.1.7 applies.~~ 3

~~Since SR 3.8.1.7 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.~~

The ~~normal~~ 31 day Frequency for SR 3.8.1.2 ~~(see Table 3.8.1-1, "Diesel Generator Test Schedule," in the accompanying LCO)~~ is consistent with Regulatory Guide 1.9 (Ref. 3). ~~The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies~~ provide, 5 adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing. 4 3

(continued)

<INSERT B3.8-17A>

The signal initiating the start of the DG is varied from one test to another (start with handswitch at control room panel and at DG local control panel) to verify all starting circuits are OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting full rated loads ~~greater than or equal to the equivalent of the maximum expected accident loads~~. A minimum run time of 60 minutes ~~is required to~~ stabilizes engine temperatures, while minimizing the time that the DG is connected to the offsite source. ensures

INSERT B3.8-18A

41 edit

~~Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between [0.8 lagging] and [1.0]. The [0.8] value is the design rating of the machine, while the [1.0] is an operational limitation [to ensure circulating currents are minimized].~~ 27

~~The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.~~ 41

The ~~[normal]~~ 31 day Frequency for this Surveillance ~~(Table 3.8.1-1) is consistent with Regulatory Guide 1.9 (Ref. 3).~~ provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing. 4 27

This SR is modified by ~~four~~ three Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients (e.g., because of changing bus loads) do not invalidate this test. ~~Similarly, momentary power factor transients above the limit will not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4~~ Note 3 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance. 5 edit 27 5

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the engine mounted day tank ~~[and engine mounted tank]~~ is ~~at or above the level at which fuel oil is automatically added~~ being properly maintained. The level is expressed as an equivalent volume in gallons, and is 6 27

(continued)

<INSERT B3.8-18A>

As discussed in Regulatory Guide 1.9 (Ref. 3), Section C.2.2, the load test is conducted at ≥ 90% of rated load, which is considered to be ≥ 90% of the intended service rating of 2750 kW, or 2475 kW. This parameter value (2475 kW) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.4 (continued)

selected to ensure adequate fuel oil for a minimum of ~~1 hour~~ 40 minutes of DG operation at full load plus 10%. 27

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

~~SR 3.8.1.5~~ 20

~~Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day [and engine mounted] tanks once every [31] days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 10). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.~~

SR 3.8.1.~~6~~ 5

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, and the fuel delivery piping is not obstructed~~, and the controls and control systems for automatic fuel transfer systems are OPERABLE~~. 27

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued) 5

The Frequency for this SR is variable, depending on individual system design, with up to a [92] day interval. The [92] day Frequency corresponds to the testing requirements for pumps as contained in the ASME Code, Section XI (Ref. 12); however, the design of fuel transfer systems is such that pumps will operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs. specified to correspond to the interval for DG testing. 27 the Therefore, monthly 27 3

SR 3.8.1.9

See SR 3.8.1.8.

SR 3.8.1.10 6

Transfer of each [4.16 kV ESF bus] power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The [18 month] Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 27 26

INSERT B3.8-20A

(i.e., during refueling shutdown)

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR. 7 8

(continued)

<INSERT B3.8-20A>

This Surveillance includes testing of the offsite power undervoltage and protective relaying interlocks associated with the required startup transformer power sources, and for startup transformer no. 2, this test also demonstrates the selective load shedding interlock function. (Note: This load shedding function is only required when startup transformer no. 2 is selected for automatic transfer.) These features provide protection of required equipment from a sustained degraded grid voltage situation.

The

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. For the CR-3 emergency DGs, the largest single load is 616 kW (HPI pump). After performance of SR 3.8.1.17, the diesel load is reduced to approximately 1200 kW and allowed to run at this load for 3 to 5 minutes. The load is then reduced to ≥ 616 kW and the DGs output breaker is opened. Verification that the DG did not trip is made. This Surveillance may be accomplished by: 9

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by IEEE-308 (Ref. 13), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The [3] seconds specified is equal to 60% of a typical 5 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover to following load rejection. The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR. In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, Note 2 requires that, if synchronized to offsite power, testing must be performed using a power factor ≤ [0.9]. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

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Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable;

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG will not trip upon loss of the load. These

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

acceptance criteria provide DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ [0.9]. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9) and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems.

Credit may be taken for unplanned events that satisfy this SR.

Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable;

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.~~17~~ 7

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~~As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1),~~ this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the non-essential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage ~~and frequency~~ within the specified time.

"ready to load" condition (i.e., minimum

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The DG auto-start time of ~~[10]~~ 15 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads, e.g. the running service water pump(s), is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue hardship or potential for undesired operation. ~~For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or decay heat removal (DHR) systems performing a DHR function are not desired to be realigned to the ECCS mode of operation.~~ In lieu of actual demonstration of connection and loading of loads during this test, testing that adequately shows the capability of the separate DG system to perform these functions is acceptable. This associated testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

edit

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The Frequency of [18 months] ~~is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1),~~ takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~11~~ 7 (continued)

This SR is modified by ~~two~~ a Note~~s~~. The reason for the Note ~~1~~ is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine ~~coolant and~~ oil continuously circulated and temperature maintained consistent with manufacturer recommendations. ~~The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.~~ ~~Credit may be taken for unplanned events that satisfy this SR.~~ edit 27 7 8

~~SR 3.8.1.12~~

~~This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time ([10] seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.~~

~~The requirement to verify the connection of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or DHR systems performing a DHR function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.~~ 10

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR.

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SR 3.8.1.13

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal, and critical protective functions (engine overspeed, generator differential current[, low lube oil pressure, high crankcase pressure, and start failure relay]) trip the DG to avert substantial damage to the DG unit. The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The [18 month] Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. Credit may be taken for unplanned events that satisfy this SR.

Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable;

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

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SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ [2] hours of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor of ≤ [0.9]. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections, in accordance with vendor recommendations, in order to maintain DG OPERABILITY.

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 7), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR.

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SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within [10 seconds]. The [10 second] time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.15 (continued)

inspections, in accordance with vendor recommendations, in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least [2] hours at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

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SR 3.8.1.16

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As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive and auto-close signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.17

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Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a LOCA actuation signal is received during operation in the test

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

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SR 3.8.1.18

Under accident [and loss of offsite power] conditions loads are sequentially connected to the bus by the [automatic load sequencer]. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The [10]% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.18 (continued)

~~The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~ 1

~~This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.~~ 7 8

~~Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable;~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~

~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~ 7

SR 3.8.1.[19] 8

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and ~~containment~~ reactor building design limits are not exceeded. edit

7

This Surveillance demonstrates the DG operation, as ~~discussed in the Bases for SR 3.8.1.[11]~~, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads during this test, separate testing that adequately shows the capability of the DG system to perform these 27

INSERT B3.8-31A

(continued)

<INSERT B3.8-31A>

This test is typically conducted by simulating an ESAS signal and either simultaneously or subsequently simulating a LOOP. In certain circumstances, many loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, DHR systems performing a DHR function are not desired to be interrupted from this mode of operation.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~19~~ 8 (continued)

functions is acceptable. This associated testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified. ⊢ 27

The Frequency of ~~[18 months]~~ takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of ~~[18 months]~~.

This SR is modified by ~~two~~ a Note~~s~~. The reason for the Note ~~1~~ is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs ~~must be~~ are started from standby conditions, that is, with the engine ~~coolant and~~ oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. ~~The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.~~ edit ⊢ 27 7 8

with application of the Note

~~SR 3.8.1.20~~

~~This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.~~

~~The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9).~~

~~This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated, and temperature maintained consistent with manufacturer recommendations.~~ 10

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

4

Diesel Generator Test Schedule

The DG test schedule (Table 3.8.1-1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 3). The purpose of this test schedule is to provide timely test data to establish a confidence level associated with the goal to maintain DG reliability above 0.95 per demand.

According to Regulatory Guide 1.9, Revision 3 (Ref. 3), each DG should be tested at least once every 31 days. Whenever a DG has experienced 4 or more valid failures in the last 25 valid tests, the maximum time between tests is reduced to 7 days. Four failures in 25 valid tests is a failure rate of 0.16, or the threshold of acceptable DG performance, and hence may be an early indication of the degradation of DG reliability. When considered in the light of a long history of tests, however, 4 failures in the last 25 valid tests may only be a statistically probable distribution of random events. Increasing the test Frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the DG. The increased test Frequency must be maintained until seven consecutive, failure free tests have been performed.

The Frequency for accelerated testing is 7 days, but no less than 24 hours. Tests conducted at intervals of less than 24 hours may be credited for compliance with Required Actions. However, for the purpose of re-establishing the normal 31-day Frequency, a successful test at an interval of less than 24 hours should be considered an invalid test and not count towards the seven consecutive failure free starts, and the consecutive test count is not reset.

A test interval in excess of 7 days (or 31 days, as appropriate) constitutes a failure to meet the SRs and results in the associated DG being declared inoperable. It does not, however, constitute a valid test or failure of the DG, and any consecutive test count is not reset.

REFERENCES

1. ~~10 CFR 50, Appendix A~~, GDC 17. SAR, Section 1.4

2. ~~F~~SAR, Chapter ~~[8]~~. 6 6

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(continued)

BASES

"Selection, Design, and Qualification of Diesel Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants,"

REFERENCES (continued)

3. Regulatory Guide 1.9, Rev. ~~[3]~~ 3, ~~[date]~~ July 1993. edit
4. ~~FSAR, Chapter [6].~~
5. FSAR, Chapter ~~[15]~~ 14. edit
6. ~~Regulatory Guide 1.93, Rev. [0], [date].~~
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984 (OCNA078423). edit
4 ~~8.~~ ~~10 CFR 50, Appendix A, GDC 18.~~ SAR, Section 1.4 51
~~9. Regulatory Guide 1.108, Rev. [1], [August 1977].~~ 27
~~10. Regulatory Guide 1.137, Rev. [ ], [date].~~
~~11. ANSI C84.1-1982.~~
~~12. ASME, Boiler and Pressure Vessel Code, Section XI.~~
~~13. IEEE Standard 308-[1978].~~

6. 10 CFR 50.36. 52

17

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources—Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources—Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst-case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4 various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in

(continued)

17

BASES

APPLICABLE SAFETY ANALYSES (continued)

recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown MODES based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration;

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both;

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems; and

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems—Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required

(continued)

17

BASES

LCO (continued)

offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

[Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. The second offsite circuit consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.]

The DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within [10] seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.

Proper sequencing of loads, including tripping of non-essential loads, is a required function for DG OPERABILITY.

[In addition, proper sequencer operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts on the ability to start and maintain energized loads required OPERABLE by LCO 3.8.10.]

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.

(continued)

17

BASES (continued)

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains are required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare features inoperable with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC

(continued)

17

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4 (continued)

power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 provides the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1 (continued)

to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.6 is not required to be met because the required OPERABLE DG(s) is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.9 is excepted because starting independence is not required with the DG(s) that is not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.~~7~~ 2 Diesel Fuel Oil~~, Lube Oil,~~ and Starting Air

BASES

BACKGROUND

3.5 fuel oil

Each diesel generator (DG) is provided with ~~a~~ storage ~~tank~~ ~~having a fuel oil~~ capacity sufficient to operate that diesel for a period of ~~7~~ days while the DG is supplying maximum post loss of coolant accident load demand discussed in the U FSAR, Section ~~[9.5.4.2]~~ 8.3 (Ref. 1). The maximum load demand is calculated using the assumption that at least two DGs are initially available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time needed to replenish the onsite supply from outside sources. 29

Fuel oil is transferred from either storage tank to either day tank by either ~~of two~~ transfer pump~~s~~ (one pump is associated with each storage tank). Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG. All required outside tanks, pumps, and piping are located underground. 29

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. ~~Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3).~~ The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, ~~specific gravity (or API gravity),~~ and impurity level. See Specification 5.5.13, "Diesel Fuel Oil Testing Program," for details. 29

~~The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of [7] days of operation. [The onsite storage in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation.] This supply is sufficient to allow the operator to replenish lube oil from outside sources.~~ 25

Each DG has ~~an~~ a designed air start system ~~with~~ consisting of two redundant banks of two tanks (receivers) each. One bank of two tanks contains adequate capacity (i.e., design margin) for five successive start attempts on the DG without recharging the air start receiver~~(s)~~. 29

(continued)

25

BASES (continued)

APPLICABLE SAFETY ANALYSES

applicable

The ~~initial conditions of~~ Design Basis Accident (DBA) and ~~transient analyses in the FSAR, Chapter [6] (Ref. 4) and Chapter [15] (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.~~

INSERT B 3.8-42A

29

Since diesel fuel oil ~~, lube oil,~~ and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 2).

25

52

LCO

Stored diesel fuel oil is required to have sufficient supply for ~~7~~ 3.5 days of full load operation. It is also required to meet specific standards for quality. ~~Additionally, sufficient lube oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days,~~ supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an ~~anticipated operational occurrence (AOO)~~ abnormality or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," ~~and LCO 3.8.2, "AC Sources - Shutdown."~~

29

26

17

~~The starting air system is required to have a minimum capacity for five successive~~ DG start attempt~~s~~ without recharging the air start receivers.

INSERT B 3.8-42B

29

APPLICABILITY

The AC sources (LCO 3.8.1 ~~and LCO 3.8.2~~) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an ~~AOO~~ abnormality or a postulated DBA. Since stored diesel fuel oil ~~, lube oil,~~ and the starting air subsystem support LCO 3.8.1 ~~and LCO 3.8.2~~, stored diesel fuel oil ~~, lube oil,~~

25

(continued)

<INSERT B3.8-42A>

... for the Diesel Fuel Oil and Starting Air systems are the same as for the DGs which they support. See the appropriate discussions in the Bases for LCO 3.8.1, "AC Sources-Operating."

<INSERT B3.8-42B>

One bank of two tanks (receivers) with pressure ≥ 175 psig provides additional margin to that required for the assumed single ...

BASES

APPLICABILITY (continued)

and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1

In this Condition, the ~~7 day~~ required fuel oil supply for a DG of 20,000 gallons (i.e., 138 inches) is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a 6 day supply of 17,140 gallons (i.e., 118 inches). These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level~~; or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations~~. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

3 29 34 3 29

B.1

~~With lube oil inventory < 500 gal, sufficient lube oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration~~ 25

(continued)

BASES

ACTIONS

~~B.1 (continued)~~ 25

~~of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.~~

B ~~E~~.1

Specification 5.5.13.c edit

This Condition is entered as a result of a failure to meet the acceptance criterion of ~~SR 3.8.3.5~~. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

C ~~D~~.1 2 2

With the new fuel oil properties defined in the Bases for SR 3.8.~~3~~.~~4~~ not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

(continued)

BASES

ACTIONS (continued)

D ~~D~~.1

in the required receivers

175

With starting air receiver pressure < ~~225~~ psig, sufficient ~~capacity for five successive~~ margin to that required for a single DG start attempt~~s~~ does not exist. However, as long as the receiver pressure is ~~≥ 125~~ ≥158 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that ~~most~~ the credited DG start~~s~~ ~~are~~ is accomplished on the first attempt, and the low probability of an event during this brief period. 42 29

E ~~E~~.1

With a Required Action and associated Completion Time required not met, or one or more DGs with fuel oil, ~~lube oil,~~ or starting air subsystem not within limits for reasons other than addressed by Conditions A through ~~E~~ D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable. 42 25

SURVEILLANCE REQUIREMENTS

SR 3.8.~~3~~2.1

This SR provides 3.5 verification that there is an 3.5 adequate inventory of fuel oil in the storage tanks to support each DG's operation for ~~7~~ days at full load. The ~~7~~ day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location. An indicated tank level of 138 inches of fuel oil assures the required volume of 20,000 gallons for tanks T-57A and T-57B. 29

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

~~SR 3.8.3.2~~

~~This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load~~ 25

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.2 (continued)

~~operation for each DG. The [500] gal requirement is based on the DG manufacturer consumption values for the run time of the DG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the DG, when the DG lube oil sump does not hold adequate inventory for 7 days of full load operation without the level reaching the manufacturer recommended minimum level.~~ 25

~~A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.~~

SR 3.8.3.3 2

The tests ~~listed below~~ of fuel oil prior to addition to the storage tanks are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine ~~combustion~~ operation. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between ~~receipt~~ sampling (and associated results) of new fuel and ~~conducting the tests~~ addition of new fuel oil to the storage tank(s) to exceed 31 days. The tests, limits, and applicable ASTM Standards for the tests listed in Specification 5.5.13, "Diesel Fuel Oil Testing Program," are as follows: 29

a. Sample the new fuel oil in accordance with ASTM D4057-[88] (Ref. 6);

b. Verify in accordance with the tests specified in ASTM D975-[81] (Ref. 6) that the sample has:
1. an absolute ~~specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an~~ API gravity at 60°F of ≥ 27° and ≤ 39°,
2. a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, ~~and~~
3. a flash point of ≥ 125°F, and
4. water and sediment within limits. 29

~~c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-[ ] (Ref. 6).~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 2 2 (continued)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO ~~concern~~ since the fuel oil is not added to the storage tanks. edit

~~Within 31 days~~ ~~following~~ further the initial new fuel oil sample, ~~the~~ fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-~~81~~ (Ref. 3) are met for new fuel oil when tested in accordance with ASTM D975-~~81~~ (Ref. 3), except that the analysis for sulfur may be performed in accordance with ASTM D1552-~~90~~ (Ref. 3) or ASTM D2622-~~87~~ (Ref. 3). The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs. edit new 29

<INSERT B 3.8-47A>

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-~~78~~, Method A (Ref. 3). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. ~~For those designs in which the total stored fuel oil volume is contained in two or more interconnected tanks, each tank must be~~ diesel fuel oil is considered and tested separately. edit 15

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

(continued)

<INSERT B3.8-47A>

These additional analyses are required by Specification 5.5.13, "Diesel Fuel Oil Testing Program," to be performed within 31 days following sampling and addition. This 31 days is intended to assure: 1) that the sample taken is not more than 31 days old at the time of adding the fuel oil to the storage tank, and 2) that the results of a new fuel oil sample (sample obtained prior to addition but not more than 31 days prior to) are obtained within 31 days after addition. For circumstances where multiple fuel oil additions are made within a short period of time, the samples taken for each batch added to the storage tank can be composited for a single follow-up analysis.

25

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.~~3.4~~ 2 3

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. ~~The system design requirements provide for a minimum of [five] engine start cycles without recharging. [A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed.]~~ The pressure specified in this SR is intended to reflect the ~~lowest value at which the [five] starts can be accomplished.~~

29

42

INSERT B 3.8-48A

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

~~SR 3.8.3.5~~

~~Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every [31] days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.~~

20

~~SR 3.8.3.6~~

~~Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are ~~required~~ recommended at~~

20

(continued)

<INSERT B3.8-48A>

... margin above the minimum pressure necessary to start the DG utilizing both receiver tanks in one of the two banks.

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.8.3.6 (continued)~~

~~10 year intervals by Regulatory Guide 1.137 (Ref. 2), paragraph 2.f. This SR also requires the performance of the ASME Code, Section XI (Ref. 8), examinations of the tanks. To preclude the introduction of surfactants in the fuel oil system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This SR is for preventive maintenance. The presence of sediment does not necessarily represent a failure of this SR, provided that accumulated sediment is removed during performance of the Surveillance.~~ 20

REFERENCES

1. ~~F~~SAR, Section ~~[9.5.4.2]~~ 8.3.
2. ~~Regulatory Guide 1.137.~~
3. ~~ANSI N195-1976, Appendix B.~~
4. ~~FSAR, Chapter [6].~~
5. ~~FSAR, Chapter [15].~~
~~6.~~ 3. ASTM Standards: D4057-~~[~~ 88 ~~]~~; D975-~~[~~ 81 ~~]~~; D4176-~~[~~86 ~~]~~; D1552-~~[~~ 90 ~~]~~; D2622-~~[~~ 87 ~~]~~; D2276-~~[~~88 ~~]~~, Method A.
7. ~~ASTM Standards, D975, Table 1.~~
8. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

2. 10 CFR 50.36. 52

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.~~4~~ 3 DC Sources—Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and ~~preferred AC~~ 120 VAC vital bus power (via inverters). As required by ~~10 CFR 50, Appendix A,~~ SAR, Section 1.4 GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

edit

51

The {125/~~[250]~~} VDC electrical power system (Red Train and Green Train) consists of two independent and redundant safety related Class 1E DC electrical power subsystems ~~([Train A and Train B])~~. Each subsystem consists of ~~[two]~~ one 125 VDC batter~~ies~~y ~~[(each battery [50% capacity])]~~, the associated battery charger~~[(s)]~~ for each battery, and all the associated control equipment and interconnecting cabling.

~~The 250 VDC source is obtained by use of the two 125 VDC batteries connected in series.~~ Additionally, there is {one} spare battery charger per subsystem, which provides backup service in the event that ~~the preferred~~ a battery charger is out of service. If the spare battery charger is substituted ~~for one of the preferred battery chargers,~~ then the requirements of independence and redundancy between subsystems are maintained.

During inservice normal operation, ~~the~~ each {125/~~[250]~~} VDC ~~load~~ subsystem is powered from the battery charger~~s~~ with the batter~~ies~~y floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batter~~ies~~y. This results in a discharge of the associated battery (and may affect both the system and cell parameters).

30

The ~~[~~Red Train ~~[A]~~ and Green Train ~~[B]~~ DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, ~~[4.16]~~ kV switchgear, and ~~[480]~~ V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the 120 VAC ~~AC~~ vital buses.

edit

(continued)

BASES

BACKGROUND (continued)

The DC power distribution (6) system is described in more detail in Bases for LCO 3.8.[8] "Distributions System—Operating." ~~and for LCO 3.8.10, "Distribution Systems—Shutdown."~~ (17)

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours ~~and~~ in addition to supplying power for the operation of momentary ~~perform three complete cycles of intermittent~~ loads during the 2 hour period as discussed in the ~~F~~SAR, Chapter ~~[8]~~ 9 6 (Ref. 4). (30)

Each 125~~/250~~ VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries for Red Train ~~A~~ and Green Train ~~B~~ DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. ~~Battery size is based on 125% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery discussed in the FSAR, Chapter [8] (Ref. 4). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).~~ (30)

Each ~~Train A and Train B DC electrical power~~ battery charger subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining designed with its battery bank fully charged. Each battery charger also ~~has~~ is sufficient capacity to restore the battery from the design minimum charge to its fully charged state ~~within 24 hours~~ while supplying normal steady state loads. ~~discussed in the FSAR, Chapter [8] (Ref. 4).~~ (30) (15)

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) ~~and~~ transient analyses in the ~~F~~SAR, ~~Chapter [6] (Ref. 6) and~~ Chapter [15] (Ref. ~~7~~), assume that Engineered ~~Safety Feature~~ Safeguards (ES~~F~~) systems are OPERABLE. The DC electrical power system (14) (5)

edit

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. ~~This includes maintaining the DC sources OPERABLE during accident conditions in the event of:~~

~~a. An assumed loss of all offsite AC power or all onsite AC power; and~~

~~b. A worst case single failure.~~

33

In MODES 1 and 2, ~~The~~ the DC sources satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 6). 52

In MODES 3 and 4, the DC sources satisfy Criterion 4 of 10 CFR 50.36.

LCO

The DC electrical power subsystems, each subsystem consisting of ~~two~~ one batter~~ies~~y, battery charger~~s~~ ~~for each battery~~ and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an ~~anticipated operational occurrence (AOO)~~ abnormality or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4). 30 26

An OPERABLE DC electrical power subsystem requires ~~all~~ the associated ~~required~~ batter~~ies~~y and respective charger~~s~~ to be ~~operating and~~ OPERABLE and capable of being connected to the associated DC bus~~(es)~~. 30

to be OPERABLE and connected to the associated DC bus

one of two

one of its

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of ~~AOOs or abnormal transients~~ abnormalities and 26

(continued)

BASES

APPLICABILITY (continued)

b. Adequate core cooling is provided, and ~~containment integrity~~ reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA. edit

The DC electrical power requirements for MODES 5 and 6 are addressed ~~in the Bases for LCO 3.8.5, "DC Sources—Shutdown."~~ by the definition of OPERABILITY ~~for~~ each required supported load 17

ACTIONS

A.1

Condition A represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 2 8 hour limit is consistent with the allowed time for an inoperable DC distribution system train. edit 13

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery chargers and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure would, however, result in the complete loss of the remaining 250/125 VDC electrical power subsystems with attendant loss of ESF functions, continued power operation should not exceed 2 8 hours. The 2 hour Completion Time ~~is based on Regulatory Guide 1.93 (Ref. 8) and~~ reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown. edit 13

B.1 and B.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 12 hours and to MODE 5 36

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 plant systems. ~~The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 8).~~

unit

edit

30

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1 3

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery for battery cell and maintain the battery or a battery cell in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8 7).

edit

(2.15 V per cell average)

IEEE-450 (Ref.7)

edit

edit

edit

~~SR 3.8.4.2~~

~~Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each intercell, interrack, intertier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.~~

~~The limits established for this SR must be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.~~

~~The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.~~

14

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The 12 month Frequency for this SR is consistent with IEEE-450 (Ref. 9), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

14

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of intercell, interrack, intertier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4.

14

Reviewer's Note: The requirement to verify that terminal connections are clean and tight applies only to nickel cadmium batteries as per IEEE Standard P1106, "IEEE Recommended Practice for Installation, Maintenance, Testing and Replacement of Vented Nickel - Cadmium Batteries for Stationary Applications." This requirement may be removed for lead acid batteries.

The connection resistance limits for SR 3.8.4.5 shall be no more than 20% above the resistance as measured during installation, or not above the ceiling value established by the manufacturer.

The Surveillance Frequencies of [12] months is consistent with IEEE-450 (Ref. 9), which recommends cell to cell and terminal connection resistance measurement on a yearly basis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.~~8~~

~~This SR requires that each battery charger be capable of supplying [400] amps and [250/125] V for ≥ [8] hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.~~

~~The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these [18 month] intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.~~

~~This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.~~

48

SR 3.8.~~4~~.~~7~~ 3 2

edit

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements ~~as specified in Reference 4~~.

edit

The Surveillance Frequency of ~~[18 months]~~ 24 is consistent with ~~the recommendations of Regulatory Guide 1.32 (Ref. 10) and Regulatory Guide 1.129 (Ref. 11), which state~~ considerations that the battery service test should be performed during refueling ~~operations~~ outages or at some other outage~~, with intervals between tests not to exceed [18 months]~~.

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edit

~~This SR is modified by two Notes. Note 1 allows the performance of a~~ modified performance discharge test may be performed in lieu of a service test ~~once per 60 months~~.

19

(continued)

BASES

edit

edit

SURVEILLANCE REQUIREMENTS

SR 3.8.~~4.7~~ 3.2 (continued)

The modified performance discharge test (Ref. 7) is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

edit

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test and the test discharge rate must envelope the duty cycle of the service test if the modified performance discharge test is performed in lieu of a service test.

edit

7

~~The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.~~

8

edit

SR 3.8.~~4.8~~ 3.3

A battery performance discharge test is a test of constant current capacity of a battery ~~normally done in the as found condition~~ after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage (Ref. 7).

30

edit

A battery modified performance discharge test is described in the Bases for SR 3.8.~~4.7~~ 3.2. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.~~4.8~~ 3.3; however, only the modified performance discharge test may be used to satisfy 3.3

edit

edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.~~4.8~~ 3.3 (continued) — edit

SR 3.8.~~4.7~~ while satisfying the requirements of SR 3.8.~~4.7~~ 3.2 at the same time. — edit

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. ~~7~~) ~~and IEEE-485 (Ref. 5). These~~, which references recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. — edit

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. ~~9~~), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ~~≥~~ [10%] below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. ~~7~~). — edit 7 30

~~This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.~~ 7 8

REFERENCES

1. ~~10 CFR 50, Appendix A,~~ GDC 17. SAR, Section 1.4 — 51
2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," March ~~10,~~ 1971. — edit
3. IEEE-308-~~1978~~ 1971, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations." — edit
4. ~~F~~SAR, Chapter ~~[8]~~ 8. — 30
5. ~~IEEE-485-[1983], June 1983.~~

(continued)

BASES

REFERENCES (continued)

~~6. FSAR, Chapter [6].~~

5 ~~7~~. FSAR, Chapter ~~[15]~~ 14. — 52

6. 10 CFR 50.36.

~~8. Regulatory Guide 1.93, December 1974.~~

7 ~~9~~. IEEE-450-~~[1987]~~ 1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

~~10. Regulatory Guide 1.32, February 1977.~~

~~11. Regulatory Guide 1.129, December 1974.~~

— edit

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources—Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [14] (Ref. 2), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The DC electrical power subsystems, each subsystem consisting of two batteries, one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the train, are required to be

(continued)

17

BASES

LCO (continued)

OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems—Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies, provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the

(continued)

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BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

(continued)

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BASES (continued)

REFERENCES	1.	FSAR, Chapter [6].
	2.	FSAR, Chapter [14].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.~~6~~ 4 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for ~~LCO 3.8.4~~ 3 "DC Sources—Operating." ~~and LCO 3.8.5, "DC Sources—Shutdown."~~ 17

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the U FSAR, ~~Chapter 6 (Ref. 1) and~~ Chapter 14 ~~15~~ (Ref. 1 ~~2~~), assume Engineered Safety ~~Feature~~ guards systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation. 30

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. ~~This includes maintaining at least one train of DC sources OPERABLE during accident conditions, in the event of:~~ 33

~~a. An assumed loss of all offsite AC power or all onsite AC power; and~~

~~b. A worst-case single failure.~~

Battery cell parameters satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 2). 52

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an ~~anticipated operational occurrence~~ abnormality or a postulated DBA. ~~Electrolyte~~ The limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met. 26 edit

(continued)

BASES (continued)

APPLICABILITY

to be within limits

The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery ~~electrolyte is~~ cell parameters are only required when the DC power source is required to be OPERABLE. ~~Refer to~~ See the Applicability discussion in Bases for LCO 3.8.~~4~~ 3 ~~and LCO 3.8.5.~~

edit 17 49

INSERT B3.8-65A

ACTIONS

A.1, A.2, and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the Category C limits specified in Table 3.8.~~6~~4-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick representative indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cell~~s~~. One hour is considered a reasonable amount of time to perform the required verification.

edit

edit

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to within the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because ~~specific gravity~~ parameter measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to within Category A and B limits. This periodic verification is consistent with the ~~normal Frequency of pilot cell Surveillances.~~ increased potential to exceed these battery cell parameter limits during these conditions.

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(continued)

<INSERT B3.8-65A>

The ACTIONS Table is modified by a Note which indicates that separate Condition entry is allowed for each battery. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DC subsystem. Complying with the Required Actions for one inoperable DC subsystem may allow for continued operation, and subsequent inoperable DC subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

may not be available

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement ~~is not assured~~ and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as ~~of Condition A within~~ ~~not completing~~ the Required Actions ~~the required~~ Completion Time or average electrolyte temperature ~~of representative~~ cells falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable. pilot cell or

edit

edit

Therefore, the battery must be immediately declared inoperable

and associated

of Condition A not met

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SURVEILLANCE REQUIREMENTS

SR 3.8.~~6~~ 4.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte level and temperature of pilot cells.

edit

30

INSERT FROM PAGE B 3.8-67

SR 3.8.~~6~~ 4.2 3

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). In addition, within 24 hours of a battery discharge < ~~[~~110~~]~~ V or a battery overcharge > ~~[150]~~ 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ ~~[~~110~~]~~ V, do not constitute a battery discharge

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.~~6~~.~~2~~ 4.3 (continued)

provided battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

MOVE & INSERT on PAGE B 3.8-66

SR 3.8.~~6~~.~~3~~ 4.2 and SR 3.8.4.4

This Surveillance verification that the pilot cell and the average temperature of representative cells is ≥ ~~60~~ 60°F is consistent with a recommendation of IEEE-450 (Ref. 3), which states that the temperature of electrolytes in the pilot cell should be determined at least once per month and that the temperature in representative cells (~10% of all connected cells) should be determined on a quarterly basis.

edit

edit

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

30

Table 3.8.~~6~~ 4-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 3.8.~~6~~ 4-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates

edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.~~6~~4-1 (continued)

suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on a recommendation of IEEE-450 (Ref. 3), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ ~~[1.200]~~ 1.195 ~~(0.015 below the manufacturer fully charged nominal specific gravity or a battery charging current that had stabilized at a low value).~~ This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C). 30

The specific gravity readings are corrected for actual electrolyte temperature ~~and level~~. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. ~~The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.~~ 54

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ ~~[1.195]~~ 1.190 ~~(0.020 below the manufacturer fully charged, nominal specific gravity)~~ with the average of all connected cells > ~~[1.205]~~ 1.195 ~~(0.010 below the manufacturer fully charged nominal specific gravity)~~. These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery. 30

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.~~6~~ 4-1 (continued)

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limits of average specific gravity ≥ [~~1.195~~] 1.190 is based on manufacturer recommendations ~~(0.020 below the manufacturer recommended fully charged, nominal specific gravity)~~. In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery. (30)

(b) and (c) ~~The~~ footnotes to Table 3.8.~~6~~ 4-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.~~6~~ 4-1 requires the above mentioned correction for electrolyte level and temperature~~, with the exception that level correction is not required when battery charging current is < [2] amps on float charge. This current provides, in general, an indication of overall battery condition~~. (54) edit

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.~~6~~ 4-1 allows the float charge current to be used as an alternate to specific gravity for

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.~~6-1~~ 4 (continued)

up to ~~[7]~~ days following a battery recharge. Within ~~[7]~~ days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than ~~that~~ ~~[7]~~ days. edit

~~Reviewer's Note: The value of [2] amps used in footnote (b) and (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition.~~ edit

REFERENCES

1. ~~F~~SAR, Chapter ~~[6].~~ 14. edit
2. ~~FSAR, Chapter [16].~~ 10 CFR 50.36. 52
3. IEEE-450-~~[1980]~~ 1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications." edit

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.~~7~~ 5 Inverters—Operating

INSERT B3.8-71A

BASES

BACKGROUND

The inverters are the preferred source of power for the ~~AC~~ 120 VAC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital bus. The inverters ~~can be~~ are normally powered from ~~an internal AC source/rectifier or from the station battery. The station battery~~ provides an uninterruptible power source for the safety significant instrumentation and controls, including ~~for~~ the Reactor Protection System (RPS), [redacted] the Engineered ~~Safety Feature~~ Safeguards Actuation System (ESFAS), and the Emergency Feedwater Initiation and Control (EFIC) System. Specific details on inverters and their operating characteristics are found in the FSAR, Chapter ~~[8]~~ (Ref. 1).

edit

31

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, ~~Chapter [6] (Ref. 2), and~~ Chapter ~~[14]~~ (Ref. ~~3~~ 2), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the ~~RPS and ESFAS~~ safety significant instrumentation and controls so that the fuel, Reactor Coolant System, and ~~containment~~ reactor building design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits," Section 3.4, "Reactor Coolant System (RCS)," and Section 3.6, "~~Containment~~ Reactor Building Systems."

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edit
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edit

INSERT B 3.8-71B

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. ~~This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:~~

~~a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and~~

~~b. A worst-case single failure.~~

33

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 3) in MODES 1 and 2. In MODES 3 and 4, the inverters satisfy Criterion 4 of 10CFR50.36.

52

(continued)

<INSERT B3.8-71A>

... the 125 VDC Electrical Power System. An alternate AC source is also provided for the inverters which can be used during maintenance, during switchover to or from a swing inverter, or in case of inverter failure. The inverters ...

<INSERT B3.8-71B>

Additionally, there are two swing inverters (one per train) which provide backup service in the event that an inverter is out of service. If the swing inverter is placed in service, requirements of independence and redundancy between trains are maintained.

BASES (continued)

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an ~~anticipated operational occurrence (AOO)~~ abnormality or a postulated DBA. 26

Maintaining the required inverters OPERABLE ensures that the ~~RPS and ESFAS~~ redundancy incorporated into the design of the safety significant instrumentation and controls is maintained. The four required inverters ~~(two per train)~~ ensure an uninterruptible supply of AC electrical power to the ~~AC~~ 120 VAC vital buses even if the 4.16 kV safety buses are de-energized. 31 edit

OPERABLE inverters require the associated 120 VAC vital bus to be powered by the inverter with output voltage ~~and frequency~~ within tolerances, and power input to the inverter from ~~a~~ the 125 VDC ~~station battery~~ Electrical Power System with associated OPERABLE. ~~Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply.~~ 31

This LCO is modified by a Note that allows ~~one (two)~~ one required 120 VAC inverter~~s~~ to be disconnected from ~~a (common) battery for~~ its associated DC bus ~~≤ 24 hours~~ ≤ 2 hours to allow load transfer to or from the swing inverter, if the vital bus~~(es)~~ is powered from an alternate ~~Class 1E constant voltage transformer or inverter using internal AC source~~ during the period and all other inverters are ~~operable. This allows an equalizing charge to be placed on one battery. If the inverters were not disconnected, the resulting voltage condition might damage the inverter(s). These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC vital bus while taking into consideration the time required to perform an equalizing charge on the battery bank.~~ 45

~~The intent of this Note is to limit the number of inverters that may be disconnected. Only those inverters associated with the single battery undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.~~

(continued)

BASES (continued)

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormalities ~~AOOs or abnormal transients~~; and 26

b. Adequate core cooling is provided, and reactor building ~~containment~~ OPERABILITY and other vital functions are maintained in the event of a postulated DBA. edit

Inverter requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load ~~covered in the Bases for LCO 3.8.8, "Inverters—Shutdown."~~ 17

ACTIONS

A.1

With a required inverter inoperable, its associated ~~AC~~ 120 VAC vital bus becomes inoperable unless ~~until~~ it remains ~~is manually re-energized from its [Class 1E constant voltage source transformer or inverter using internal AC source]~~. edit 31

For this reason, a Note has been included in Condition A requiring entry into the Conditions and Required Actions of LCO 3.8.~~9~~ 6, "Distribution Systems—Operating." This ensures the vital bus is re-energized within ~~2~~ 8 hours. Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit takes ~~is based upon engineering judgment, taking~~ into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the ~~AC~~ 120 VAC vital bus is powered from its alternate AC ~~constant voltage~~ source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the ~~AC~~ 120 VAC vital buses is the preferred source for powering instrumentation trip setpoint devices. 13 26 edit 31 edit

B.1 and B.2

If the Required Action and associated ~~inoperable devices or components cannot be restored to OPERABLE status within the required~~ Completion Time, are not met the edit

(continued)

BASES

ACTIONS

B.1 and B.2 (continued) 12

unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within [6] hours and to 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 plant systems. 36

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and ~~[AC]~~ 120 VAC vital buses energized from the inverter. The verification of proper voltage ~~[and frequency]~~ output ensures that the required power is readily available for the instrumentation ~~[of the RPS and ESFAS]~~ connected to the ~~[AC]~~ 120 VAC vital buses. The [7] day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions. edit 31 edit 46 31

REFERENCES

1. ~~F~~SAR, Chapter ~~[8]~~ 9. edit
~~2. FSAR, Chapter [6].~~ edit
2 ~~3~~. ~~F~~SAR, Chapter ~~[14]~~ 9.

3. 10 CFR 50.36. 52

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters—Shutdown

17

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [14] (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protection System and Engineered Safety Features Actuation System (ESFAS) instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each AC vital bus during MODES 5 and 6 ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of the NRC Policy Statement.

(continued)

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BASES (continued)

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the vital bus be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, "Distribution Systems—Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity

(continued)

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BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

(continued)

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BASES (continued)

REFERENCES

1. FSAR, Chapter [6].
2. FSAR, Chapter [14].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.~~9~~ 6 Distribution Systems—Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and ~~AC~~ 120 VAC vital bus electrical power distribution systems are divided by train into [two] redundant and independent AC, DC, and ~~AC~~ 120 VAC vital bus electrical power distribution subsystems. edit edit

~~The~~ Each AC electrical power subsystem ~~for each train~~ consists of ~~a primary~~ Engineered ~~Safety Feature~~ Safeguards (ESF) 4.16 kV bus and ~~secondary~~ 480 ~~and 120~~ V buses, ~~distribution panels, motor control centers and load centers~~. Each 4.16 kV ESF bus ~~has at least one separate and independent~~ two offsite sources of power as well as a dedicated onsite diesel generator (DG) source, as described in the Bases for LCO 3.8.1, "AC Sources - Operating." ~~Each [4.16 kV ESF bus] is normally connected to a preferred offsite source. After a loss of the preferred offsite power source to a 4.16 kV ESF bus, a transfer to the alternate offsite source is accomplished by utilizing a time delayed bus undervoltage relay.~~ If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. ~~Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources—Operating," and the Bases for LCO 3.8.4, "DC Sources—Operating."~~ (32) (32)

~~The secondary AC electrical power distribution system for each train includes the safety related load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1.~~ (32)

The 120 VAC vital buses are arranged in two load groups per ~~train~~ subsystem and are normally powered from the inverters. The alternate power supply for the so vital buses ~~are Class 1E constant voltage source transformers~~ is powered from the same ~~train~~ subsystem as ~~the~~ associated inverter, and its use is governed by LCO 3.8.~~7~~ 5, "Inverters—Operating." ~~Each constant voltage source transformer is powered from a Class 1E AC bus.~~ (32)

There are two independent 125~~/250~~ VDC electrical power distribution subsystems (one for each train).

The list of ~~all~~ required distribution buses ~~is presented in Table B 3.8.9-1.~~ (6) Other ES buses are not required by this Specification, but may be required to support OPERABILITY of other required equipment. Inoperability of such buses would result in inoperability of the supported equipment.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, ~~Chapter [6] (Ref. 1) and~~ Chapter [14] 2 (Ref. 1), assume ESF systems are OPERABLE. The AC, DC, and AC 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment reactor building design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits," Section 3.4, "Reactor Coolant System (RCS)," and Section 3.6, ~~Containment~~ "Reactor Building Systems."

edit
edit
edit
edit
edit

The OPERABILITY of the AC, DC, and AC 120 VAC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. ~~This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:~~ 33

~~a. An assumed loss of all offsite power or all onsite AC electrical power; and~~

~~b. A worst-case single failure.~~

In MODES 1 and 2, The distribution systems satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2). In MODES 3 and 4, the distribution systems satisfy Criterion 4 of 10 CFR 50.36. 52

LCO

The required power distribution subsystems listed in Table B 3.8.9 6-1 ensure the availability of AC, DC, and AC 120 VAC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an ~~anticipated operational occurrence (AOO)~~ abnormality or a postulated DBA. The AC, DC, and AC 120 VAC vital bus electrical power distribution subsystems are required to be OPERABLE. edit 26 edit 32

Maintaining the ~~Train A and Train B~~ AC, DC, and AC 120 VAC vital bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

(continued)

BASES

LCO (continued)

OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, motor control centers, ~~and distribution panels~~ to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE 120 V AC vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated ~~inverter via inverted DC voltage, inverter using internal AC source,~~ or from its alternate ~~Class 1E constant voltage transformer.~~ AC source. — 26 edit — 32 edit

In addition, Cross- tie breakers between redundant safety related AC, DC, and AC vital bus power distribution subsystems, ~~if they exist,~~ must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any cross- tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit. edit

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of ~~AOOs~~ abnormalities or abnormal transients; and — 26 edit

b. Adequate core cooling is provided, and ~~containment~~ reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are ~~covered in the Bases for LCO 3.8.10, "Distribution Systems—Shutdown."~~ addressed by the definition of OPERABILITY for each required supported load — 17

(continued)

BASES (continued)

ACTIONS

A.1

OPERABLE portions of the may be

With one or more required AC electrical power distribution subsystems inoperable, the remaining AC electrical power distribution subsystem(s) ~~in the other train is~~ capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, and motor control centers~~, and distribution panels~~ must be restored to OPERABLE status within 8 hours. 47 edit 26 edit

Condition A worst case ~~scenario~~ is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to ~~2~~ 8 hours. This could lead to a total of ~~10~~ 16 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again 13

(continued)

BASES

ACTIONS

A.1 (continued)

become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

may be
electrical power distribution subsystems
47
35

B.1

or more
120 VAC

portions of the 120 VAC
subsystem(s)

With one required AC vital bus inoperable, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus must be restored to OPERABLE status within 2 hours by powering the bus from the associated inverter via inverted DC, ~~inverter using internal AC source~~, or ~~Class 1E constant voltage transformer~~. 120 VAC 8 edit 13 32

affected bus(es)
AC source.
from its alternate

or more 120 VAC
subsystem(s) edit

Condition B represents one AC vital bus without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital buses and restoring power to the affected vital bus. alternate 47

un-
subsystem(s)
subsystem(s).

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of: 8 13

(continued)

BASES

ACTIONS

B.1 (continued)

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The $\boxed{2}$ 8 hour Completion Time 120 VAC takes into subsystem(s) account the importance to safety of restoring the $\boxed{AC}$ vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital bus$\boxed{es}$ subsystem, and the low probability of a DBA occurring during this period. 13 edit

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of $\boxed{10}$ 16 hours, since initial failure of the LCO, to restore the 120 VAC vital bus ~~distribution system~~ subsystem(s). At this time, an AC train could again become inoperable, and 120 VAC vital bus $\boxed{distribution}$ subsystem(s) restored OPERABLE. This could continue indefinitely. 13

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

(continued)

BASES

ACTIONS (continued)

C.1 one or more subsystems may be OPERABLE portions of the 47 32

With DC ~~bus(es) in one train~~ inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the [required] DC buses must be restored to OPERABLE status within 2 8 hours by powering the bus from the associated battery ~~or charger~~ one of the two associated chargers. 13 47

Condition C represents one or more DC subsystem(s) ~~train~~ without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This 2 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 8 hours, is acceptable because of: 13

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

~~The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).~~ 13

(continued)

BASES

ACTIONS

C.1 (continued)

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of ~~10~~ 16 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable and DC distribution restored OPERABLE. This could continue indefinitely. 13

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to 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 plant systems. 12 36

E.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. ~~When more than one Condition is entered, and this results in the loss of a required function, the plant is in a condition outside the accident analysis.~~ Therefore, no additional time is justified for 47

(continued)

BASES

ACTIONS

E.1 (continued)

continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.~~9~~ 6.1

This Surveillance verifies that the ~~[required]~~ AC, DC, and ~~AC~~ 120 VAC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained. ~~and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses.~~ The ~~7~~ day Frequency takes into account the redundant capability of the AC, DC, and ~~AC~~ 120 VAC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions. 31

edit 43 44 edit

REFERENCES

~~1. FSAR, Chapter [6].~~

1 ~~2~~. ~~F~~SAR, Chapter ~~[~~14~~]~~.

~~3. Regulatory Guide 1.93, December 1974.~~

2. 10 CFR 50.36.

edit 13 52

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Table B 3.8.~~9~~ 6-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	RED TRAIN ~~A*~~	GREEN TRAIN ~~B*~~
AC ~~safety buses~~ Electrical Power Distribution Subsystems	~~[~~4160 V~~]~~ ~~[~~480 V~~]~~ ~~[~~480 V~~]~~ ~~[120 V]~~	ESF Bus ~~[NB01]~~ A3 Load Centers ~~[NG01, NG03]~~ B5 Motor Control Centers ~~[NG01A, NG01I, NG01B, NG03C, NG03I, NG03D]~~ B51, B52, B53, B57 ~~Distribution Panels [NP01, NP03]~~	ESF Bus ~~[NB02]~~ A4 Load Centers ~~[NG02, NG04]~~ B6 Motor Control Centers ~~[NG02A, NG02I, NG02B, NG04C, NG04I, NG04D]~~ B61, B62, B63, B65, B56* and B55 ~~Distribution Panels [NP02, NP04]~~
DC ~~Buses~~	~~[~~125 V~~]~~	Bus ~~[NK01]~~ D01 Bus ~~[NK03]~~ RA1 Distribution Panels ~~[NK41, NK43, NK51]~~ D11	Bus ~~[NK02]~~ D02 Bus ~~[NK04]~~ RA2 Distribution Panels ~~[NK42, NK44, NK52]~~ D21
120 VAC Vital ~~Buses~~ Subsystems	~~[~~120 V~~]~~	Bus ~~[NN01]~~ RS1 Bus ~~[NN03]~~ RS3	Bus ~~[NN02]~~ RS2 Bus ~~[NN04]~~ RS4

~~* Each train of the AC and DC electrical power distribution systems is a subsystem.~~

* Swing bus (normally associated with Green Train). Bus B55 is powered from bus B56.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems—Shutdown

17

BASES

BACKGROUND

A description of the AC, DC and AC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 1) and Chapter [14] (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

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BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC, DC, and AC vital bus electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

(continued)

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BASES (continued)

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required decay heat removal (DHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.5 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the DHR ACTIONS would not be entered. Therefore, Required Action A.2.6 is provided to direct declaring DHR inoperable, which results in taking the appropriate DHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.8.10.1 This Surveillance verifies that the AC, DC, and AC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.
REFERENCES	1. FSAR, Chapter [6]. 2. FSAR, Chapter [14].

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
3.9.1	3.9.1	Boron Concentration .
3.9.2	3.9.2	Nuclear Instrumentation
3.9.3	3.9.3	Reactor Building Penetrations
3.9.4	3.9.4	Decay Heat Removal (DHR) and Coolant Circulation-High Water Level
3.9.5	3.9.5	Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level
3.9.6	3.9.6	Refueling Canal Water Level

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2

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

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

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable during CORE ALTERATIONS.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. No OPERABLE source range neutron flux monitor.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

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

3.9 REFUELING OPERATIONS

3.9.3 Reactor Building Penetrations

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

a. The equipment hatch is capable of being closed;

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

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

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

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

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.9 REFUELING OPERATIONS

3.9.4 Decay Heat Removal (DHR) and Coolant Circulation

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

--NOTE--

The required DHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DHR loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy DHR loop requirements.	Immediately
	AND	
	A.4 Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

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

3.9 REFUELING OPERATIONS

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

LCO 3.9.5 Two DHR loops shall be OPERABLE.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal during refueling ensures that the reactor remains subcritical during MODE 6. The refueling boron concentration is specified for the coolant in each of these volumes since each volume has direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit specified in the COLR ensures an overall core reactivity of $k_{eff} \leq 0.99$ during fuel handling, with all CONTROL RODS out.

SAR, Section 1.4, GDC 26 requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Makeup and Purification System has the ability to initiate and maintain a cold shutdown condition in the reactor.

During refueling, the spent fuel pool, the transfer tube, the refueling canal and the reactor vessel are connected. As a result, the soluble boron concentration is relatively the same in each of these volumes.

Operation of the Decay Heat Removal (DHR) System in the RCS mixes the added concentrated boric acid with the water in the refueling canal. The DHR System is in operation during refueling (see LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures ensure the k_{eff} of the core will remain $\leq$ 0.99 during the refueling operation.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36. (Ref. 2).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling canal while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of $\leq$ 0.99 is maintained during fuel handling operations with CONTROL RODS and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

Violation of the LCO provides a potential for an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical.

Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.5, "Safety Rod Insertion Limits," and LCO 3.2.1, "Regulating Rod Insertion Limits," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of the RCS or the refueling canal is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, action to restore the concentration must be initiated immediately.

There is no unique design basis event analysis that requires a specific rate of boration. The only requirement is to restore the boron concentration to its required value as soon as possible.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined every 72 hours by chemical analysis.

The Frequency is based on industry experience, which has shown 72 hours to be adequate.

REFERENCES

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

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation (NI) System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of temporary detectors is permitted, provided the LCO requirements are met.

The installed source range neutron flux monitor channels include fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux. The instrumentation also provides continuous visual indication in the control room to alert operators to a significant change in neutron flux. The NI system is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

An OPERABLE source range neutron flux monitor is required to provide indication to alert the operator to unexpected changes in core reactivity, such as may be caused by a boron dilution accident or an improperly loaded fuel assembly (Ref. 1).

The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that the reactor remains subcritical. The source range neutron flux monitors are not credited for boron dilution event mitigation in the safety analysis.

The source range neutron flux monitors satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO requires one source range neutron flux monitor OPERABLE to ensure that monitoring capability is available to detect changes in core reactivity. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS. This additional requirement ensures redundant monitoring capability when positive reactivity changes are being made to the core.

The use of temporary detectors is permitted for purposes of complying with this LCO. If used, the temporary detectors should be functionally equivalent to the installed source range monitors and satisfy applicable Surveillance Requirements.

APPLICABILITY

In MODE 6, the source range neutron flux monitor must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux."

In MODE 1, the neutron flux level is above the indicated range of the monitors. Thus, they are no longer relied upon for reactivity or power level monitoring. Hence, there are no requirements on source range neutron flux monitors in MODE 1.

ACTIONS

A.1 and A.2

With only one required source range neutron flux monitor OPERABLE during CORE ALTERATIONS, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no required source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status or until the Applicability is exited.

B.2

With no required source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made in accordance with Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are restored to an OPERABLE status. This stabilized condition is verified by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Changes in fuel loading and core geometry can also result in significant differences between source range channels, but each channel should be consistent with its local conditions. When in MODE 6 with only one channel OPERABLE, a CHANNEL CHECK is still required. However, in this condition, a redundant source range instrument may not be available for comparison. The CHANNEL CHECK provides verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for the same instruments in LCO 3.3.9.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear instrument is a complete check and re-adjustment of the channel, from the pre-amplifier input to the indicator. The 18 month Frequency is based on industry experience which has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. SAR, Section 1.4, GDC 13, GDC 26, GDC 28, and GDC 29.
2. SAR, Section 14.1.2.4.
3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Reactor Building Penetrations

BASES

BACKGROUND

During the movement of irradiated fuel assemblies within the reactor building, a release of fission product radioactivity within the reactor building will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, the containment of fission products is accomplished by maintaining the reactor building OPERABLE as described in LCO 3.6.1, "Reactor Building". In MODE 6, the potential for reactor building pressurization as a result of an accident is not likely; therefore, requirements to isolate the reactor building from the outside atmosphere can be less stringent. In order to make this distinction, the penetration requirements are referred to as "reactor building closure" rather than "reactor building OPERABILITY." Reactor building closure means that all potential direct release paths are closed or capable of being closed. Since there is no potential for significant reactor building pressurization, the Appendix J leakage criteria and tests are not required.

The reactor building serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100. Additionally, the reactor building provides radiation shielding from the fission products that may be present in the reactor building atmosphere following accident conditions.

The reactor building equipment hatch, which is part of the reactor building pressure boundary, provides a means for moving large equipment and components into and out of the reactor building. During the movement of irradiated fuel assemblies within the reactor building, the equipment hatch must be capable of being closed.

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch cover following a required evacuation of the reactor building, and that any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover be capable of being quickly removed (Ref. 1). Should a fuel handling accident occur inside the reactor building, 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. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

The reactor building air locks, which are also part of the reactor building pressure boundary, provide a means for personnel access. During MODES 1, 2, 3, and 4 unit operation is in accordance with LCO 3.6.2, "Reactor Building Air Locks." Each air lock has a door at each end. The doors are normally interlocked to prevent simultaneous opening when the reactor building OPERABILITY is required. During unit shutdown when reactor building OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods. During the movement of irradiated fuel assemblies within the reactor building, closure requires that one door in each air lock be capable of being closed. The door interlock mechanism may remain disabled.

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors are open, that a specific individual(s) is designated and available to close an airlock door following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door be capable of being quickly removed (Ref. 3). Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency air lock doors will be closed following evacuation of the reactor building.

The requirements on reactor building penetration closure ensure that a release of fission product radioactivity from within the reactor building will be restricted to within regulatory limits.

The Reactor Building Purge System includes a supply penetration and exhaust penetration. During MODES 1, 2, 3, and 4, the valves in the supply and exhaust penetrations are secured in the closed position. The system is not subject to a Specification in MODE 5.

In MODE 6, the purge system is used for temperature control, and all four valves may be closed by an operator based on an indication of high radiation. This LCO requires that an OPERABLE radiation monitor be present on the purge exhaust flow path to provide the necessary indication to the operator.

Other reactor building penetrations that provide direct access from the reactor building atmosphere to outside atmosphere must be isolated on at least one side by a closed manual or automatic isolation valve, blind flange, or equivalent, or capable of being isolated by an OPERABLE isolation valve. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other reactor building penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During the movement of irradiated fuel assemblies within the reactor building, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 4). The requirement of a minimum decay time of 100 hours prior to CORE ALTERATIONS ensures that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in Reference 4.

Reactor building penetrations satisfy Criterion 4 of 10 CFR 50.36 (Ref. 5).

LCO

This LCO limits the consequences of a fuel handling accident in the reactor building by limiting the potential escape paths for fission product radioactivity from the reactor building. The LCO requires any penetration providing direct access from the reactor building atmosphere to the outside atmosphere to be closed or capable of being closed by an OPERABLE reactor building isolation valve. This LCO requires the reactor building purge isolation valves and the purge exhaust flow path radiation monitor be OPERABLE.

The reactor building personnel airlock doors and/or the equipment hatch may be open during movement of irradiated fuel in the reactor building provided that one door is capable of being closed in the event of a fuel handling accident. Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, that a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed (Ref. 1 and 3). For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

The definition of "direct access from the reactor building atmosphere to the outside atmosphere" is any path that would allow for the transport of reactor building atmosphere to any atmosphere located outside of the reactor building structure. This includes the Auxiliary Building. As a general rule, closed systems do not constitute a direct path between the reactor building and the outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomenon should not be postulated as part of the evaluation process.

APPLICABILITY

The reactor building penetration requirements are applicable during movement of irradiated fuel assemblies within the reactor building because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, the reactor building penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within the reactor building is not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on reactor building penetration status.

ACTIONS

A.1

With the reactor building equipment hatch, air locks, or any reactor building penetration that provides direct access from the reactor building atmosphere to the outside atmosphere not in the required status, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within the reactor building. Performance of this action shall not preclude moving a component to a safe position.

These actions remove the potential for an event which may require reactor building closure to prevent a significant radioactivity release.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the reactor building penetrations required to be in its closed position is in that position.

The Surveillance is performed every 7 days during the movement of irradiated fuel assemblies within the reactor building. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

This Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the reactor building will not result in a release of fission product radioactivity to the environment in excess of that recommended by Standard Review Plan Section 15.7.4 (Ref. 1, 3 and 6).

SR 3.9.3.2

This Surveillance demonstrates that each reactor building isolation valve actuates to its isolation position on manual initiation. The 18 month Frequency maintains consistency with other similar reactor building isolation valve testing requirements found in Section 3.6. This Surveillance will ensure that the isolation valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the reactor building.

The SR is modified by a Note stating that this demonstration is not applicable to valves in isolated penetrations. LCO 3.9.3.c.1 provides the option to close penetrations in lieu of requiring isolation capability.

SR 3.9.3.3

This SR requires a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor. The CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION is performed consistent with the setpoint requirements. The 18 month Frequency is based on operating experience and is consistent with the typical operating cycle.

REFERENCES

1. Safety Evaluation Report related to ANO-1 Amendment No. 195, April 16, 1999.
2. SAR, Section 5.2.2.1.3.
3. Safety Evaluation Report related to ANO-1 Amendment No. 184, September 20, 1996.
4. SAR, Section 14.2.2.3.
5. 10 CFR 50.36.
6. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Decay Heat Removal (DHR) and Coolant Circulation

BASES

BACKGROUND

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1), and to provide mixing of the reactor coolant to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Service Water System. The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s), bypassing the heat exchanger(s) and throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is provided by the continuous operation of the DHR System.

APPLICABLE SAFETY ANALYSES

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not reduced. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.

The DHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

Only one DHR loop is required for decay heat removal in MODE 6. The operating DHR loop provides:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

To be considered OPERABLE, a DHR loop includes a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in the 'A' hot leg and is returned to the reactor vessel via the core flood tank injection nozzles.

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode.

The LCO is modified by a Note that allows the required DHR loop to be removed from operation for up to 1 hour in an 8 hour period, provided no operation that would cause reduction of the RCS boron concentration is in progress. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allowance permits operations such as core mapping, alterations or maintenance in the vicinity of the reactor vessel nozzles and RCS to DHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling canal.

APPLICABILITY

One DHR loop must be OPERABLE and in operation in MODE 6 to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level < 23 feet above the top of the fuel assemblies seated in the reactor vessel, are located in LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

If DHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Therefore, actions that reduce boron concentration shall be suspended immediately.

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling canal water level 23 feet above the fuel assemblies seated in the reactor vessel provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading an irradiated fuel assembly, is prudent under this condition.

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

Restoration of one decay heat removal loop is required because this is the only active method of removing decay heat. Dissipation of decay heat through natural convection to the large inventory of water in the refueling canal should not be relied upon for an extended period of time. The immediate Completion Time reflects the importance of restoring an adequate decay heat removal loop.

A.4

If DHR loop requirements are not met, all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere shall be closed within 4 hours.

If no means of decay heat removal can be restored, the core decay heat could raise temperatures and cause boiling in the core which could result in increased levels of radioactivity in the reactor building atmosphere. Closure of the penetrations providing access to the outside atmosphere will prevent the uncontrolled release of radioactivity to the environment.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. Verification includes flow, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the DHR System.

REFERENCES

1. SAR, Section 1.4.
2. SAR, Section 9.5.
3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

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

BASES

BACKGROUND

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (Ref. 1), and to provide mixing of the reactor coolant to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the Service Water System. The coolant is then returned to the reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for normal cooldown/decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s), bypassing the heat exchanger(s) and by throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is provided by the continuous operation of the DHR System.

APPLICABLE SAFETY ANALYSES

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. However, without a large water inventory to provide a backup means of decay heat removal, an additional train of the DHR System is required to be OPERABLE in order to provide a backup.

The DHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

In MODE 6, with the water level < 23 feet above the top of the fuel seated in the reactor vessel (corresponds to approximately 390 feet above sea level), two DHR loops must be OPERABLE. Additionally, as required by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation," one DHR loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

To be considered OPERABLE, a DHR loop must consist of a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the temperature. The flow path starts in the 'A' hot leg and is returned to the reactor vessel via the core flood tank injection nozzles.

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.

Both DHR pumps may be aligned to the Borated Water Storage Tank (BWST) to support filling of the refueling canal or the performance of required testing.

APPLICABILITY

Two DHR loops are required to be OPERABLE, and one in operation in MODE 6, with the water level < 23 feet above the top of the fuel seated in the reactor vessel, (corresponds to approximately 390 feet above sea level), to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6 are located in LCO 3.9.4.

ACTIONS

A.1 and A.2

With fewer than the required loops OPERABLE, action shall be immediately initiated and continued until the DHR loop is restored to OPERABLE status or until ≥ 23 feet of water level is established above the fuel seated in the reactor vessel. When the water level is established at ≥ 23 feet above the fuel seated in the reactor vessel, the Applicability will change to that of LCO 3.9.4, and only one DHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary due to the increased risk of operating without a large available heat sink.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

Verification that each required pump is available ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. Alternatively, verification that a DHR pump is in operation as required by SR 3.9.4.1 also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. SAR, Section 1.4.
2. SAR, Section 9.5.
3. 10 CFR 50.36.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the reactor building requires a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. During refueling, this maintains sufficient water level to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident within 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in the reactor building postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) 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 refueling cavity water. The fuel pellet to cladding gap is assumed to contain 12% of the total fuel rod iodine inventory (Ref. 2).

The fuel handling accident analysis inside the reactor building is described in Reference 2. With a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, 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. 3).

Refueling canal water level satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

A minimum refueling canal water level of 23 feet above the top of the irradiated fuel assemblies seated in the reactor pressure vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside the reactor building are within acceptable limits as provided by 10 CFR 100.

APPLICABILITY

LCO 3.9.6 is applicable during movement of irradiated fuel assemblies within the reactor building. The LCO minimizes the possibility of a fuel handling accident in the reactor building that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in the reactor building, there can be no significant radioactivity release as a result of a postulated fuel handling accident in the reactor building.

ACTIONS

A.1

With a water level of < 23 feet above the top of the irradiated fuel assemblies seated with the reactor pressure vessel, all operations involving the movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside the reactor building (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. SAR Section 14.2.2.3.

3. 10 CFR 100.10.

4. 10 CFR 50.36.

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, 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 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 CTS 3.8.9 provides the required actions should one or more of the preceding Specifications not be met. CTS 3.8.9 establishes measures that are considered equivalent to the Required Actions of ITS 3.9.1 Condition A, ITS 3.9.2 Condition A and Required Action B.1, and ITS 3.9.3 Condition A. Although the exact wording is not the same, these are considered equivalent actions and adoption of the ITS requirements constitutes an administrative change. In addition, the Completion Time of "immediately" has been annotated on the CTS markup. This is implicit in a number of CTS actions and explicit in other CTS actions. The addition of this immediate Completion Time establishes Required Actions consistent with those specified in the ITS.

A4 The CTS 3.8.3.a Note * to allow the decay heat removal loop to be secured for periods up to 1 hour per 8 hour period was modified to reflect the exact wording of the ITS LCO 3.9.4 Note. The modification of the CTS 3.8.3.a Note * involved two changes that are both considered administrative in nature.

The first change added words that state that reactor coolant boron concentration reductions are not allowed during the period of time associated with the secured decay heat removal loop. This is consistent with the CTS (per CTS 3.1.1.1.B) which permits boron concentration reductions only when at least one decay heat removal pump is circulating reactor coolant. This requirement is implicitly retained in the ITS through 3.9.4 Required Action A.1 which directs that operations involving a reduction of the reactor coolant boron concentration be immediately suspended should the required reactor coolant circulation not be present, and is explicitly established in the LCO Bases for 3.9.4.

The second change involved the deletion of the words that restricted the applicability of this Note to "during the performance of core alterations." The allowance to secure the decay heat removal loop for a limited period of time in the CTS was dependent upon the availability of a backup source of decay heat removal because the Note modified the decay heat loop OPERABILITY requirements when reactor coolant level was greater than 23 feet above the fuel seated in the reactor pressure vessel. This restriction is inherently present in the ITS through the structure of the Applicability statements for LCOs 3.9.4 and 3.9.5 and the presence of the Note in LCO 3.9.4.

A5 CTS 3.8.9 and 3.8.10 state that the provisions of CTS 3.0.3 are not applicable. This exception is necessary in the CTS because of the concurrent use of CTS 3.8.9 as the Required Actions and associated Completion Times for a number of CTS Specifications (CTS 3.8.1 through CTS 3.8.8), several of which are MODE independent. The ITS 3.9, "REFUELING OPERATIONS" series of specifications will contain appropriate MODES, Applicabilities, Conditions and Surveillance Requirements such that the exception to LCO 3.0.3 will no longer be necessary. Further, the LCO 3.0.3 exception is unnecessary for the ITS 3.9 series of specifications because LCO 3.0.3 does not apply in MODES 5 and 6. This change is classified as administrative because the operating flexibility employed by the CTS 3.0.3 exception is inherent in the structure of the ITS.

A6 The CTS markup was annotated to show adoption of ITS LCO 3.9.4 Applicability. ITS LCO 3.9.4 is comparable to CTS 3.8.3.a. However, the CTS did not explicitly establish an Applicability for this Specification. This is considered an administrative change because the intended Applicability for the CTS was during refueling activities which corresponds to MODE 6 in the ITS. In addition, CTS 3.8.3.b established LCO requirements comparable to those stated by ITS 3.9.5 (i.e., DHR requirements when less than 23 feet of water covered the irradiated fuel). Because CTS 3.8.3.b established LCO requirements when the water level was less than 23 feet above the fuel, it is implied that CTS 3.8.3.a had an Applicability when the water level was greater than 23 feet above the fuel. Based on this reasoning, the adoption of the ITS 3.9.4 Applicability is administrative.

A7 ITS 3.9.5 Required Action A.2 is shown as being adopted on the CTS markup. This Required Action is an alternative to A.1 which requires restoration of the inoperable DHR loop. Required Action A.2 serves to remove the unit from the MODE of Applicability. This is cited as an Administrative change because this action (i.e., removing the unit from the Applicability) was available as an option in the CTS although not explicitly written as a Required Action. This change is consistent with NUREG-1430.

A8 CTS 3.8.3.a was annotated to show the explicit Completion Time of "immediately" for the ITS Required Actions that reference CTS 3.8.3.a. This is shown as an administrative adoption because the assigned Completion Time is consistent with other CTS required actions in this series of Specifications. This change is consistent with NUREG-1430.

A9 Not used.

A10 Not used.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 The CTS markup was annotated to show adoption of NUREG-1430 SR 3.9.5.2 (ITS SR 3.9.5.1) which requires verification of correct breaker alignment and indicated power availability to the required DHR pump that is not in operation with a Frequency of 7 days. This SR verifies the availability of the non-operating DHR loop required when the reactor coolant level is less than 23 feet above the top of the fuel seated in the reactor pressure vessel. The adoption of this ITS SR results in additional operational requirements or constraints beyond those imposed by the CTS. This change is consistent with NUREG-1430.

M2 Not used.

M3 The last paragraph of CTS 3.8.3 established the last of the required actions for CTS 3.8.3.a and 3.8.3.b. This paragraph is connected to the previous paragraphs with an "otherwise" which would imply this to be an alternative to the previous required actions. The CTS action established by this paragraph will be connected to the equivalent ITS Required Actions with an "and." This conjunction will eliminate the apparent alternative that is present in the CTS. Thus, the ITS Required Actions (3.9.4 RA A.3 and 3.9.5 RA A.1) that reference this specification will be more restrictive than the CTS. This change is consistent with NUREG-1430.

M4 CTS 3.8.4 established the requirement for minimum boron concentration during "reactor vessel head removal and while loading and unloading fuel from the reactor." The Applicability for ITS LCO 3.9.1 will be MODE 6. MODE 6 is entered with the detensioning of the first reactor vessel head stud and will be in effect as long as fuel is in the vessel until the last reactor vessel head stud is retensioned. Thus, the Applicability of ITS LCO 3.9.1 will be more inclusive and more restrictive than the requirements of the CTS because it includes the period of time associated with vessel head reinstallation. This change is consistent with NUREG-1430.

M5 The CTS markup was annotated to show the adoption of ITS LCO 3.9.2 Required Action B.2. ITS 3.9.2 Condition B establishes the Required Actions should both of the required source range neutron flux monitors become inoperable. Required Action B.1 is established by CTS 3.8.9. ITS 3.9.2 Required Action B.2 requires performance of SR 3.9.1.1 with a Completion Time of once per 12 hours. ITS SR 3.9.1.1 verifies that the boron concentration of the RCS, refueling canal and refueling cavity is within its limits. No comparable CTS required action exists. Therefore, through the adoption of ITS 3.9.2 Required Action B.2, the ITS will impose an additional restriction on the unit. The adoption of ITS 3.9.2 Required

Action B.2, in conjunction with the current requirements of ITS 3.9.2 Condition A and Required Action B.1, ensures that the core's reactivity condition is not changing during the period when no OPERABLE source range nuclear instrument is available for the detection of changes in core reactivity. This change is consistent with NUREG-1430.

M6 The CTS markup was annotated to show the adoption of ITS SR 3.9.2.1, SR 3.9.2.2 and the SR 3.9.2.2 Note. SR 3.9.2.1 established requirements for a CHANNEL CHECK every 12 hours. SR 3.9.2.2 established requirements that a CHANNEL CALIBRATION be performed every 18 months. The SR 3.9.2.2 Note excludes the neutron detectors from the CHANNEL CALIBRATION requirements because of the inability to calibrate these detectors. The ANO-1 CTS did not include similar surveillance requirements in this MODE of Applicability. Therefore, the ITS will impose additional restrictions on the unit. These SRs are necessary because they serve to demonstrate the functional capability of the source range nuclear instruments to respond to changes in core conditions. This change is consistent with NUREG-1430.

M7 The CTS markup was annotated to show adoption of ITS SR 3.9.3.2 and its associated Note. SR 3.9.3.2 requires verification that each required reactor building isolation valve and each reactor building purge isolation valve can actuate to the isolation position with a Frequency of 18 months. This SR demonstrates that each of the reactor building isolation valves are capable of being placed in its closed position. The 18 month surveillance Frequency is commensurate with the normal duration of an operating cycle. The SR Note is administrative in nature in that it establishes that the application of this SR requirement does not apply to valves that have been closed in accordance with ITS LCO 3.9.3.c.1. The CTS does not presently contain such a Surveillance Requirement. Thus, the adoption of this SR results in the ITS being more restrictive than the CTS. This change is consistent with the NUREG-1430.

M8 The CTS markup was annotated to show adoption of ITS SR 3.9.3.1. SR 3.9.3.1 requires verification that each required reactor building penetration is in the required status with a Frequency of 7 days. This SR demonstrates that each of the reactor building penetrations required to be in its closed position is in that position. The 7 day surveillance Frequency is commensurate with the normal duration of fuel handling activities during a refueling. The CTS does not presently contain such a Surveillance Requirement. Thus, the adoption of this SR results in the ITS being more restrictive than the CTS. This change is consistent with the NUREG-1430.

M9 CTS 3.8.10 established the LCO requirements for the reactor building purge isolation system. These requirements are comparable to the LCO requirements of NUREG-1430 3.9.3. However, the CTS does not establish specific required actions or associated completion times should the LCO not be satisfied. ITS 3.9.3 Condition A will establish the Required Actions and associated Completion Times for this LCO in the ITS. The Required Actions remove the unit from the LCO Applicability and eliminate the possibility of fuel handling accident during the period of the inoperable reactor building purge isolation valve(s). The CTS markup was annotated to show ITS 3.9.3 Action A as correlated to CTS 3.8.9 because its contains the intended ITS Actions. This really constitutes the adoption of the ITS Required Actions and Completion Times for Condition A when applied to CTS 3.8.10 LCO requirements. The imposition of the Actions for CTS 3.8.10 will establish additional restrictions that are not present in the CTS. The establishment of Required Actions and associated Completion Times for inoperability of the reactor building purge isolation valves is consistent with NUREG-1430.

M10 The CTS markup was annotated to show adoption of ITS SR 3.9.6.1. SR 3.9.6.1 requires verification that the refueling canal level is greater than or equal to 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. This SR demonstrates that the Fuel Handling Accident analysis initial condition assumptions regarding the refueling canal level are satisfied during the movement of irradiated fuel assemblies within the reactor building. The 24 hour surveillance Frequency is considered appropriate in view of the large volume of water and the normal procedural controls in place during fuel handling activities. The CTS does not presently contain such a Surveillance Requirement. Thus, the adoption of this SR results in the ITS being more restrictive than the CTS.

M11 Not used.

M12 CTS Table 4.1-3 is annotated to show the NUREG-1430 SR 3.9.1.1 Frequency of 72 hours. The adoption of the 72 hour Frequency reduces the degree of scheduling freedom present in CTS Table 4.1-3 Item 1.f, Boron Concentration, sampling frequency of 3 times per week. This CTS frequency does not stipulate that the samples obtained at approximately equal intervals. The ITS 72 hour Frequency imposes a more structured requirement with specific sampling intervals that are not as flexible as the CTS Frequency. The adoption of this Frequency establishes requirements that are consistent with NUREG-1430.

M13 CTS 3.8.10 is annotated to show its correlation to ITS SR 3.9.3.3 which specifies a Frequency of 18 months. The 18 month surveillance Frequency is consistent with the refueling frequency when this SR can be performed. Because the CTS established the Frequency based on a time commensurate with refueling activities, the imposition of a fixed 18 month increment will be more restrictive than CTS requirements. In addition, the CTS simply required that the radiation monitors be tested and verified to be OPERABLE. The ITS will specify that this is accomplished by a CHANNEL CALIBRATION. This change is consistent with the NUREG-1430.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 CTS 3.8.7 requires that isolation valves in lines containing automatic containment isolation valves be OPERABLE, or at least one shall be closed. ITS 3.9.3.c requires that each penetration providing direct access from the reactor building atmosphere to the outside atmosphere be 1) closed by a manual valve or automatic isolation valve, blind flange, or equivalent, or 2) be capable of being closed by an OPERABLE isolation valve. CTS 3.8.7 requires containment closure capability of components in fluid systems that are ordinarily incapable of releasing radioactive material from the reactor building atmosphere to the outside atmosphere because they are not exposed to the reactor building atmosphere (i.e. the system is intact). ITS 3.9.3 will only apply to those penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. Thus, the scope of the penetrations requiring closure by a manual or power operated isolation valve will be reduced. However, the reduction in scope of penetrations subject to the closure specification will not appreciably change the protective nature of the reactor building. This is because fluid systems that are not open to the reactor building atmosphere have never been a credible release path. Only those penetrations that allow reactor building atmosphere release to the environment are credible offsite dose contributors. Therefore, the reduction in the scope of reactor building penetrations requiring closure still results in the same level of protection for a member of the public. This change is consistent with NUREG-1430.

L2 CTS Table 4.1-3, Item 1.f required the determination of the RCS boron concentration with a Frequency of "3 times per week." The CTS did not establish that these samples were to be obtained on an equal interval. But if they were drawn at equal intervals, the interval would equate to three equal increments of 56 hours each. NUREG-1430 SR 3.9.1.1 specifies a Frequency of 72 hours. The ITS will retain the NUREG Frequency for this SR. This results in the SR being performed less frequently. The less frequent determination of the RCS boron concentration is acceptable based on: 1) administrative actions taken to prevent boron dilution events, 2) the relatively large inventory present during much of the time spent in MODE 6, and 3) historical experience associated with boron concentration changes during refueling conditions. This change is consistent with NUREG-1430.

L3 CTS 3.8.10 requires that the reactor building purge isolation valves "be tested and verified to be operable within 7 days prior to refueling operations." The ITS equivalent Surveillance Requirement is SR 3.9.3.2 which will have a Frequency of 18 months. This can be less restrictive than CTS requirements: 1) if refueling activities should occur on a more frequent or unexpected basis, or 2) if the SR is performed at a time other than refueling which would reestablish the SR interval such that it overlapped refueling activities; thus, avoiding the performance of this SR prior to the subsequent refueling activities. This change is consistent with NUREG-1430.

L4 CTS 3.6.2 established a requirement that reactor building integrity be maintained when the reactor coolant system (RCS) is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. When combined with the definition of a refueling shutdown (CTS 1.2.6), this establishes a conditional requirement that only exists when the RCS is open to the reactor building atmosphere and the degree of subcriticality is less than 1% ΔK/K assuming all rods are removed from the core. This reactivity condition is prohibited in the ITS through the imposition of a SHUTDOWN MARGIN requirement in MODE 5 (ITS 3.1.1) and imposition of a required degree of subcriticality ($K_{eff} \leq 0.99$) in MODE 6 (ITS 3.9.1). In both of these ITS Specifications, the Required Actions will be to restore the required SHUTDOWN MARGIN or degree of subcriticality, and while in MODE 6, terminate those activities that may result in the possibility of fission product release to the reactor building atmosphere or otherwise affect the core reactivity condition, for example, CORE ALTERATIONS. Thus, the ITS will be less restrictive than the CTS in that reactor building integrity will not have to be established as a direct result of a loss of SHUTDOWN MARGIN or degree of subcriticality. This change is acceptable because the ITS will direct actions to restore the required SHUTDOWN MARGIN or degree of subcriticality which are not present in the CTS. This change is consistent with NUREG-1430.

L5 CTS 3.8.3 established specific LCO requirements and explicit required actions for Decay Heat Removal. In addition, CTS 3.8.9 established a generic set of required actions for all of the preceding CTS 3.8 series of LCO requirements. CTS 3.8.3 directed that the operator "suspend all operations involving an increase in the reactor decay heat load." CTS 3.8.9 directed that "movement of the fuel into the reactor core shall cease." These actions correspond to ITS 3.9.4 Required Action A.2 which directs the operator to "suspend loading of irradiated fuel assemblies in the core." The ITS will be less restrictive than the CTS 3.8.9 requirements in that it would allow the continued introduction of non-irradiated fuel assemblies. ITS 3.9.4 Required Action A.2 is appropriate because it addresses the unavailability of a decay heat removal system to dissipate the decay heat being generated by the irradiated fuel assemblies within the reactor vessel. Non-irradiated fuel assemblies would not contribute to an increased decay heat load within the reactor vessel. This change is consistent with NUREG-1430.

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases or TRM. 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 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. The details of performance of the surveillances have been relocated to the TRM. Changes to the TRM will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
3.1.1.1.B	Bases, 3.9.4 & 3.9.5 LCO
3.8.2	TRM
3.8.6 Note *	Bases, 3.9.3, Background, LCO

LA2 CTS 3.8.11 is being relocated to the TRM. This Specification places restrictions on the removal of irradiated fuel from the reactor to ensure that sufficient time will elapse to allow the radioactive decay of short-lived fission products.

Although the Specification satisfied Criterion 2 of 10 CFR 50.36, the time to perform necessary activities prior to commencing movement of irradiated fuel ensures that there will normally be greater than 100 hours of subcriticality before any movement of irradiated fuel. Hence, the Specification is relocated in accordance with a prior industry/NRC agreement in the generic split report. Changes to the TRM are controlled under 10 CFR 50.59. This change is consistent with NUREG-1430.

3.9.4
3.9.5

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

⟨LATER⟩ (3.4A) 3.1.1 Operational Components — LATER

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical.

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. (LA1 Bases) — LATER

3.9.4 RA A.1 — With no ~~reactor coolant pumps or~~ decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system.

& ⟨LATER⟩ (3.4A)

⟨LATER⟩ (3.4A)

3.1.1.2 Steam Generator

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F. — LATER

⟨LATER⟩ (3.4B)

3.1.1.3 Pressurizer Safety Valves

A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable. — LATER

⟨LATER⟩ (3.4A)

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.2 are not applicable. — LATER

⟨LATER⟩ (3.4A)

3.1.1.5 Reactor Coolant Loops

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable: — LATER

3.6 REACTOR BUILDING

Applicability

Applies to the operability of the reactor building.

Objective

To assure reactor building operability.

<LATER> (3.6)

Specification

3.6.1 The reactor building shall be operable whenever all three (3) of the following conditions exist: —LATER

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable. —L4

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% Δk/k shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable.

<LATER> (3.6)

3.6.4 The reactor shall not be taken critical or remain critical if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches Hg. With the reactor critical, restore the containment pressure to within its limits within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours. —LATER

3.6.5 Prior to criticality following a refueling shutdown, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked, as required. The provisions of Specification 3.0.3 are not applicable.

3.9.1
3.9.2
3.9.4
3.9.5

<Add 3.9.5 RA A.2> A7

<Add SR 3.9.2.1> M6

<Add SR 3.9.2.2 with Note> M6

<Add 3.9.2 RA B.2> M5

<Add 3.9.4 Appl.> A6

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.

Specification

A1

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service. R TRM

3.9.2 LCO b. 3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, ~~each with continuous indication available,~~ whenever core geometry is being changed. ~~When core geometry is not being changed, at least~~ one neutron flux monitor shall be in service. LA1 TRM A1

3.9.2 Appl. MODE 6

3.9.2 LCO a.

3.9.4 LCO Note

3.9.4/3.9.5 LCO 3.8.3.a. ~~At least~~ one decay heat removal loop shall be in operation.* immediately A8

3.9.4 R.A. A.2 ~~Otherwise~~ suspend all operations involving an increase in the reactor decay heat load ~~or~~ a reduction in boron concentration of

3.9.4 RA A.1 the reactor coolant system ~~and~~ close all containment penetrations providing access from the containment atmosphere to the outside

3.9.4 RA A.4 atmosphere within 4 hours.

3.9.5 Appl. b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.~~**~~ LATER (3.8) M3

3.9.5 LCO

3.9.4 RA A.3

3.9.5 RA A.1 ~~Otherwise~~ immediately initiate corrective action to return the required loops to operable status ~~as soon as possible~~. immediately A1

MODE 6

3.9.1 Appl. 3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained

3.9.1 LCO at not less than that required ~~for refueling shutdown~~ in the COLR. M4

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place. R TRM

3.9.4 LCO Note *The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period ~~during the performance of core alterations~~. A4

<LATER (3.8)> ~~**The normal or emergency power source may be inoperable for each shutdown cooling loop.~~ LATER

provided no operations are permitted that would cause reduction in RCS boron concentration. A4

3.9.1
3.9.2
3.9.3
3.9.6

<ADD SR 3.9.3.2 with Note> M7
<ADD SR 3.9.6.1> M10
<ADD SR 3.9.3.1> M8

3.9.6 APPL
3.9.3 APPL
3.9.3 LCO b
3.9.3 LCO a
3.9.6 LCO

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be capable* of being closed. ~~The equipment hatch cover shall also be capable* of being closed.~~ At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel. LA1 Bases

3.9.3 LCO c.1
3.9.3 LCO c.2

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed. L1

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times. R TRM

3.9.5 RA A.1
3.9.4 RA A.1, A.2, A.3
3.9.1 Cond A
3.9.2 Cond A
3.9.2 RA B.1
3.9.6 Cond A
3.9.3 Cond A

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated immediately to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. ~~The provisions of Specification 3.0.3 are not applicable.~~ L5 A3 + R TRM A5 M9

3.9.3 LCO c.3
SR 3.9.3.3

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable ~~within 7 days prior to refueling operations.~~ Every 18 months ~~The provisions of Specification 3.0.3 are not applicable.~~ A5 L3 M13

3.8.11 ~~Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 100 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.~~ LA2 TRM

3.8.12 ~~All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

3.8.13 ~~No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

3.8.14 ~~Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

~~* Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed.~~ LA1 Bases (3.9.3)

3.9.1 3.9.4
3.9.2 3.9.5
3.9.3 3.9.6

<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

<LATER> (3.7 & 4.0)

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. — LATER

<LATER> (3.7)

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

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.

Bases

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) — (A2)

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

3.9.1
3.9.3
3.9.6

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.

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.

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.

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.

A2 + R TRM

<u>REFERENCES</u>

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

3.9.1

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
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 (3.4B)	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
<LATER> (3.1)	d. Dissolved Gases	d. Weekly (7)	LATER
<LATER> (3.4A)	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
SR 3.9.1.1	f. Boron Concentration	f. ~~3 times/week~~ 72 hours	M12, L2, & R TRM
LATER (3.4B)	g. Radiochemical Analysis for Ē 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	
<LATER> (3.7) 4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LATER
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)	R 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)

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 the gross radioactivity concentration exceeds 10% of the limit specified in 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. — LATER

3.9.4
3.9.5

4.27 DECAY HEAT REMOVAL

APPLICABILITY

Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal. —LATER

<LATER> (3.4A)

OBJECTIVE

To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.

SPECIFICATION

<LATER> (3.4A) — 4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability. —LATER

<LATER> (5.6) — 4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2. —LATER

4.27.3 The required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥20 inches on the startup range at least once per 12 hours. —LATER

<LATER> (3.4A) — 4.27.4 The required reactor coolant loop(s) shall be determined operable by verifying the required loop(s) to be in operation and circulating reactor coolant at least once per 12 hours.

SR 3.9.4.1 & <LATER> (3.4A) — 4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours. — & LATER

<Add SR 3.9.5.1> — M1

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:

NSHC 3.9 L1

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The reduction in scope of the number of reactor building penetrations requiring OPERABLE automatic isolation valves does not affect the postulated initiator for any evaluated MODE 6 accident. Therefore, no significant increase in probability of any previously evaluated accident will occur. Further, the reduction in scope of the number of reactor building penetrations requiring OPERABLE automatic isolation valves will not significantly increase the consequences of an evaluated accident. This is because these penetrations are associated with closed loop systems that did not have direct access to either the reactor building atmosphere or the outside atmosphere. Without assuming a failure which resulted in a break in these systems, these penetrations were not previously a credible pathway for the release of radioactivity to the outside atmosphere should a fuel handling accident have occurred. The assumption of this additional failure resulting in the breach of these systems is not consistent with the assumptions of the analysis. 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 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?**

An appropriate scope of applicability has been determined based on the safety analysis function that the reactor building penetrations maintain. The reduction in scope of the number of reactor building penetrations requiring OPERABLE automatic isolation valves does not result in a reduction in a margin of safety associated with any postulated MODE 6 accident. Because the leakage of radioactive materials via these penetrations was not previously credible, their exclusion from the ITS LCO requirements does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.9 L2

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate that the refueling canal boron concentration is within its limits. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. 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 the parameters 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 proposed change does not impact any physical mechanism or process that would allow the refueling canal boron concentration to change in an undetected manner such that any resultant increase in core reactivity would occur. Therefore, a change in the Surveillance Frequency does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.9 L3

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate the reactor building purge isolation valves are OPERABLE. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. 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 proposed change does not impact the mitigatory function of the reactor building isolation valves such that any resultant increase in offsite exposure would occur. Therefore, a change in the Surveillance Frequency does not involve a significant reduction in the margin of safety.

NSHC 3.9 L4

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

While in MODES 5 or 6, the elimination of the requirement to establish reactor building integrity, when the reactor coolant system is open to the reactor building atmosphere with the required degree of subcriticality specified for a refueling shutdown not met, will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS provides specific requirements for SHUTDOWN MARGIN (MODE 5) or degree of subcriticality (MODE 6). The ITS will establish Required Actions that initiate the restoration of the required SHUTDOWN MARGIN while in MODE 5. And while in MODE 6, ITS Required Actions will terminate activities that may result in the possibility of a Fuel Handling Accident which results in fission product release to the reactor building atmosphere, or otherwise affect the core reactivity condition through fuel loading errors or moderator dilution events. These ITS actions are the appropriate mitigatory actions to re-establish the initial conditions assumed in the analyses. Because these Required Actions re-establish the initial conditions assumed in the safety analyses, the consequences of a postulated event from this condition would not be significantly increased.

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 result in the ITS establishing the proper control of the required SHUTDOWN MARGIN (MODE 5) or required degree of subcriticality (MODE 6) 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 proposed change does not involve a significant reduction in the margin of safety. During MODE 5, existing margins of safety would be preserved through the ITS 3.1.1, SHUTDOWN MARGIN (SDM), Required Actions. During MODE 6, three possible events could be postulated. The three are the fuel handling accident, fuel loading error and moderator dilution. Reactor building closure requirements exist, independent of the subject of this change, that would maintain the reactor building's mitigatory function as previously assumed in the Fuel Handling Accident analysis. The fuel loading error event is not expected to occur due to stringent administrative controls; and should it occur, the event is expected to manifest itself during power operations. Specific administrative controls are in place to limit the source, rate and total quantity of dilution available. Because of the administrative controls, this event would occur at a slow rate with observable indications of the abnormal condition; thus, the operator could then initiate mitigatory measures.

NSHC 3.9 L5

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The elimination of the requirement to suspend the addition of non-irradiated fuel assemblies to the reactor core when a required decay heat removal (DHR) loop was inoperable will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS will establish Required Actions that: 1) suspend operations involving a reduction in the reactor coolant boron concentration thus preserving the necessary degree of subcriticality and mixing of the reactor coolant during dilution, 2) suspend the loading of irradiated fuel assemblies in the core thus stopping an increase in the decay heat magnitude present in the core, 3) initiate action to restore the required DHR loop to operation, and 4) provide closure of all reactor building penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. These actions are all consistent with the requirements of the CTS. ITS Required Actions will terminate activities that may result in increased levels of decay heat within the core, the possibility of a Fuel Handling Accident which results in fission product release to the reactor building atmosphere, or otherwise affect the core reactivity condition through a moderator dilution event. These ITS actions are the appropriate mitigatory actions to re-establish the initial conditions assumed in the analyses. Because these Required Actions re-establish the initial conditions assumed in the safety analyses, prevent the occurrence of evaluated events, and preserve the mitigatory response mechanisms should an event occur, the consequences of a postulated event from this condition would not be significantly increased.

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 result in the ITS establishing the proper control of refueling activities 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 proposed change does not involve a significant reduction in the margin of safety. The allowance to continue to add non-irradiated fuel bundles to the reactor core during a period in which the required DHR loop was inoperable does not result in an increase in the decay heat magnitude of the core. Thus, any margins present during the period when the required DHR loop was inoperable, would continue to be present with the non-irradiated fuel bundles present. ITS Required Actions will terminate activities that may result in the possibility of a Fuel Handling Accident which results in fission product release to the reactor building atmosphere, or otherwise affect the core reactivity condition through fuel loading errors or moderator dilution events. These ITS Actions preserve the appropriate mitigatory actions in response to the inoperability of the required decay heat removal loop.

1. NUREG 3.9.3, 3.9.4, and 3.9.5 - At numerous locations, the ITS and ITS Bases have been marked to reflect the ANO-1 unit specific terminology for its "reactor building;" rather than the NUREG-1430 term "containment." This change has been annotated at each occurrence in the ITS. This editorial change is made to retain conformity to the current license basis documents.

2. NUREG 3.9.3 Bases - An insert to the Bases for Specification 3.9.3 clarifies that a temporary equipment hatch that is securely held in place may satisfy the requirement that the equipment hatch be closed and held in place by four bolts. ANO-1 has a steel temporary equipment hatch for the purpose of providing a secure reactor building closure. This insertion clarifies that it is acceptable for ANO-1 to continue to use the temporary equipment hatch structure as provided in Unit 1 SAR, Section 5.2.2.1.3. This change is consistent with current license basis.

3. NUREG 3.9.3 - LCO 3.9.3.c.2 makes reference to "an OPERABLE Containment Purge and Exhaust Isolation System." Several aspects of this LCO require modification in order to reflect the ANO-1 purge system configuration and operational capability. CTS 3.8.7 allows the isolation valves for reactor building penetrations to be open provided an isolation valve associated with those penetrations is OPERABLE. Further, the reactor building purge isolation valves do not automatically isolate on high radiation levels. Valve closure must be manually initiated by the operator. Lastly, CTS 3.8.10 requires that the reactor building purge isolation valves, and the associated purge exhaust radiation monitor be OPERABLE. Therefore, ITS 3.9.3 was modified to reflect the current license requirements and system configuration. ITS 3.9.3.c.2 was modified to reflect that the penetration must be "capable of being closed by an OPERABLE reactor building isolation valve, except for reactor building purge isolation valves." This preserves the CTS 3.8.7 requirements. This change is consistent with current license basis.

   ITS 3.9.3.c.3 was inserted to require the reactor building purge isolation valves be capable of being closed and the associated purge exhaust radiation monitoring channel be OPERABLE. This LCO requirement preserves the CTS 3.8.10 requirements. This separate LCO requirement was provided to specifically differentiate the requirement for an OPERABLE purge exhaust radiation monitor for this penetration flowpath from the requirements for the ITS 3.9.3.c.2 penetration flowpaths. This change is consistent with current license basis.

   Because of the ANO-1 reactor building purge system configuration and the CTS 3.8.7 allowance incorporated into ITS 3.9.3.c.2, SR 3.9.3.2 was modified to remove reference to closure initiation "on an actual or simulated actuation signal." Valve closure would be as a result of operator initiated action. No automatic closure interlock based on high radioactivity levels exists for these isolation valves. In addition, no requirement for an OPERABLE Engineered Safeguards (ES) actuation capability exists in MODE 6 as stated in the Bases. SR 3.9.3.2 and its Note (Ref. DOD 4) were further modified to specifically include the ITS 3.9.3.c.3 reactor building isolation valves. This change does not alter the intent of the SR, which is to verify that the isolation valves will close when required to do so. This change is consistent with current license basis.

ITS SR 3.9.3.3 was added to require a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor with an 18 month Frequency. This SR presents the equivalent requirements of CTS 3.8.10. This SR ensures the OPERABILITY of the radiation monitoring instrumentation used to alert the operator of the need to isolate the reactor building purge release path. This change is consistent with current license basis.

The Bases description for ITS 3.9.3 required several modifications to reflect the LCO and SR changes. First, all reference to a "mini-purge" system was eliminated. ANO-1 has no such system. Second, reference to automatic isolation capability for the reactor building purge system penetrations was removed. These valves may be closed by an operator from the control room following receipt of indication that a high radiation level exists in the reactor building purge exhaust stream. No automatic closure interlock based on high radioactivity levels exists for these isolation valves. Third, all reference to ESAS functional capability was removed from the Bases supporting the OPERABILITY requirement while in MODE 6 (during Refueling) because these requirements are not pertinent. Fourth, the text was revised to reflect current license requirements that allow other reactor building penetrations to be open provided they are capable of being closed by an OPERABLE isolation valve. Lastly, the new LCO and SR requirements were incorporated into the Bases. These changes are consistent with current license basis.

4. NUREG 3.9.3 - Incorporates TSTF-092, Rev. 1.

5. NUREG 3.9.2 - Incorporates TSTF-096, Rev. 1.

6. NUREG 3.9.2 Bases - The Background section was modified to accurately describe the ANO-1 source range monitors. Gamma-Metrics fission chambers were installed as a post-TMI commitment that required environmentally qualified nuclear instrumentation. These instruments have superseded the original Bailey Model 880, BF_3 source range instrumentation. The Bases were modified to reflect that the fission chamber units are the principle nuclear instruments and that the original BF_3 instruments are not used for satisfying source range nuclear instrumentation monitoring requirements during shutdown conditions. This change is consistent with current license basis.

   As discussed in the first paragraph of the Background Section, portable source range instruments may be used to satisfy the LCO requirements. For clarification, the word portable was replaced with the word temporary. Although no change in intent exists with this change, it does eliminate the connotation that the substitute instrument would have to posses some degree of mobility. Further, temporary better describes the type of instrument that may be used in this context. To further clarify the usage of temporary source range monitors, an additional paragraph was inserted into the LCO that establishes the requirement that the temporary instrument be functionally equivalent to the installed instrumentation.

7. NUREG 3.9.4 - The Applicability will be MODE 6. This change preserves the Applicability established in general for the CTS 3.8 series of LCOs and implied by CTS 3.8.3.a. Implied because CTS 3.8.3.b addresses the decay heat removal requirements when the water level is less than 23 feet above the irradiated fuel seated in the reactor vessel. And, CTS 3.8.3.b is premised on already having a requirement for one DHR loop being OPERABLE and in operation. This Applicability preserves the large inventory requirement that is capable of providing decay heat removal for an extended period of time. This change is consistent with current license basis.

8. NUREG 3.9.5 - The Applicability for LCO 3.9.5 was changed to cover operation in MODE 6 with the water level less than 23 feet above the top of the irradiated fuel assemblies seated in the reactor pressure vessel. This change in Applicability replicates that established in CTS 3.8.3.b. Associated with the change in LCO 3.9.5 Applicability, Required Action A.2 was modified to provide consistent Actions for exiting the Applicability as one of the options available to the operator. This change is consistent with current license basis.

   The Bases were modified as necessary to reflect these changes.

9. NUREG 3.9.5 Bases - The Bases discussion for LCO 3.9.5 was modified by an inserted sentence that clarifies that the DHR loops may be considered OPERABLE when aligned to the Borated Water Storage Tank (BWST). This provision is necessary to support filling of the refueling canal or the performance of required testing of the DHR trains. Further, this clarification is necessary because of the explicit discussion in the LCO Bases of what constitutes a DHR flow path. This change to the Bases acknowledges these special operational conditions. This change is consistent with current license basis.

10. NUREG 3.9.3 - ITS 3.9.3.a and ITS 3.9.3.b were modified to reflect the CTS 3.8.6 requirements regarding the personnel and emergency air locks. The CTS requires that one door in each air lock be capable of being closed while moving irradiated fuel within the reactor building. The CTS also requires that the equipment hatch be capable of being closed while moving irradiated fuel within the reactor building. Associated with this requirement are administrative controls that ensure that personnel are capable of closing the airlock door and equipment hatch cover at the appropriate time. These administrative controls are discussed in the ITS 3.9.3 Bases. This change reflects current license basis.

11. NUREG 3.9.4 - SR 3.9.4.1 was modified to remove reference to a minimum decay heat removal system volumetric flowrate. The ANO-1 CTS does not establish a minimum flow requirement. The actual minimum flow rate is administratively controlled in operating procedures. Operation of the system is sufficient to ensure adequate mixing of the coolant to prevent boron stratification. Adequate heat removal is a function of a number of system parameters in addition to a minimum volumetric flowrate. As such, the operator has direct indication of the adequacy of the decay heat removal system in removing decay heat and adjustments would be made based on the trended indications. Although not done for this reason, this change establishes consistency between this SR and NUREG SR 3.9.5.1 and numerous ITS Section 3.4 SRs requiring verification of DHR operation. ANO-1 continues to employ administrative and procedural controls to ensure adequate DHR flow, which have been acceptable for operation under CTS. This change is consistent with current license basis.

    The Bases were modified as necessary to reflect this change.

12. NUREG 3.9.5 - LCO 3.9.5 was modified to remove the "one DHR loop shall be in operation" requirement because this duplicates the LCO requirement of ITS 3.9.4. ITS 3.9.5 Applicability of MODE 6 with the water level < 23 feet above the top of the irradiated fuel seated in the reactor pressure vessel is completely enveloped within the ITS 3.9.4 Applicability of MODE 6. And, ITS 3.9.4 already requires one DHR loop in operation, except as permitted by the LCO Note. This change maintains the current license basis requirements of CTS 3.8.3.

    NUREG 3.9.5 Condition B is shown as not incorporated in the ITS. Condition B is no longer necessary because it duplicates the Required Actions of ITS 3.9.4 Condition A. With the ITS 3.9.4 Applicability of MODE 6, the Required Actions for ITS 3.9.4 Condition A would be performed any time that a DHR loop was not in operation, except as permitted by the LCO Note. Thus, the Actions for ITS 3.9.5 need only address a condition in which the second required loop is inoperable. This change is consistent with current license requirements.

    The NUREG SR 3.9.5.1 requirement to verify one DHR loop in operation was deleted because it duplicates ITS SR 3.9.4.1. In addition, NUREG SR 3.9.5.1 is no longer necessary because the LCO 3.9.5 requirement for one DHR loop to be in operation was deleted, as discussed above. This change is consistent with current license requirements.

    The Bases were modified to reflect these changes.

13. NUREG 3.9.1 – Incorporates TSTF-214.

14. NUREG 3.9.2 - CTS 3.8.2 established the requirements for (source range) neutron flux monitoring in MODE 6. This Specification required one OPERABLE monitor when "core geometry is not being changed," and two OPERABLE monitors "whenever core geometry is being changed." NUREG-1430 has been modified to reflect these CTS requirements. ITS 3.9.2.a requires one source range neutron flux monitor be OPERABLE during the LCO Applicability (MODE 6). ITS 3.9.2.b requires one additional source range neutron flux monitor be OPERABLE during CORE ALTERATIONS. This change is consistent with current license basis.

    Condition A was modified to establish a Condition that was entered when one of the required source neutron flux monitors was inoperable "during CORE ALTERATIONS." NUREG-1430 Required Actions A.1 and A.2 replicate CTS 3.8.9 requirements for this entry Condition. With this change, the Condition A entry condition matches the Applicability of the ITS 3.9.2.b requirements and provides the appropriate Required Actions for this Condition.

    Condition B was modified to establish a Condition that is entered when there are no OPERABLE source range neutron flux monitors. Required Action B.1, in addition to Required Actions A.1 and A.2 if during CORE ALTERATIONS, replicates the required CTS 3.8.9 requirements for this condition.

    These changes maintain the requirements of the CTS while providing adequate monitoring capability of changes in the core's neutron flux. When core reactivity conditions are stable, i.e., no CORE ALTERATIONS are in progress, one neutron source range monitor is adequate. During the conditions when the core's reactivity condition is subject to change, i.e., during CORE ALTERATIONS, two monitors are required to provide independent and redundant monitoring capability of the reactivity changes in the core. This change is consistent with current license basis.

15. NUREG 3.9.3 - CTS 3.8.6 established the Applicability for reactor building closure penetrations as "during the handling of irradiated fuel in the reactor building." This CTS requirement will be retained as the Applicability for ITS 3.9.3. The NUREG Applicability of "during CORE ALTERATIONS" will not be adopted in the ITS. Retention of the CTS Applicability results in NUREG 3.9.3 Required Action A.1 not being adopted because it presents requirements that are inconsistent with the LCO Applicability. These changes maintain the current license requirements.

16. Not used.

17. NUREG 3.9.5 - SR 3.9.5.2 was revised to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating. The Bases are also revised to indicate that if a pump is verified to be in operation, this is also sufficient to verify the correct breaker alignment and indicated power availability. This change is consistent with changes made to NUREG LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8 - NUREG SR 3.4.5.2, SR 3.4.6.2, SR 3.4.7.3 and SR 3.4.8.2.

    The Bases are also revised to reflect this change.

18. Not used

19. NUREG 3.9.2 Bases - The Applicability discussion for ITS 3.9.2 covered MODES 2 through 6. An editorial change added a paragraph describing the lack of applicability in MODE 1. This editorial change preserves the unit specific configuration and functional capabilities and was made only for completeness. This change is consistent with current license basis.

20. Bases ITS 3.9.3 - Additional guidance on what constitutes a "direct access" path from the reactor building to the outside atmosphere was provided. This is intended to assist the operator in determining the scope of the LCO and assist in determining the acceptability of temporary closures. This avoids the need for future interpretation of what constitutes "direct access." This change preserves the interpretations allowed under the current license basis.

21. NUREG 3.9.6 - Incorporates TSTF-020.

22. NUREG 3.9.6 - The LCO was revised to reflect the CTS 3.8.6 requirement that the refueling canal level be maintained greater than or equal to 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. This change preserves the initial conditions of the ANO-1 Fuel Handling Accident in the reactor building. This change is consistent with current license basis as recently approved in Amendment 184.

    NUREG 3.9.6 Bases - The Applicable Safety Analyses discussion has been revised to describe the initial assumptions of the ANO-1 Fuel Handling Accident in the reactor building. This change maintains consistency with the ANO-1 license basis.

    CTS 3.8.6 defined the Applicability for the refueling canal water level requirements as "during the handling of irradiated fuel in the reactor building." This Applicability is preserved in ITS 3.9.6. The NUREG-1430 requirements of during CORE ALTERATIONS, except during latching and unlatching or CONTROL ROD drive shafts, is not adopted. The assumed initiator of the Fuel Handling Accident is the accidental drop of an irradiated fuel assembly with its subsequent fall to a horizontal position. Protective requirements for this assumed initiation condition are preserved by limiting the Applicability to "during movement of irradiated fuel assemblies ...." This change is consistent with current license basis.

NUREG 3.9.6 Required Action A.1 is not adopted because it established Required Actions contrary to the ITS 3.9.6 Applicability. NUREG Required Action A.2 (ITS 3.9.6 RA A.1) that requires the suspension of the movement of irradiated fuel assemblies is sufficient to prevent the occurrence of a Fuel Handling Accident should the refueling canal water level drop below 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel. This change is consistent with current license basis.

NUREG SR 3.9.6.1 was modified to reflect the LCO required level of greater than or equal to 23 feet above the top of irradiated fuel assemblies seated with the reactor pressure vessel. This change is consistent with current license basis.

23. NUREG Bases - ANO-1 was designed and licensed to the AEC's General Design Criteria (GDC) which was published in the Federal Register on July 11, 1967 [32FR10213]. Appendix A to 10 CFR 50 effective in 1971 [36FR3256] and subsequently amended, is somewhat different from the proposed 1967 criteria. SAR Section 1.4 includes an evaluation of ANO with respect to the 1967 criteria. The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the SAR.

24 NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

The 10 CFR 50.36 Criterion satisfied by the ITS LCOs was modified to preserve consistency with the ANO-1 license basis. The NUREG Criterion specified were modified to be consistent with the analysis assumptions regarding equipment availability and operating condition (i.e., MODE).

25 NUREG 3.9.1 Bases, 3.9.4 and 3.9.4 Bases - With the changes discussed in DOD-07, ITS 3.9.4 will apply at all times in MODE 6, regardless of the water level. Since the LCO now applies at all times in MODE 6, the title is misleading. Therefore, the titles is revised by the deletion of "-High Water Level." This change is made for consistency within the ITS.

26 NUREG 3.9.1 Bases - The NUREG states that the refueling boron concentration is intended to ensure an overall core reactivity of $K_{eff} \leq 0.95$ during fuel handling with control rods and fuel assemblies assumed to be in the most adverse condition. These statements have been modified to reflect the current ANO-1 license basis. The ANO-1 analysis assumptions for the boron dilution accident are based on an initial K_{eff} of ≤ 0.99. The Bases for CTS 3.8.4 states: "The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. Although the refueling boron concentration is sufficient to maintain the core $K_{eff} \leq 0.99$ if all control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The K_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9." Therefore, the required overall core reactivity has been changed from $K_{eff} \leq 0.95$ to $K_{eff} \leq 0.99$ for consistency with the current license basis.

CTS

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1	~~a.~~ Boron concentrations of the Reactor Coolant System, and the refueling canal, ~~and the refueling cavity~~ shall be maintained within the limit specified in the COLR. ~~b. Boron concentration shall not be reduced unless reactor coolant is circulating.~~ 13	edit 3.8.4 edit

APPLICABILITY: MODE 6. 3.8.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately	3.8.9
	AND		
	A.2 Suspend positive reactivity additions.	Immediately	3.8.9
	AND		
	A.3 Initiate action to restore boron concentration to within limit.	Immediately	3.8.9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours	Table 4.1-3 Item 1.f.

CTS

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

(14)

LCO 3.9.2 a. One ~~Two~~ source range neutron flux monitor~~s~~ shall be OPERABLE~~.~~ and 3.8.2
b. One additional source range neutron flux monitor shall 3.8.2
be OPERABLE during CORE ALTERATIONS. 3.8.2

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
A. One ~~(required)~~ source range neutron flux monitor inoperable~~.~~ during CORE ALTERATIONS	A.1	Suspend CORE ALTERATIONS.	Immediately	3.8.9
	AND			
	A.2	Suspend positive reactivity additions.	Immediately	3.8.9
				(14)
B. No OPERABLE ~~Two (required)~~ source range neutron flux monito~~rs inoperable~~.	B.1	Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately	3.8.9
	AND			
	B.2	Perform SR 3.9.1.1.	~~4 hours~~ ~~AND~~ Once per 12 hours ~~thereafter~~	N/A (5)

SURVEILLANCE REQUIREMENTS

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

CTS

3.9 REFUELING OPERATIONS (1)

3.9.3 ~~Containment~~ Reactor Building Penetrations

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

a. The equipment hatch is capable of being closed ~~and held in place by four bolts~~; 3.8.6 (10)

b. One door in each air lock closed; and 3.8.6

c. Each penetration providing direct access from the ~~containment~~ reactor building atmosphere to the outside atmosphere either: (1)

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or 3.8.7

2. capable of being closed by an OPERABLE ~~Containment~~ reactor building ~~Purge and Exhaust~~ Isolation ~~System~~ valve, except reactor building purge isolation valves, or 3.8.7 (3) 3.8.10

(1c)

<INSERT 3.9-4A> (15)

APPLICABILITY: ~~During CORE ALTERATIONS,~~ During movement of irradiated fuel assemblies within ~~containment~~ the reactor building. 3.8.6 (1)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~containment~~ reactor building penetrations not in required status.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~ (15) (1)
	~~AND~~	
	~~A.2~~ A.1 Suspend movement of irradiated fuel assemblies within ~~containment~~ the reactor building.	Immediately 3.8.9 (1)

<INSERT 3.9-4A>

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

(1)

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.3.1	Verify each required reactor building ~~containment~~ penetration is in the required status.	7 days	N/A
SR 3.9.3.2	Verify each required reactor building ~~containment purge and exhaust~~ isolation valve actuates to the isolation position ~~on an actual or simulated actuation signal~~ and each reactor building purge isolation valve	~~{18}~~ 18 months	N/A (3)

- - - - - - - NOTE - - - - - - - - - - - - - (4)

Not applicable to ~~containment purge and exhaust valve(s)~~ reactor building isolation valves and reactor building purge isolation valves in penetrations closed to comply with LCO C.1. (3)

INSERT

SR 3.9.3.3	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor.	18 months	3.8.10 (3)

CTS

3.9 REFUELING OPERATIONS

3.9.4 Decay Heat Removal (DHR) and Coolant Circulation ~~High Water Level~~ (25)

LCO 3.9.4 One DHR loop shall be OPERABLE and in operation. 3.8.3.a

----------NOTE----------
The required DHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration. 3.8.3.a Note *

APPLICABILITY: MODE 6 ~~with the water level ≥ 23 ft above the top of reactor vessel flange.~~ N/A (7)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. DHR loop requirements not met.	A.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately	3.8.3.a 3.1.1.1.B 3.8.9
	AND			
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately	3.8.3.a 3.8.9
	AND			
	A.3	Initiate action to satisfy DHR loop requirements.	Immediately	3.8.9 3.8.3.b (2ND ¶)
	AND			
			(continued)	

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close all ~~containment~~ reactor building penetrations providing direct access from ~~containment~~ the reactor building atmosphere to outside atmosphere.	4 hours

3.8.3.a (1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.4.1 Verify one DHR loop is in operation ~~and circulating reactor coolant at ≥ flow rate of ≥ [2800] gpm~~.	12 hours

4.27.5 (11)

CTS

3.9 REFUELING OPERATIONS

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

LCO 3.9.5 Two DHR loops shall be OPERABLE. ~~and one DHR loop shall be in operation.~~ 3.8.3.a 3.8.3.b 12

APPLICABILITY: MODE 6 with the water level < 23 ~~ft~~ feet above the top of ~~reactor vessel flange.~~ the irradiated fuel seated in the reactor pressure vessel. edit 3.8.3.b 8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Less than required number of DHR loops OPERABLE.	A.1 Initiate action to restore DHR loop to OPERABLE status.	Immediately	3.8.9 3.8.3.b (2nd ¶)
	OR		
	A.2 Initiate action to establish ≥ 23 ~~ft~~ feet of water above the top of ~~reactor vessel flange.~~ the irradiated fuel seated in the reactor pressure vessel.	Immediately	N/A edit 8
~~B. No DHR loop OPERABLE or in operation.~~	~~B.1 Suspend operations involving a reduction in reactor coolant boron concentration.~~	~~Immediately~~	12
	~~AND~~		
	~~B.2 Initiate action to restore one DHR loop to OPERABLE status and to operation.~~	~~Immediately~~	
	~~AND~~		
		~~(continued)~~	

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

(12)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one DHR loop is in operation.	12 hours
SR 3.9.5.~~2~~ 1	Verify correct breaker alignment and indicated power available to ~~the~~ each required DHR pump~~s~~ ~~that is not in operation~~.	7 days

(12)

N/A

(17)

CTS

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

feet

3.8.6

LCO 3.9.6 Refueling canal water level shall be maintained ≥ 23 ~~ft~~ feet above the top of ~~the reactor vessel flange.~~ irradiated fuel assemblies seated within the reactor pressure vessel. (22)

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts,~~ (22)
During movement of irradiated fuel assemblies within ~~containment.~~ the reactor building. (1)

3.8.6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	~~A.1 Suspend CORE ALTERATIONS.~~ (22)	~~Immediately~~
	~~AND~~	
	A.~~2~~ 1 Suspend movement of irradiated fuel assemblies within ~~containment.~~ the reactor building. (1)	Immediately
	~~AND~~	
	~~A.3 Initiate action to restore refueling cavity water level to within limit.~~ (21)	~~Immediately~~

edit 3.8.9

CTS

SURVEILLANCE REQUIREMENTS

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

N/A

22

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS)~~,~~ and the refueling canal~~, and the refueling cavity~~ during refueling ensures that the reactor remains subcritical during MODE 6. The Refueling boron concentration ~~is the soluble boron concentration in~~ specified for the coolant in each of these volumes ~~having~~ since each volume has direct access to the reactor core during refueling. edit edit edit edit

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit ~~is~~ specified in the COLR~~. Unit procedures~~ ensures ~~the specified boron concentration in order to maintain~~ an overall core reactivity of $k_{eff} \leq 0.9$~~8~~ (9) ~~during fuel handling~~, with all control rods ~~and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures~~ OUT. 26

SAR, Section 1.4, GDC 26 ~~of 10 CFR 50, Appendix A~~, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The ~~Chemical Addition~~ Makeup and Purification System ~~serves as the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration~~ has the ability to initiate and maintain a cold shutdown condition in the reactor. 23

INSERT From Applicable Safety Analyses (B 3.9-2)

~~The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the borated water storage tank into the open reactor vessel by gravity feeding or by the use of the Decay Heat Removal (DHR) System pumps.~~ edit

~~The pumping action~~ Operation of the ~~DHR~~ Decay Heat Removal (DHR) System in the RCS~~, and the natural circulation due to thermal driving heads in the reactor vessel and the refueling cavity,~~ mixes the added concentrated boric acid with the water in the refueling canal. The DHR System is in operation during refueling (see edit edit edit

(continued)

Decay Heat Removal (DHR)

BASES

BACKGROUND (continued)

LCO 3.9.4, "~~DHR~~ and Coolant Circulation ~~High Water Level~~," and LCO 3.9.5, "~~DHR~~ and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the and refueling canal ~~and the refueling cavity~~ above the COLR limit. 25 edit edit edit

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis ~~and is conservative for MODE 6~~. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance. edit

The required boron concentration and the unit refueling procedures ~~that demonstrate the correct fuel loading plan (including full core mapping)~~ ensure the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. ~~Hence, at least a 5% Δk/k margin of safety is established during refueling.~~ 9 26 edit

MOVE to BACKGROUND (B 3.9-1)

During refueling, ~~the water volume in~~ the spent fuel pool, the transfer ~~canal~~ tube, the refueling canal, ~~the refueling cavity~~, and the reactor vessel ~~form a single mass~~ are connected. As a result, the soluble boron concentration is relatively the same in each of these volumes. edit edit edit

The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 2). 24

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, and the refueling canal ~~and the refueling cavity~~ while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.9~~5~~ 9 is maintained during fuel handling operations with CONTROL RODS and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. edit edit 26 edit

~~This LCO also requires that coolant be circulated during any boron dilution. Providing forced coolant circulation during changes in boron concentration ensures mixing of the coolant, eliminating the potential for pockets of diluted, unmixed coolant, which may cause loss of required SDM.~~ 13

(continued)

BASES

LCO (continued)

~~Adequate mixing prevents stratification to ensure that dilution induced reactivity changes are gradual, as well as recognizable and controllable by the operator. Forced circulation will also ensure that the boron concentration determined by chemical analysis is representative of the entire coolant volume.~~ 13

Violation of the LCO ~~could lead to~~ provides a potential for an inadvertent criticality during MODE 6. edit

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. ~~The required boron concentration ensures a $k_{eff} \leq 0.95$.~~ edit

¶ (New paragraph) Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ~~and LCO 3.1.2, "Reactivity Balance,"~~ LCO 3.1.5, "Safety Rod Insertion Limits," and LCO 3.2.1, "Regulating Rod Insertion Limits," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical. edit

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of ~~any coolant volume~~ ~~in~~ or the RCS, the refueling canal, ~~or the refueling cavity~~ is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. edit edit

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS ~~or~~ and positive reactivity additions, action to restore the concentration must be initiated immediately. edit

~~In determining the required combination of boration flow rate and concentration,~~ There is no unique Design Basis edit edit

(continued)

BASES

ACTIONS

A.3 (continued)

analysis that requires a specific rate of boration

Event ~~that must be satisfied~~. The only requirement is to restore the boron concentration to its required value as soon as possible. ~~In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.~~ edit edit

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS, and the refueling canal ~~and the refueling cavity~~ is within the COLR limits. The boron concentration of the coolant in each volume is determined ~~periodically~~ every 72 hours by chemical analysis. edit edit edit

~~A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples.~~ The Frequency is based on industry ~~operating~~ experience, which has shown 72 hours to be adequate. edit edit

REFERENCES

1. ~~10 CFR 50, Appendix A~~ SAR, Section 1.4, GDC 26. 23

2. 10 CFR 50.36. 24

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System ~~(NIS)~~ (NI). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of ~~portable~~ temporary detectors is permitted, provided the LCO requirements are met. edit 6

The installed source range neutron flux monitors ~~are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve~~ Channels include fission chamber. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux ~~(1E6 cps) with a (5)% instrument accuracy~~. The ~~detectors~~ Instrumentation also provide continuous visual indication in the control room ~~and an audible alarm~~ to alert operators to a ~~possible dilution accident~~ significant change in neutron flux. The NIS System is designed in accordance with the criteria presented in Reference 1. ~~If used, portable detectors should be functionally equivalent to the installed NIS source range monitors.~~ edit 6 edit

APPLICABLE SAFETY ANALYSES

~~Two~~ AN OPERABLE source range neutron flux monitor~~s~~ ~~are~~ is required to provide ~~a signal~~ indication to alert the operator to unexpected changes in core reactivity, such as may be caused by a boron dilution accident or an improperly loaded fuel assembly (Ref. 1). The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that the ~~normally available SDM would not be lost,~~ reactor remains subcritical and there is ~~sufficient time for the operator to take corrective action.~~ The source range neutron flux monitors are not credited for boron dilution event mitigation in the safety analyses. edit edit edit edit edit

The source range neutron flux monitors satisfy Criterion ~~3~~ 4 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 24

(continued)

BASES (continued)

LCO <INSERT B 3.9-6A>

This LCO requires one ~~two~~ source range neutron flux monitors OPERABLE to ensure that ~~redundant~~ monitoring capability is available to detect changes in core reactivity. (14) (6)

<INSERT B 3.9-6B>

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, ~~these same installed~~ source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux." edit (19)

<INSERT B 3.9-6C>

ACTIONS

A.1 and A.2

With only one ~~(required)~~ source range neutron flux monitor OPERABLE, during CORE ALTERATIONS redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. edit (14)

B.1

With no ~~(required)~~ source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status or until the Applicability is exited. edit edit

B.2

With no ~~(required)~~ source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made in accordance with Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are restored to an OPERABLE status ~~OPERABLE~~. This stabilized condition is ~~determined~~ Verified by performing SR 3.9.1.1 to ensure that the required boron concentration exists. edit edit edit edit

(continued)

<INSERT B3.9-6A>

One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS. This additional requirement ensures redundant monitoring capability when positive reactivity changes are being made to the core.

<INSERT B3.9-6B>

The use of temporary detectors is permitted for purposes of complying with this LCO. If used, the temporary detectors should be functionally equivalent to the installed source range monitors and satisfy applicable Surveillance Requirements.

<INSERT B3.9-6C>

In MODE 1, the neutron flux level is above the indicated range of the monitors. Thus, they are no longer relied upon for reactivity or power level monitoring. Hence, there are no requirements on source range neutron flux monitors in MODE 1.

BASES

ACTIONS

B.2 (continued)

once per 12

The Completion Time of ~~4~~ hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. ~~The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified.~~ The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

5
edit

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

normally

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. ~~It is based on the assumption that the two indication channels should be consistent with core conditions.~~ Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

edit
edit

<Insert B3.9-7A>

also

edit

<Insert B 3.9-7B>

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified ~~similarly~~ for the same instruments in LCO 3.3.9.

edit

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every ~~[~~18~~]~~ months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear instrument is a complete check and re-adjustment of the channel~~s~~, from the pre-amplifier input to the indicator~~s~~. The 18 month Frequency is based on ~~the need to perform this Surveillance during the conditions that apply during a plant outage.~~ ~~Operating~~ industry experience, which has shown these components usually pass the Surveillance when performed at the ~~[~~18~~]~~ month Frequency.

edit
instrument
edit
edit
edit
edit

REFERENCES

1. ~~10 CFR 50, Appendix A~~ SAR, Section 1.4, GDC 13, GDC 26, GDC 28, and GDC 29.

23

2. ~~F~~SAR, Section ~~[ ]~~ 14.1.2.4.

edit

3. 10 CFR 50.36.

24

<INSERT B3.9-7A>

It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious.

<INSERT B3.9-7B>

When in MODE 6 with only one channel OPERABLE, a CHANNEL CHECK is still required. However, in this condition, a redundant source range instrument may not be available for comparison. The CHANNEL CHECK provides verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

B 3.9 REFUELING OPERATIONS

B 3.9.3 ~~Containment~~ Reactor Building Penetrations

1

BASES

15

BACKGROUND

During ~~CORE ALTERATIONS or~~ the movement of irradiated fuel assemblies within ~~containment with irradiated fuel in containment,~~ the reactor building, a release of fission product radioactivity within ~~containment~~ the reactor building will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, ~~this~~ the containment of fission products is accomplished by maintaining ~~containment~~ Reactor building OPERABLE as described in LCO 3.6.1, "~~Containment~~" "Reactor Building." In MODE 6, the potential for ~~containment~~ reactor building pressurization as a result of an accident is not likely; therefore, requirements to isolate ~~the containment~~ the reactor building from the outside atmosphere can be less stringent. ~~The LCO~~ In order to make this distinction, the penetration requirements are referred to as "~~containment~~ reactor building closure" rather than "~~containment~~ reactor building OPERABILITY." ~~Containment~~ Reactor building closure means that all potential ~~escape~~ direct release paths are closed or capable of being closed. Since there is no potential for ~~containment~~ significant reactor building pressurization, the Appendix J leakage criteria ~~and tests~~ are not required.

edit

edit

The ~~containment~~ reactor building serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the ~~containment~~ reactor building provides radiation shielding from the fission products that may be present in the ~~containment~~ reactor building atmosphere following accident conditions.

edit

edit

The ~~containment~~ reactor building equipment hatch, which is part of the ~~containment~~ reactor building pressure boundary, provides a means for moving large equipment and components into and out of ~~containment~~ the reactor building. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building, the equipment ~~hatch must be held in place by at least four bolts~~. ~~Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.~~

edit 15

edit

edit 10

2

Capable of being closed

<<INSERT B3.9-8A>>

The ~~containment~~ reactor building air locks, which are also part of the ~~containment~~ reactor building pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "~~Containment~~ Reactor Building Air Locks." Each air lock has a door at ~~both~~ each end~~s~~. The doors are normally interlocked to prevent simultaneous opening when ~~containment~~ the reactor building OPERABILITY is required~~d~~. During ~~periods of~~ unit shutdown

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edit 15

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(continued)

<INSERT B3.9-8A>

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch cover following a required evacuation of the reactor building, and that any obstruction(s) (e.g., cables and hoses) that could prevent closure of the equipment hatch cover be capable of being quickly removed (Ref. 1). Should a fuel handling accident occur in the reactor building, 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. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

BASES

OPERABILITY

BACKGROUND (continued)

when ~~containment~~ ~~closure~~ reactor building OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods ~~when frequent containment entry is necessary~~. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building, ~~containment closure is required; therefore,~~ the door interlock mechanism may remain disabled, ~~but one air lock door must always remain closed~~.

<INSERT B3.9-9A>

requires that one door in each air lock be capable of being closed.

The requirements on ~~containment~~ reactor building penetration closure ensure that a release of fission product radioactivity within the ~~containment~~ reactor building will be restricted from ~~escaping to the environment~~. ~~The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.~~

to within regulatory limits.

The Reactor Building ~~Containment~~ Purge ~~and~~ Exhaust System includes a supply ~~two subsystems. The normal subsystem includes a [42] inch purge penetration and a [42] inch exhaust penetration. The second subsystem, or minipurge system, includes an [8] inch purge penetration and an [8] inch exhaust penetration.~~ During MODES 1, 2, 3, and 4, the ~~two~~ valves in ~~each of~~ the ~~normal~~ supply ~~purge~~ and exhaust penetrations are secured in the closed position. ~~The two valves in each of the two minipurge penetrations can be opened intermittently but are closed automatically by the Engineered Safety Feature Actuation System (ESFAS).~~ ~~Neither of~~ the ~~subsystems~~ system is not subject to a Specification in MODE 5.

Temperature control ~~In MODE 6, large air exchangers are necessary to conduct refueling operations.~~ The ~~normal [42] inch~~ purge system is ~~used for this purpose~~, and all four valves may be ~~are~~ closed ~~on~~ by an operator based on an indication of ~~reactor building (RB) high radiation signal in accordance with LCO 3.3.15, "Reactor Building (RB) Purge Isolation High Radiation."~~

<INSERT B3.9-9B>

~~The~~ Other ~~containment~~ the reactor building penetrations that provide direct access from ~~containment~~ reactor building atmosphere to outside atmosphere must be isolated on at least one side. ~~Isolation may be achieved by~~ or capable of being isolated by an OPERABLE ~~automatic~~ isolation valve, ~~or~~ by a closed manual or automatic isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other ~~containment~~ reactor building penetrations during fuel movements. (Ref. X)

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(3)
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(3)

(continued)

<INSERT B3.9-9A>

During the movement of irradiated fuel assemblies within the reactor building, administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors are open, that a specific individual(s) is designated and available to close an airlock door following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door be capable of being quickly removed (Ref. 3). Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency air lock doors will be closed following evacuation of the reactor building.

<INSERT B3.9-9B>

This LCO requires that an OPERABLE radiation monitor be present on the purge exhaust flow path to provide the necessary indication to the operator.

 1

BASES (continued) 15

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS or~~ movement of the irradiated fuel assemblies within ~~containment~~ the reactor building ~~with irradiated fuel in containment~~, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel. (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Canal Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in 10 CFR 100. The acceptance limits for offsite radiation exposure are contained in Reference ~~2~~ 4. 1 edit edit edit edit edit edit

4 a

Reactor building ~~Containment~~ penetrations satisfy Criterion ~~3~~ 4 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 5). 24

LCO

This LCO limits the consequences of a fuel handling accident in ~~containment~~ the reactor building by limiting the potential escape paths for fission product radioactivity from ~~containment~~ the reactor building. The LCO requires any penetration providing direct access from the ~~containment~~ reactor building atmosphere to the outside atmosphere to be closed ~~except for the~~ or capable of being closed by an OPERABLE ~~containment purge and exhaust~~ isolation valve. ~~penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the RB purge isolation signal.~~ 2 3

<< INSERT B3.9-10C >>

¶ << INSERT B 3.9-10A >> ~~The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated such that radiological doses are within the acceptance limit.~~ 3 10 20

¶ << INSERT B 3.9-10B >>

APPLICABILITY

The ~~containment~~ reactor building penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, ~~containment~~ penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within ~~containment~~ 15

(continued)

<INSERT B3.9-10A>

The reactor building personnel airlock doors and/or the equipment hatch may be open during movement of irradiated fuel in the reactor building provided that one door is capable of being closed in the event of a fuel handling accident. Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, that a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed (Ref. 1 and 3). For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. During outages, a temporary equipment hatch cover may be used in lieu of the permanent equipment hatch cover (Ref. 2).

<INSERT B3.9-10B>

The definition of "direct access from the reactor building atmosphere to the outside atmosphere" is any path that would allow for the transport of reactor building atmosphere to any atmosphere located outside of the reactor building structure. This includes the Auxiliary Building. As a general rule, closed systems do not constitute a direct path between the reactor building and the outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomena should not be postulated as part of the evaluation process.

<INSERT B3.9-10C>

This LCO requires the reactor building purge isolation valves and the purge exhaust flow path radiation monitor be OPERABLE.

edit

1 edit

BASES

APPLICABILITY (continued)

is ~~are~~ not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on ~~containment~~ reactor building penetration status.

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ACTIONS

A.1 ~~and A.2~~

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With the ~~containment~~ reactor building equipment hatch, air locks, or any ~~containment~~ reactor building penetration that provides direct access from the ~~containment~~ reactor building atmosphere to the outside atmosphere not in the required status, ~~including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open,~~ the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies within ~~containment~~ the reactor building. Performance of ~~these~~ this action~~s~~ shall not preclude moving a component to a safe position.

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<<INSERT B 3.9-11A>>

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the ~~containment~~ reactor building penetrations required to be in its closed position is in that position. ~~The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic RB purge isolation signal.~~

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The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ the movement of irradiated fuel assemblies within the ~~containment~~ reactor building. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. ~~A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO.~~

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edit

~~As such,~~ This Surveillance ensures that a postulated fuel handling accident that releases fission product

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(continued)

<INSERT B3.9-11A>

These actions remove the potential for an event which may require reactor building closure to prevent a significant radioactivity release.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

radioactivity within the ~~containment~~ reactor building will not result in a release of fission product radioactivity to the environment in excess of that recommended by Standard Review Plan Section 15.7.4 (Ref. 1, 3 and 6).

edit

SR 3.9.3.2

This Surveillance demonstrates that each ~~containment purge and exhaust~~ reactor building isolation valve actuates to its isolation position on manual initiation ~~or on an actual or simulated high radiation signal~~. The 18 month Frequency maintains consistency with other similar ~~ESFAS instrumentation and valve testing requirements~~ reactor building isolation found in Section 3.6. ~~In LCO 3.3.15, "RB Purge Isolation High Radiation," the isolation instrumentation requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.~~ ~~These~~ This Surveillance~~s~~ ~~performed during MODE 6~~ will ensure that the isolation valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the ~~containment~~.

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<INSERT B3.9-12A>

3

<INSERT B3.9-12B>

REFERENCES

~~1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~

1. Safety Evaluation Report related to ANO-1 Amendment No. 195, April 16, 1999.

~~2. FSAR, Section [ ].~~

~~3.~~ 6. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

4. SAR, Section 14.2.2.3.

5. 10 CFR 50.36.

2. SAR, Section 5.2.2.1.3.

3. Safety Evaluation Report related to ANO-1 Amendment No. 184, September 20, 1996.

edit

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edit

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<INSERT B3.9-12A>

The SR is modified by a Note stating that this demonstration is not applicable to valves in isolated penetrations. LCO 3.9.3.c.1 provides the option to close penetrations in lieu of requiring isolation capability.

<INSERT B 3.9-12B>

SR 3.9.3.3

This SR requires a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor. The CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION is performed consistent with the setpoint requirements. The 18 month Frequency is based on operating experience and is consistent with the typical operating cycle.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Decay Heat Removal (DHR) and Coolant Circulation ~~High Water Level~~ 25

BASES 23

BACKGROUND

The purposes of the DHR System (Ref. 1) ~~in MODE 6~~ are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, and to provide mixing of ~~borated~~ coolant, ~~to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and~~ to prevent boron stratification (Ref 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the ~~Component Cooling~~ Service Water System ~~via the DHR heat exchanger(s)~~. The coolant is then returned to the ~~RCS via the RCS cold legs~~ reactor vessel via the core flood tank injection nozzles. Operation of the DHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s) ~~and~~ ~~the reactor bypassing the heat exchanger(s)~~ and throttling of Service Water through the heat exchanger(s). Mixing of the reactor coolant is ~~maintained~~ provided by ~~this~~ the continuous ~~circulation~~ operation of ~~reactor coolant through~~ the DHR System.

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APPLICABLE SAFETY ANALYSES

<INSERT B3.9-13A>

~~If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction in boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the DHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not diluted. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.~~

edit

(continued)

<INSERT B3.9-13A>

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. The LCO does permit de-energizing the DHR pump for short durations under the condition that the boron concentration is not reduced. This conditional de-energizing of the DHR pump does not result in a challenge to the fission product barrier.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Satisfies Criterion 4 of 10CFR50.36 (Ref. 3). 24

~~Although the DHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the DHR System is retained as a Specification.~~

LCO

Only one DHR loop is required for decay heat removal in MODE 6~~, with a water level ≥ 23 ft above the top of the reactor vessel flange. Only one DHR loop is required to be OPERABLE because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one DHR loop must be OPERABLE and in operation to provide:~~ The operating DHR loop provides: EDIT. 25 7 EDIT.

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

To be considered ~~An~~ a OPERABLE~~,~~ DHR loop includes a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the ~~low end~~ temperature. The flow path starts in ~~one of~~ the ~~RCS~~ hot leg~~s~~ and is 'A' returned to the ~~RCS cold legs~~. reactor vessel via the core flood tank injection nozzles. EDIT. EDIT. EDIT. EDIT.

Additionally, each DHR loop ~~is considered OPERABLE if it can be~~ must be capable of being manually aligned (remote or local) in the ~~shutdown cooling~~ mode ~~for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.~~ decay heat removal to be considered OPERABLE. EDIT. EDIT. EDIT.

The LCO is modified by a Note that allows the required DHR loop to be removed from operation for up to 1 hour in an 8 hour period, provided no operation that would cause reduction of the RCS boron concentration is in progress. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allowance permits operations such as core mapping~~,~~ or ~~hot leg~~ ~~or~~ alterations in the vicinity of the reactor vessel nozzles and RCS to DHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling ~~cavity~~ canal. edit edit edit

(continued)

BASES (continued)

APPLICABILITY

One DHR loop must be OPERABLE and in operation in MODE 6, ~~with the water level > 23 ft above the top of the reactor vessel flange,~~ to provide decay heat removal. ~~The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Canal Water Level."~~ Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6, with the water level < 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel, are located in LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level."

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7

EDIT.

ACTIONS

~~DHR loop requirements are met by having one DHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.~~

EDIT.

A.1

If DHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations can occur by adding water with a lower boron concentration than that contained in the RCS.~~ Therefore, actions that reduce boron concentration shall be suspended immediately.

EDIT

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the ~~core~~ Canal. A minimum refueling water level 23 ~~ft~~ feet above the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading ~~a~~ an irradiated fuel assembly, is prudent under this condition.

7

EDIT.

(continued)

BASES

ACTIONS (continued)

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

edit

<< INSERT B 3.9-16A >> —>

A.4

If DHR loop requirements are not met, all ~~containment~~ reactor building penetrations providing direct access from the ~~containment~~ atmosphere to outside atmosphere shall be closed within 4 hours.

edit

<< INSERT B 3.9-16B >> —>

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. ~~The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core.~~ The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the DHR System. 11

<INSERT B 3.9-16C>>

edit

edit

REFERENCES

2 ~~1~~. ~~F~~SAR, Section ~~[ ]~~ 9.5

3. 10 CFR 50.36. 24

1. SAR, Section 1.4. 23

<INSERT B3.9-16A>

Restoration of one decay heat removal loop is required because this is the only active method of removing decay heat. Dissipation of decay heat through natural convection to the large inventory of water in the refueling canal should not be relied upon for an extended period of time. The immediate Completion Time reflects the importance of restoring an adequate decay heat removal loop.

<INSERT B3.9-16B>

If no means of decay heat removal can be restored, the core decay heat could raise temperatures and cause boiling in the core which could result in increased levels of radioactivity in the reactor building atmosphere. Closure of the penetrations providing access to the outside atmosphere will prevent the uncontrolled release of radioactivity to the environment.

<INSERT B3.9-16C>

Verification includes flow rate, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal.

B 3.9 REFUELING OPERATIONS

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

BASES

23

BACKGROUND

(Ref. 1) and

The purposes of the DHR System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of ~~borated~~ coolant, ~~to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and~~ to prevent boron stratification (Ref. 2). Heat is removed from the RCS by circulating reactor coolant through the DHR heat exchanger(s), where the heat is transferred to the ~~Component Cooling~~ Water System ~~via the DHR heat exchanger~~. The coolant is then returned to the ~~RCS~~ via the ~~RCS cold leg(s)~~. Operation of the DHR System for normal cooldown/decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by control of the flow of reactor coolant through the DHR heat exchanger(s) ~~and~~ bypassing the heat exchanger(s). Mixing of the reactor coolant is ~~maintained~~ by ~~this~~ continuous ~~circulation~~ of ~~reactor coolant through~~ the DHR System.

the reactor

Service

reactor vessel

core flood tank injection nozzles.

and by throttling of Service Water through the heat exchanger(s).

provided

the

operation

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edit

edit

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edit

APPLICABLE SAFETY ANALYSES

~~If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the DHR System are required to be OPERABLE, and one is required to be in operation, to prevent this challenge.~~

<INSERT B 3.9-17A>

edit

Satisfies Criterion 4

24

~~Although~~ the DHR System ~~does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk~~

10CFR50.36 (Ref. 3),

(continued)

<INSERT B3.9-17A>

Without a DHR loop in operation, the reactor coolant temperature may not be maintained below the boiling point. This could lead to inadequate cooling of the reactor fuel as a result of a loss of coolant in the reactor vessel due to boiling. The loss of reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Operation of one train of the DHR System in MODE 6 is sufficient to prevent this challenge. However, without a large water inventory to provide a backup means of decay heat removal, an additional train of the DHR System is required to be OPERABLE in order to provide a backup

BASES

~~APPLICABLE SAFETY ANALYSES (continued)~~ ~~reduction. Therefore, the DHR System is retained as a Specification.~~ edit

LCO

In MODE 6, with the water level < 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel (corresponds to approximately 390 feet above sea level), two DHR loops must be OPERABLE. Additionally, as required by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation", one DHR loop must be in operation to provide: edit 8 12

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

~~An~~ a OPERABLE DHR loop ~~consists~~ must consist of a DHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the ~~low end~~ 'A' temperature. The flow path starts in ~~one of~~ the ~~RCS~~ hot leg~~s~~ and is returned to the ~~RCS cold legs~~ reactor vessel via the core flood tank injection nozzles. edit edit edit 9

To be considered

<INSERT B3.9-18a> —>>

APPLICABILITY

Two DHR loops are required to be OPERABLE, and one in operation in MODE 6, with the water level < 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel, (corresponds to approximately 390 feet above sea level), to provide decay heat removal. Requirements for the DHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). DHR loop requirements in MODE 6~~, with the water level ≥ 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level."~~ edit 8 edit

ACTIONS

A.1 and A.2

With fewer than the required loops OPERABLE, action shall be immediately initiated and continued until the DHR loop is restored to OPERABLE status or until ≥ 23 ~~ft~~ feet of water level is established above the ~~reactor vessel flange~~ fuel assemblies seated in the reactor vessel. When the water level is established at ≥ 23 ~~ft~~ feet above the ~~reactor~~ edit 8 edit

(continued)

<INSERT B3.9-18A>

Additionally, to be considered OPERABLE, each DHR loop must be capable of being manually aligned (remote or local) in the decay heat removal mode for removal of decay heat. Operation of one subsystem can maintain the reactor coolant temperature as required.

Both DHR pumps may be aligned to the Borated Water Storage Tank (BWST) to support filling of the refueling canal or the performance of required testing.

BASES

ACTIONS

A.1 and A.2 (continued)

~~vessel flange~~, the Applicability will change to that of LCO 3.9.4, and only one DHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary ~~for an operator to initiate corrective actions to restore the required forced circulation or water level.~~

8

edit

due to the increased risk of operating without a large available heat sink for decay heat removal

~~B.1~~

~~If no DHR loop is in operation or no DHR loop is OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentration can occur by adding water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.~~

~~B.2~~

~~If no DHR loop is in operation or no DHR loop is OPERABLE, actions shall be initiated immediately and continued without interruption to restore one DHR loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE DHR loops and one operating DHR loop should be accomplished expeditiously.~~

~~If no DHR loop is OPERABLE or in operation, alternate actions shall have been initiated immediately under Condition A to establish ≥ 23 ft of water above the top of the reactor vessel flange. Furthermore, when the LCO cannot be fulfilled, alternate decay heat removal methods, as specified in the unit's Abnormal and Emergency Operating Procedures, should be implemented. This includes decay heat removal using the charging or safety injection pumps through the Chemical and Volume Control System with consideration for the boron concentration. The method used to remove decay heat should be the most prudent as well as the safest choice, based upon unit conditions. The choice could be different if the reactor vessel head is in place rather than removed.~~

12

(continued)

BASES

ACTIONS (continued)

~~B.3~~

~~If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.~~

~~The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.~~ (12)

SURVEILLANCE REQUIREMENTS

~~SR 3.9.5.1~~

~~This Surveillance demonstrates that one DHR loop is in operation. The flow rate is determined by the flow rate necessary to provide efficient decay heat removal capability and to prevent thermal and boron stratification in the core.~~

~~In addition, during operation of the DHR loop with the water level in the vicinity of the reactor vessel nozzles, the DHR loop flow rate determination must also consider the DHR pump suction requirement. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the DHR System in the control room.~~ (12)

SR 3.9.5.~~2~~ 1

edit (17)

Verification that ~~the~~ each required pump is ~~OPERABLE~~ available ensures that an additional DHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. (17)

<INSERT B3.9-20A> The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

(23) edit

REFERENCES

1. SAR, Section 1.4.
2. ~~F~~SAR, Section ~~[ ]~~ 9.5.
3. 10 CFR 50.36.

(24)

<INSERT B3.9-20A>

Alternatively, verification that a DHR pump is in operation as required by SR 3.9.4.1 also verifies proper breaker alignment and power availability.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies ~~or performance of CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts,~~ within ~~containment~~ requires a minimum water level of 23 ~~ft~~ above the top of the ~~reactor vessel flange~~. ~~During refueling, this maintains sufficient water level in the containment, the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary~~ to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident within 10 CFR 100 limits, as provided by the guidance of Reference 3. 22 1 22

the reactor building

irradiated fuel assemblies seated within the reactor pressure vessel.

feet

edit

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS and during~~ movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in ~~containment~~ postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ~~ft~~ (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) 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 refueling cavity water. The fuel pellet to cladding gap is assumed to contain ~~10%~~ 12 of the total fuel rod iodine inventory (Ref. ~~1~~ 2). 22 1 22 22

the reactor building

feet

above the top of the irradiated fuel assemblies seated within the reactor pressure vessel

The fuel handling accident analysis inside ~~containment~~ the reactor building is described in Reference 2. With a minimum water level of 23 ~~ft~~, and a minimum decay time of ~~72~~ hours prior to fuel handling, the analysis ~~and test programs~~ 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. 3). 1 22 22

100

feet

Refueling canal water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 4). 24

(continued)

BASES (continued)

edit

LCO

A minimum refueling ~~cavity~~ canal water level of 23 ~~ft~~ feet above the ~~reactor vessel flange~~ top of the irradiated fuel assemblies seated in the reactor pressure vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside ~~containment~~ the reactor building are within acceptable limits as provided by 10 CFR 100. 22 1

APPLICABILITY

LCO 3.9.6 is applicable during ~~CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts, and when moving~~ movement of irradiated fuel assemblies within the ~~containment~~ the reactor building. The LCO minimizes the possibility of a fuel handling accident in ~~containment~~ the reactor building that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in ~~containment~~ the reactor building, there can be no significant radioactivity release as a result of a postulated fuel handling accident~~s~~ in. ~~Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."~~ 22 1

edit

ACTIONS

A.1 ~~and A.2~~

With a water level of < 23 ~~ft~~ feet above the top of the ~~reactor vessel flange~~ irradiated fuel assemblies seated with the reactor pressure vessel, all operations involving ~~CORE ALTERATIONS or~~ the movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur. 22

The suspension of ~~CORE ALTERATIONS and~~ irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

~~A.2~~

~~In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.~~ 21

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ~~ft~~ feet above the top of the reactor pressure vessel ~~flange~~ ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel ~~flange~~ limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside ~~containment~~ the reactor building (Ref. 2).

irradiated fuel assemblies seated within the

edit

22

1

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls ~~of valve positions,~~ which make significant unplanned level changes unlikely.

edit.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. ~~F~~SAR Section ~~[ ]~~. 14.2.2.3.
3. 10 CFR 100.10.
4. 10 CFR 50.36.

edit.

24

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
4.1	4.1	Site Location
4.2	4.2	Reactor Core
4.3	4.3	Fuel Storage

4.0 DESIGN FEATURES

4.1 Site Location

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

4.0 DESIGN FEATURES

4.2 Reactor Core

4.2.1 Fuel Assemblies

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

4.2.2 Control Assemblies

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

DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

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

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

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties;

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

d. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure 3.7.14-1 allowed unrestricted storage in either fuel storage rack Region 1 or Region 2; and

e. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure 3.7.14-1 stored in Region 1, or in checkerboard configuration in Region 2.

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

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

b. $k_{eff} \leq 0.95$ under normal conditions;

c. $k_{eff} \leq 0.98$ with optimum moderation;

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

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

<-----NORTH

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

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

Figure 4.3.1.2-1 (page 1 of 1)

Fresh Fuel Storage Rack
Loading Pattern

CTS DISCUSSION OF CHANGES

ITS Section 4.0: Design Features

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 Babcock and Wilcox (B&W) revised 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 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The "less than" requirements for k_{eff} ,in CTS 5.4.1.1, have been revised to $\leq$ in ITS 4.3.1.2. These are considered to be essentially equivalent since the parameter can be less than than the limit, but be so close as to be imperceptible. This change is consistent with design basis and with NUREG-1430.

A4 The statement regarding the applicability of the provisions of Specification 3.0.3 is not retained. This statement is no longer required since the Specification is moved to the Design Features section for which LCO 3.0.3 is not applicable. Since there is no change in the application of the requirements, this change is considered administrative.

A5 Not used.

A6 Not used.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS 5.4.2 is revised to include additional information to describe the nominal center to center distance between fuel assemblies placed in the spent fuel storage racks. This change provides a safe geometric spacing in the spent fuel storage racks. There are only high density spent fuel storage racks provided at ANO-1 as discussed in SAR Section 9.6.2.3. Therefore, there is no need to differentiate between high density and low density racks in ITS 4.3.1, nor to provide any information on low density storage racks pursuant to RSTS 4.3.1:1.d. This change is consistent with RSTS 4.3.1.1.c.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- LESS RESTRICTIVE

L None

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to a licensee controlled document such as the Technical Requirements Manual (TRM), Safety Analysis Report (SAR), etc. This information provides details of the method of implementation which are not directly pertinent to the actual requirement. 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 details relocated to the TRM will be controlled by 10 CFR 50.59. The details relocated to the SAR will be controlled by 10 CFR 50.59 and 50.71. This change is consistent with NUREG-1430.

CTS Location	New Location
5.1	SAR 1.2.1
5.1	SAR 2.2
5.2.1	SAR 5.2.1
5.2.1	SAR 14.2.2.5.5
5.2.2	SAR 5.2.5
5.2.3	SAR 6.5
5.3.1.2	SAR Table 3-2
5.3.1.2	SAR 3.2.2.1.1
5.3.1.3	SAR Table 3-2
5.3.1.4	SAR 3.2.1
5.3.1.4	SAR Fig. 3-60
5.3.1.4	SAR 3A.3
5.3.1.4	SAR Fig. 3A-4
5.3.1.5	SAR 3.2.4.2
5.3.1.5	SAR Fig. 3-2
5.3.1.6	TRM
5.3.2.1	SAR 4.1.3
5.3.2.2	SAR 4.1.2
5.3.2.3	TRM
5.4.1.1	TRM
5.4.1.2	SAR 9.6.2.1
5.4.2.2	SAR 5.1.2.1.2

4.3.1.1.a 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. A4 LATER

+<LATER> (3.7)

4.3.1.1.d 4.3.1.1.e 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. A4 +LATER

<LATER> (3.7)

<LATER> (3.7) 3.8.17 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million. LATER

<LATER> (3.7) 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. LATER

Bases

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.

A2

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 $\leq$ 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

4.1

5.0 DESIGN FEATURES

(A1)

Specifications for design features are intended to cover characteristics of importance to each of the physical barriers, and to the maintenance of safety margins in the design.

5.1 SITE

Applicability

Applies to the location and extent of the exclusion area.

Objective

To define the location and the size of the site area as pertains to safety.

Specification

(LA1) SAR

4.1 Arkansas Nuclear One-Unit 1 is located on a site consisting of approximately 1100 acres which provides for 0.65 statute mile exclusion radius from the reactor building. This exclusion area includes certain portions of the bed and banks of the Dardanelle Reservoir which are owned by the Federal Government. An easement authorizes exclusion of all persons from these areas during periods of emergency. The site is approximately 6 statute miles WNW from the City of Russellville (Latitude 35°-18'-36" N, Longitude 93°-13'-53" W) in an area characterized by remoteness from population centers.

REFERENCE

FSAR, Section 2.2

4.0

LA1
SAR

5.2 REACTOR BUILDING

Applicability

Applies to those design features of the reactor building relating to operational and public safety.

Objective

To define the significant design features of the reactor building structure, reactor building isolation system, and penetration room ventilation system.

Specification

5.2.1 Reactor Building Structure

The reactor building completely encloses the reactor and the associated reactor coolant system. It is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal net free volume of the reactor building is approximately 1.81×10^6 cu. ft. The approximate inside dimensions are: diameter--116'; height--207'. The approximate thickness of the concrete forming the buildings are: cylindrical wall--3-3/4'; dome--3-1/4'; and the foundation slab--9'.

The concrete reactor building structure provides adequate shielding for both normal operation and accident situations. Design pressure and temperature are 59 psig and 286 F, respectively.

The reactor building is designed for an external atmospheric pressure of 3.0 psi greater than the internal pressure. This corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110 F and it is subsequently cooled to an internal temperature of less than 50 F. Since the building is designed for this pressure differential, vacuum breakers are not required.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 14 with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is

4.0

LA1
SAR

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)

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)

⟨LATER⟩
(3.7)

LATER

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)

REFERENCES:

(1) FSAR Section 5.1

(2) FSAR Section 5.2.5

(3) FSAR Section 6.5

LA1
SAR

⟨LATER⟩
(3.7)

LATER

4.2

5.3 REACTOR

Specification

5.3.1 Reactor Core

4.2.1

5.3.1.1 The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide ~~pellets.~~ (UO_2) as fuel material. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. A1

5.3.1.2 ~~The reactor core approximates a right circular cylinder with an equivalent diameter of 128.9 inches and an active height of 144 inches. The active fuel length is approximately 142 inches.(2)~~ LA1 SAR

5.3.1.3 ~~The average enrichment of the initial core is a nominal 2.62 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is less than 3.5 weight percent U-235.~~ LA1 SAR

4.2.2

5.3.1.4 There are 60 ~~full-length~~ control rod assemblies (CRA~~s~~) and 8 axial power shaping rod assemblies (APSRA~~s~~) ~~distributed in the reactor core as shown in FSAR Figure 3-60. The full-length~~ CRA~~s contain a 134-inch length of~~ silver-indium-cadminum alloy, and the ~~clad with stainless steel. Each~~ APSRA~~s contains a 63-inch length of~~ control material of Inconel-600 alloy ~~(3)~~, as approved by the NRC. LA1 SAR A1

5.3.1.5 ~~The initial core had 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding is Zircaloy-4 filled with alumina-boron and placed in the core as shown in FSAR Figure 3-2.~~ LA1 SAR

5.3.1.6 ~~Reload fuel shall conform to the design and evaluation described in FSAR and shall not exceed an enrichment of 4.1 weight percent of U-235.~~ LA1 SAR

5.3.2 ~~Reactor Coolant System~~

5.3.2.1 ~~The reactor coolant system is designed and constructed in accordance with code requirements.(4)~~ LA1 SAR

4.2

5.3.2.2 ~~The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, are designed for a pressure of 2500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line are designed for a temperature of 670 F.(5)~~

LA1

SAR

5.3.2.3 ~~The reactor coolant system volume is less than 12,200 cubic feet.~~

LA1

TRM

~~REFERENCES:~~

~~(1) FSAR, Section 3.2.1~~

~~(2) FSAR, Section 3.2.2~~

~~(3) FSAR, Section 3.2.4.2~~

~~(4) FSAR, Section 4.1.3~~

~~(5) FSAR, Section 4.1.2~~

A1

4.3

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

~~Applicability~~ A1

~~Applies to storage facilities for new and spent fuel assemblies.~~

~~Objective~~

~~To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.~~

Specification

5.4.1 New Fuel Storage

4.3.1.2.a-e

1. New fuel assemblies may be stored in the Fresh Fuel Storage Rack (FFSR). The FFSR consists of ~~a nine by eight array of~~ storage cells on nominal center to center distance of 21 inches in both directions. Ten interior storage cells, as shown in Figure 5.4-1, are precluded from use ~~and will be physically blocked prior to any storage in the fresh fuel rack.~~ This configuration is sufficient to maintain a K_{eff} of less than 0.98 with optimum moderation and 0.95 under normal conditions, based on fuel with an enrichment of 4.1 weight percent U-235. LA1 TRM LA1 TRM A3

4.3.1.1.d
4.3.1.1.e

2. New fuel may also be stored in the spent fuel pool ~~or in its shipping containers.~~ LA1 SAR

5.4.2 Spent Fuel Storage

4.3.1.1.b

1. The spent fuel racks are designed and shall be maintained so that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) when the pool is flooded with unborated water.

~~2. The spent fuel pool and the new fuel pool racks are designed as seismic Class I equipment.~~ LA1 SAR

~~REFERENCES~~

~~FSAR, Section 9.6~~ A1

<Add 4.3.1.1.c> M1

4.3

Fig. 4.3.1.2-1 ~~FIGURE 5.4-1~~ ANO FFSR LOADING PATTERN A1

<-----NORTH

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

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

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 4.0: Design Features

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:

No unit specific "Less Restrictive" changes identified.

ITS DISCUSSION OF DIFFERENCES

ITS Section 4.0: Design Features

1 NUREG 4.2.1 - Minor revisions are incorporated into the Improved Technical Specifications (ITS) description of fuel assemblies pursuant to the Revised Standard Technical Specification (RSTS) 4.2.1. The use of zircaloy is clarified as cladding material by the addition of the term "clad." ZIRLO is omitted since it is not intended to be used as cladding material for this unit. This change is consistent with the requirements of 10 CFR 50.46, which allows the use of either cladding material. The allowance for "limited substitutions of zirconium alloy filler rods for fuel rods" is currently not approved for use in ANO-1 and is omitted in the ITS. These changes are consistent with current license basis.

2 NUREG 4.2.2 - Incorporates TSTF-123, Rev 1.

3 NUREG 4.2.2 - The plant specific "control material" in the CONTROL RODS is silver indium cadmium and Inconel in the APSRs as identified in CTS 5.3.1.4. This change is consistent with current license basis.

4 NUREG 4.3.1.1 - There are only high density spent fuel storage racks provided at ANO-1. Therefore, there is no need to differentiate between high density and low density racks in ITS 4.3.1.1, nor to provide any information on low density storage racks pursuant to NUREG 4.3.1.1.d. This change is consistent with current license basis.

5 NUREG 4.3.1.2 - The CTS 5.4.1.1 plant specific controls which preclude storage in ten of the interior new (fresh) fuel storage rack locations are retained. These controls are necessary to assure the margin to criticality required by ITS 4.3.1.2.c is maintained as discussed in the submittal documents and the Safety Evaluation Report related to Amendment No. 166. This change is consistent with current license basis.

6 NUREG 4.3.2 & 4.3.3 - There are no CTS requirements to be transferred to the ITS which provide limitations on the drainage of the fuel storage pools, nor on the capacity of the spent fuel storage pool. Such controls have been adequately maintained through administrative controls since original licensing and are proposed to continue to be so controlled. This change is consistent with current license basis.

7 NUREG 4.3.1.2 - Details of reactivity conditions of the fuel storage racks were revised to reflect requirements from CTS 5.4. The ANO-1 SAR does not provide sufficient detail to support adoption of requirements as presented in NUREG-1430. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

8 NUREG 4.3.1.1 - The specific reference to a description in a particular section of the FSAR is omitted. Such a specific reference is not in the CTS as part of the requirement, and is undesirable due to potential interpretation that documents incorporated by reference require a license amendment to be changed, i.e., the referenced discussion in the SAR could not be changed because it is referenced by TS. Such limitations would not be appropriate for changes to the FSAR; therefor, the specific reference is not included. This change is consistent with current license basis.

CTS

4.0 DESIGN FEATURES

4.1 Site Location ~~[Text description of site location.]~~ < INSERT 4.0-1A > 5.1

4.2 Reactor Core

5.3.1.1

4.2.1 Fuel Assemblies

177 clad

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

1

edit

Control Assemblies CONTROL ROD assembly CONTROL ROD assemblies

4.2.2 ~~CONTROL RODS~~

5.3.1.4

60

8 APSR assemblies.

The reactor core shall contain ~~[60]~~ safety and regulating and ~~[8] axial power shaping CONTROL RODS~~. The control material shall be ~~[silver-indium-cadmium, boron carbide, or hafnium metal]~~ as approved by the NRC. alloy and the APSR assembly control material shall be an Inconel alloy,

a.

2

3

4.3 Fuel Storage

4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ weight percent; 3.8.15

4.1

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties ~~as described in [Section 9.1 of the FSAR]~~; 5.4.2.1

8

(continued)

<INSERT 4.0-1A>

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

CTS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

NA

~~c.~~ f. A nominal ~~[ ]~~ 10.65 inch center to center distance between fuel assemblies placed in ~~[the high density fuel~~ storage racks~~];]~~ 4

~~[d. A nominal [ ] inch center to center distance between fuel assemblies placed in [the low density fuel storage racks];]~~

~~e.~~ g. d. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure ~~[3.7.17-1]~~ 3.7.14-1 ~~may be~~ allowed unrestricted storage in ~~[either]~~ Region 1 or Region 2 fuel storage rack~~(s)~~; and 5.4.1.2 3.8.16 edit

~~f.~~ g. e. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure ~~[3.7.17-1]~~ ~~will be~~ stored in ~~compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~ Region 1, or in a checkerboard configuration in Region 2. 3.8.16 edit

edit

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

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 4.1 weight percent; 5.4.1.1

b. $k_{eff} \leq 0.95$ under normal conditions ~~if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]~~; 5.4.1.1 7

c. $k_{eff} \leq 0.98$ with optimum moderation; ~~if moderated by aqueous foam, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]; and~~ 5.4.1.1

d. A nominal ~~[21.125]~~ 21 inch center to center distance between fuel assemblies placed in the storage racks; and 5.4.1.1

e. Ten interior storage cells, as shown in Figure 4.3.1.2-1, precluded from use during fuel storage. 5.4.1.1 5

(continued)

CTS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

~~4.3.2 Drainage~~

NA

~~The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation [138 ft 4 inches].~~

6

~~4.3.3 Capacity~~

NA

~~The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than [1357] fuel assemblies [and six failed fuel containers].~~

<Insert 4.0-3A>

5

<u><INSERT 4.0-3A></u>

<----North

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

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

Figure 4.3.1.2-1 (page 1 of 1)
Fresh Fuel Storage Rack Loading Pattern

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
5.1	5.1	Responsibility
5.2	5.2	Organization
5.3	5.3	Unit Staff Qualifications
5.4	5.4	Procedures
5.5	5.5	Programs and Manuals
5.6	5.6	Reporting Requirements
5.7	5.7	High Radiation Area

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

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

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

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

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

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

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

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

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

5.2.2 Unit Staff

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

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

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

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

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

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

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

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

5.5.1 Offsite Dose Calculation Manual (ODCM)

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

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

Licensee initiated changes to the ODCM:

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

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

a. Preventive maintenance and periodic visual inspection requirements; and

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

5.5.3 Post Accident Sampling

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

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

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

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

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

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

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

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.5 (Not Used).

5.5.6 (Not Used).

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

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

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

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

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

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

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

5.5.9 Steam Generator (SG) Tube Surveillance Program

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

b. Examination Methods:

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

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

c. Selection and Testing:

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

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

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

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

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

v. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to 5.5.9.d. Tubes with ODIGA identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with topical report BAW-10235P, Revision 1.

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

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

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

NOTES:

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

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

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

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

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

---

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

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

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

iv. A main steam line or feedwater line break.

e. Acceptance Criteria:

1. Terms as used in this program:

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

vii. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply during Cycle 16 to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with topical report BAW-10235P, Revision 1.

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

The reroll repair process will only be used to repair tubes with defects in the upper tubesheet area. The reroll repair process will be performed only once per steam generator tube using a 1 inch roll length. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. The reroll repair process is described in the topical report, BAW-10232P, Revision 00.

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

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

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

5.0 ADMINISTRATIVE CONTROLS

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TABLE 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

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TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION [2,3]

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

NOTES:

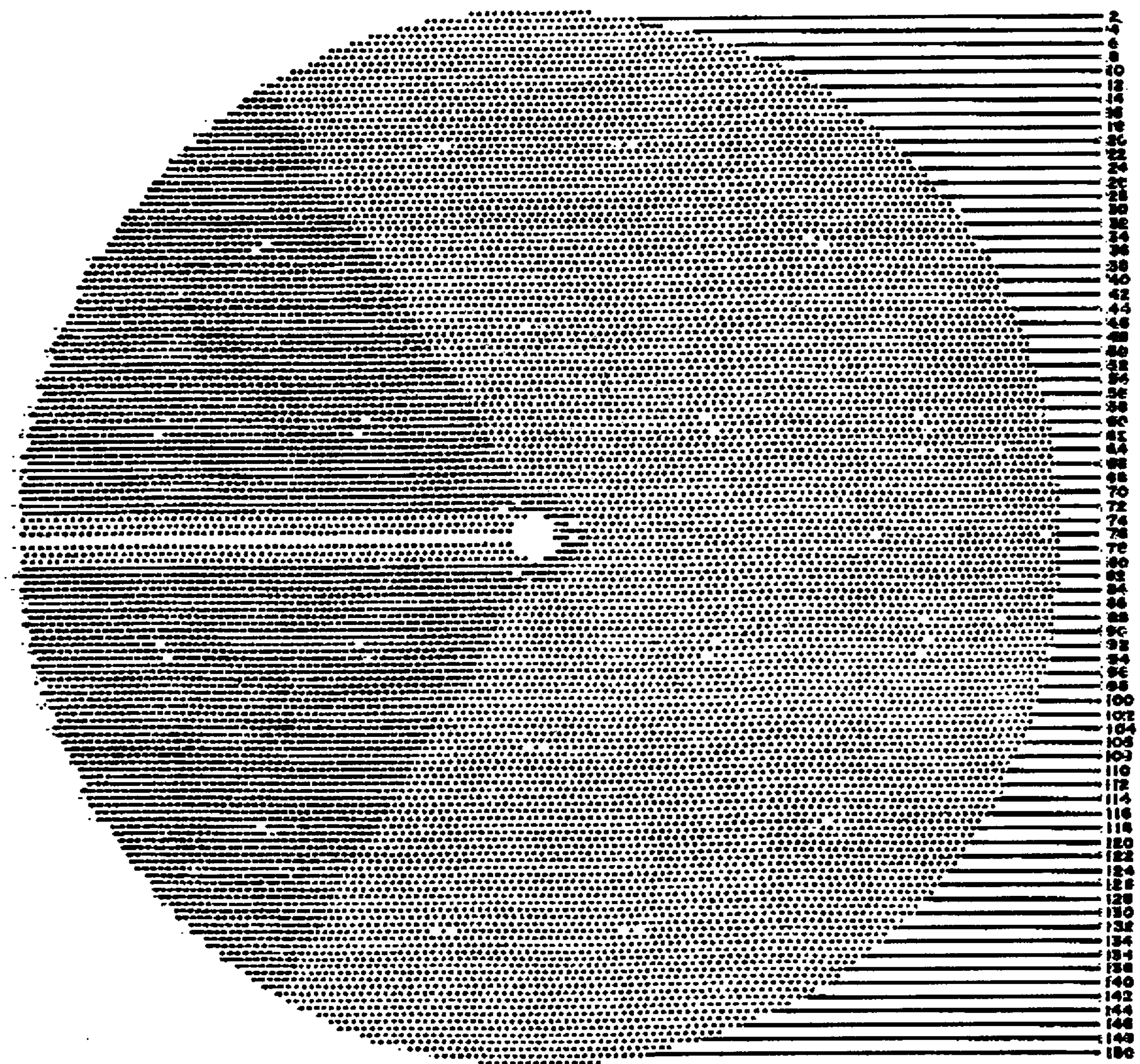
1 $S = \frac{3N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals



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

FIGURE 5.5.9-1 (page 1 of 1)

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry

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

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

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

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

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

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

5.5.11 Ventilation Filter Testing Program (VFTP)

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

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

   1. $\geq$ 99% halogenated hydrocarbon removal for the PRVS and FHAVS when tested at the system design flowrate $\pm$ 10%; and
   2. $\geq$ 99.95% halogenated hydrocarbon removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm $\pm$ 10%.

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

   1. $<$ 5% for the PRVS when tested at the system design flowrate $\pm$ 20%;

   2. $<$ 5% for the FHAVS when tested at the system design flowrate $\pm$ 20%; and

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

      i. $\leq$ 2.5% for 2 inch charcoal adsorber beds; and

      ii. $\leq$ 0.5% for 4 inch charcoal adsorber beds.

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

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

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The program shall include:

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

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

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

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

5.5.13 Diesel Fuel Oil Testing Program

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

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

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

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

    3. water and sediment within limits;

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

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

5.5.14 Technical Specifications (TS) Bases Control Program

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

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

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

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

   2. A change to the updated SAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

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

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

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5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP)

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

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

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

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

d. Other appropriate limitations and remedial or compensatory actions.

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.16 Reactor Building Leakage Rate Testing Program

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

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

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.1 Occupational Radiation Exposure Report

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

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

5.6.2 Annual Radiological Environmental Operating Report

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

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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

5.6.3 Radioactive Effluent Release Report

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

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

5.6.4 Monthly Operating Reports

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.6 Reactor Building Inspection Report

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

5.6.7 Steam Generator Tube Surveillance Reports

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

   1. Number and extent of tubes inspected;

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

   3. Identification of tubes plugged and tubes sleeved;

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

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

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

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

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

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

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

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

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

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

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

CTS DISCUSSION OF CHANGES

ITS Section 5.0: Administrative Controls

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 Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG-1430, Revision 1 and 10 CFR Part 20. 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 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 A statement regarding the Applicability of SR 3.0.2 and SR 3.0.3 is added for clarification that the allowances provided by these general Surveillance Requirements are applicable to the identified program. This is an administrative change since the CTS 4.0.2 and 4.0.3 are currently applicable to the requirements being moved to the program that will be identified in the Administrative Controls (Section 5). This change is applicable for CTS 4.2.6 which is to be incorporated into the Reactor Coolant Pump Flywheel Inspection Program, ITS 5.5.7, and to CTS 4.10, 3.13, and 3.15 which are to be incorporated into the Ventilation Filter Testing Program, ITS 5.5.11. This change is also applicable for CTS 3.24, 3.25.1 and 3.25.2 which are to be incorporated into the Explosive Gas and Storage Tank Radioactivity Monitoring Program, ITS 5.5.12, and to CTS 4.6.1.4.e which is to be incorporated into the Diesel Fuel Oil Testing Program, ITS 5.5.13. Additionally, this change is applicable for CTS 4.0.5 which is to be incorporated into the Inservice Testing Program, ITS 5.5.8.

A4 CTS 4.18.6 and Table 4.18-2 reference to a Special Report are removed from the markup to show the editorial removal of cross references in the ITS. This is considered an administrative change because ITS 5.6.7 will continue to have the additional reporting requirements prescribed in the "special" report. This is considered editorial and no change in requirements are associated with this change. This change is consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

A5 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
4.0.5	10 CFR 50.55a(f) and 50.55a(g)
4.2.2	10 CFR 50.55a(g)
4.2.3	10 CFR 50.55a(g)
4.3.1 & 4.3.3	10 CFR 50.55a(g)
4.27.2	10 CFR 50.55a(g)
Table 6.2-1 Note *	10 CFR 50.54(m)(2)(iv)
Table 6.2-1 Add. Req. 1	10 CFR 50.54(m)(2)(iii)
Table 6.2-1 Add. Req. 2	10 CFR 50.54(m)(2)(iii)
Table 6.2-1 Add. Req. 4	10 CFR 50.54(m)(2)(iv)
6.10	10 CFR 20
6.12.1	10 CFR 50.4
6.12.3.4	10 CFR 50.4
6.12.5	10 CFR 50.4

The following CTS sections also detail requirements duplicated in the referenced Regulation. However, since 10 CFR 55.4 infers a requirement for Technical Specifications to reference these specifics, the CTS requirement is editorially revised to reflect the Regulation:

6.2.2	10 CFR 50.54(m)(2)(i)
Table 6.2-1 SOL	10 CFR 50.54(m)(2)(i)
Table 6.2-1 OL	10 CFR 50.54(m)(2)(i)

A6 The CTS 4.2.1 pre-operational requirements have been previously completed. Therefore, this surveillance is no longer required, and it deletion is an administrative change.

A7 NUREG 5.5.8, Inservice Testing Program, includes "every 9 months" and "biennially or every 2 years" as ASME test frequencies, and provides a specific number of days (276 and 731 days respectively) by which to interpret these frequencies. Since these frequencies are already provided in the ASME Code and/or NUREG-1430, and the interpretation is simply an obvious editorial clarification, this change is administrative.

A8 The presentation of the requirements for ventilation filter testing is revised for consistency. All frequencies are replaced by a reference to perform the testing at the frequencies specified in Regulatory Guide 1.52, Rev. 2. Since there is no actual change in the Frequencies, this change is considered to be one of presentation only, and therefore, administrative in nature.

CTS DISCUSSION OF CHANGES

A9 Not used.

A10 This page is not yet approved in its current form. Therefore, this markup is dependent on the expected NRC approval of the November 23, 1999, license amendment request (Ref. 0CAN119906) related to revision of the laboratory testing of activated charcoal requirements.

A11 The "$\leq 0.60\ L_a$" and "$\leq 0.75\ L_a$" limits for acceptable reactor building leakage in CTS 6.8.4 have been revised to "$< 0.60\ L_a$" and "$< 0.75\ L_a$" for consistency with the acceptance criteria provided in 10 CFR 50, Appendix J. These are considered to be essentially equivalent since the parameter can be less than the limit, but be so close as to be imperceptible. Therefore, this change has no impact on application of the regulations and is considered administrative.

A12 CTS markup Insert 110jA shows adoption of a statement that the ITS SR 3.0.2 and ITS SR 3.0.3 allowances are applicable to the ITS SG Tube Inspection Program. This is necessary in the ITS to clearly establish that the Section 3.0 allowance is applicable to the Section 5.0 requirements regarding SG tube inspection. The CTS did not require this statement because the SG tube inspection requirements were located within the Surveillance Requirements Section of the CTS and was clearly subject to the CTS 4.0.2 and CTS 4.0.3 allowances. This change is consistent with NUREG-1430 as modified by TSTF-118 with the addition of ITS SR 3.0.3, consistent with the ANO-1 current licensing basis.

A13 CTS 6.12.2.2 is revised to reflect the correct 10 CFR 20 terminology for the units of occupational exposure. A statement limiting the report scope to those persons monitored was added as a statement of the obvious. Lastly, the pocket dosimeter was revised to refer to a pocket ionization chamber and the electronic dosimeter was specified as an additional means of collecting the exposure data. These changes are considered purely administrative since they result in no relaxation of requirements, result in compliance with 10 CFR 20, more accurately reflect the principal of operation of the pocket dosimeter, and acknowledge industry usage of advanced dosimetry devices. These changes are consistent with 10 CFR 20 and NUREG-1430 as revised by TSTF-152.

A14 CTS 6.12.2.6 is revised to reflect the reporting requirements consistent with 10 CFR 20 and minor editorial changes. These changes are considered purely administrative since they result in no relaxation of requirements and result in compliance with 10 CFR 20. These changes are consistent with 10 CFR 20 and NUREG-1430 as revised by TSTF-152.

A15 This page is not yet approved in its current form. Therefore, this markup is dependent on the expected NRC approval of the July 14, 1999, license amendment request (Ref. 0CAN079901) related to revision of PASS requirements.

A16 CTS 4.18.5.b, 2nd paragraph, was added by amendment 203 as a one-time, temporary change -- only applicable through Cycle 16. Since ANO-1 will complete Cycle 16 prior to implementation of ITS, this provision can be deleted. This is an administrative change.

A17 CTS 6.8.5 is updated to reflect the latest changes to 10 CFR Part 20. The changes maintain the same overall level of effluent control while retaining the operational flexibility that currently exists. The Specification continues to provide reasonable assurance that acceptable limits will be maintained and eliminate possible confusion or improper implementation of the revised 10 CFR Part 20 requirements. Additionally, consistent with the intent of performing periodic surveillances, a statement regarding the Applicability of SR 3.0.2 and SR 3.0.3 is added. Since no change to the regulatory requirements is made this change is considered administrative.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS 6.3.1 is updated to reflect the latest changes to the QAPM approved by the NRC on November 6, 1998 (TAC No. M97893). Unit staff qualifications are revised to reflect commitments to ANSI ANS 3.1-1978 (in lieu of ANSI N18.1-1971). Additional experience and education requirements are imposed for certain positions due to this change. This change is an additional restriction on unit operation.

M2 Not used.

M3 CTS 6.8.1 is revised to incorporate additional procedure requirements. The reference to Regulatory Guide 1.33, Appendix A, is updated from November 1972, to reference Revision 2 of the guidance, dated February 1978. This updated reference is consistent with the current reference in the ANO-2 CTS, and with the RSTS. An additional item is incorporated to require emergency operating procedures for implementation of the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33. This is consistent with the CTS requirements prior to Amendment 179 and with the RSTS. Finally, additional requirements are included to provide procedures for each of the programs identified in proposed ITS 5.5. Of these, only two programs are totally new: the Technical Specification Bases Control Program and the Safety Function Determination Program (see DOC M7 below). The remaining programs are based on requirements in the CTS. This change is also consistent with the RSTS and is an additional restriction on unit operation.

M4 Not used.

M5 CTS 3.13.1.d, CTS 3.15.1.d, CTS 4.10.2.d.1, CTS 4.11.1, and CTS 4.17.1 are revised to include the prefilters and "roughing" filters in the ventilation system differential pressure testing requirements. The revision is shown as "other filters in the system" to accommodate system specific nomenclature and system design variances. These filters are part of the system and obviously do contribute to the system pressure drop and capability of the system to perform its function. Therefore, inclusion of the prefilters in this testing is appropriate. This change is an additional restriction on unit operation.

M6 Not used.

M7 Two new programs are proposed for inclusion in the ITS. These are ITS 5.5.14, "Technical Specification Bases Control Program," and ITS 5.5.15, "Safety Function Determination Program." Both of these programs are necessary for proper implementation of the ITS, and are consistent with NUREG-1430. These new programs are an additional restriction on unit operation.

CTS DISCUSSION OF CHANGES

M8 CTS 4.26.2 requires the reactor building purge supply and exhaust isolation valves to be local leak rate tested in accordance with the 10 CFR 50, Appendix J requirements, but on a Frequency which is more restrictive than the Appendix J frequency. The CTS Frequency is related to reactor building integrity, and the ITS Frequency will also require the testing on a Frequency similarly related to the Applicability for reactor building OPERABILITY. However, the Applicability for reactor building integrity/OPERABILITY has been revised (see ITS Section 3.6) in a manner which is more restrictive than CTS. This change in Applicability is also reflected in this Surveillance Requirement Frequency. This is an additional restriction on unit operation.

M9 CTS 4.6.1.4.e is revised to include testing of new fuel oil. Immediate confirmation of fuel oil quality (by monitoring for specific gravity, viscosity, and appearance/color) as well as follow up confirmatory testing within 30 days after adding new fuel oil to the bulk storage tank will provide added assurance of acceptable fuel oil. This broad spectrum testing will not be routinely performed (refer to DOC L6) since this initial verification provides the necessary confirmation of fuel oil quality. Additionally, this testing is in accordance with NUREG-1430. This is an additional restriction on unit operation.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 Not used.

L2 The CTS 6.11 requirements for high radiation areas are revised to include additional, previously approved methods for implementation of alternates to the "control device" or "alarm signal" requirements of 10 CFR 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG-1430 and approval of generic change TSTF-258, Revision 4.

L3a CTS 6.12.2.2 is revised to require the submittal of the Occupational Exposure Data Report by April 30 of each calendar year. This change is consistent with the comprehensive revisions to 10 CFR 20. The date of submittal for the Annual Occupational Exposure Report is revised from March 1 to April 30. This report is provided to supplement the information required by 10 CFR 20.2206(b) which is filed on or before April 30 in accordance with 10 CFR 20.2206(c). The supplemental information report submittal date is therefore revised to correspond to the required submittal date of the report being supplemented.

L3b The CTS 6.12.2.4 requirements for reporting of all challenges to the pressurizer electromatic relief valves (ERVs) and the pressurizer safety valves is omitted. Reporting of these challenges was incorporated into the CTS in response to TMI Action Item II.K.3.3. This action plan item was originally implemented only to provide a venue for data gathering, and this requirement has been in effect since 1980. There is no plant specific safety basis for submitting routine information on the operation of this particular equipment. Finally, any challenges to these valves that result in a potential impact on safety would be evaluated for reportability under 10 CFR 50.73. See also: NUREG-0565, items 2.1.2.c & 2.1.2.e; NUREG-0611, items 3.2.4.h & 3.2.4.j; NUREG-0626, items F-2.5 & F-3.5; and NUREG-0635, item 3.2.4.d for background information on this report.

L4 CTS 3.7.3.A.1 provides a requirement for a redundant subsystem verification for the purpose of identifying a potential loss of safety function. CTS 3.7.3.B would require a shutdown if a potential loss of safety function were discovered. The ITS does not always require a shutdown if a loss of function is identified. Rather, it requires that both redundant components be declared inoperable and the corresponding ACTIONS of the LCO applicable for those components be entered. These ACTIONS may provide for other compensatory measures that have been determined to be appropriate for the condition. Therefore, this CTS requirement is more appropriately addressed with the added Safety Function Determination Program of ITS 5.5.15. This change is consistent with NUREG-1430.

L5 CTS Table 4.1-2, item 12 requires verification, at normal operating conditions that a gap of at least 0.025 inches exists between the main feedwater line pipe and the flow limiting annulus at the reactor building penetration. The Frequency is identified as "one year, two years, three years, and every five years thereafter measured from date of initial test." This requirement was included in the original operating license when issued in 1974. The purpose of the annulus attached to each main feedwater line reactor building penetration is to limit the blowdown into the penetration room from a postulated double ended rupture inside the penetration sleeve (see SAR Section A.7.3.5). The circular plate that provides the flow limitation is welded to the penetration and not subject to fluctuation except due to radial expansion during heatup. Therefore, the Frequency for this verification is revised to "following any modifications which may affect the required gap." Additionally, this verification is removed from the Technical Specifications since it is a specific design feature of a structure that is only subject to change via the design change process. As such, the "post-modification" verifications are also required by the design change process, and as with other post-modification type requirements, can be removed from the Technical Specifications without a significant impact on safety. This change is consistent with NUREG-1430.

L6 CTS 4.6.1.4.e is revised to require the periodic testing of the stored fuel oil only for particulates (replacing the periodic testing per ASTM-D975) once every 31 days per ITS 5.5.13 (refer to DOC M9 for added testing requirements). This change also relaxes CTS requirement that the sample and testing be in conjunction with the monthly DG run. These changes reflect industry-standard acceptable DG fuel oil testing programs reflected in NUREG-1430. Over the storage life of ANO-1 DG fuel oil, the properties tested by ASTM-D975 are not expected to change and performing these tests once on the new fuel oil (see DOC M9) provides adequate assurance of the proper quality fuel oil. The periodic testing for particulates monitors a parameter that reflects degradation of fuel oil and can be trended to provide increased confidence that the stored DG fuel oil will support DG operability.

CTS DISCUSSION OF CHANGES

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 Where possible, plant specific management position titles in the CTS are replaced with generic titles as provided in ANSI/ANS 3.1. Personnel who fulfill these positions are still required to meet the qualifications detailed in proposed Specification 5.3. In addition, compliance details relating to the plant specific management position titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document, such as the Quality Assurance Program Manual (QAPM). This approach is consistent with the intent of Generic Letter 88-06 which recommended, as a line item improvement, relocation of the corporate and unit organization charts to licensee controlled documents. The intent of the Generic Letter, and of this proposed change, is to reduce the unnecessary burden on NRC and licensee resources being used to process changes due solely to personnel titles changes during reorganizations. Since this change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions, the change is considered to be a change in presentation only and is therefore administrative. The specific replacements are:

ITS 5.5.1	ANO general manager for General Manager, Plant Operations

LA2 The requirement for a third non-licensed operator is moved to a licensee controlled document such as the Safety Analysis Report (SAR). This information provides details of the method of implementation of other requirements (e.g., fire protection) which are not directly pertinent to the specific shift manning requirements, and which are no longer directly controlled by Technical Specifications (since they did not meet the inclusion criteria of 10 CFR 50.36). Since this detail is not necessary to adequately describe the actual regulatory requirement, it can be moved to a licensee controlled document without a significant impact on safety. Placing this detail in controlled documents provides adequate assurance that it will be maintained. The SAR will be controlled by 10 CFR 50.59 and 50.71(e).

CTS DISCUSSION OF CHANGES

LA3 This information has been moved to a licensee controlled document such as the Ventilation Filter Testing Program (VFTP), Diesel Fuel Oil Testing Program (DFOTP), or Reactor Building Leakage Rate Testing Program (RBLRTP), etc. A description of the Program is incorporated into the Administrative Controls section of ITS. This information provides details of the method of implementation which are not directly pertinent to the actual requirement. 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 details relocated to the VFTP, RBTSP, DFOTP, and RBLRTP will be controlled by 10 CFR 50.59. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
3.13.1.e	VFTP
3.15.1	VFTP
4.6.1.4.e	DFOTP
4.10	VFTP
4.11	VFTP
4.17	VFTP

LA4 The CTS Operating License Condition 2.C(6) requirement for monitoring of iodine in vital areas (except the containment atmosphere which is retained in ITS 5.5.3) is moved to a licensee controlled document such as the Safety Analysis Report (SAR). This information provides details of the method of implementation of other requirements (e.g., radiation protection) which are not directly pertinent to the safe shutdown of the unit, and which are no longer directly controlled by Technical Specifications (since they did not meet the inclusion criteria of 10 CFR 50.36.) Since this detail is not necessary to adequately describe the actual regulatory requirement, it can be moved to a licensee controlled document without a significant impact on safety. Placing this detail in controlled documents provides adequate assurance that it will be maintained. The SAR will be controlled by 10 CFR 50.59 and 50.71(e).

CTS DISCUSSION OF CHANGES

LA5 This information has been moved to a licensee controlled document such as the Explosive Gas and Storage Tank Radioactivity Program (EG&STRMP). A description of the Program is incorporated into the Administrative Controls section of ITS which includes appropriate limits, actions, and surveillance requirements. The information moved to the TRM provides details of the method of implementation which are not directly pertinent to the actual requirement. 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 details relocated to the EG&STRMP will be controlled by 10 CFR 50.59. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
3.24	EG&STRMP
3.25.1	EG&STRMP
3.25.2	EG&STRMP
4.28 & Figure 3.7.4-1	EG&STRMP
4.29.1	EG&STRMP
4.29.2	EG&STRMP

LA6 This information has been moved to a licensee controlled document such as the Bases, Safety Analysis Report (SAR), QAPM, TRM, etc. This information provides details of the method of implementation that are not directly pertinent to the actual requirement. 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 Process identified in Chapter 5 of the proposed ITS. The details relocated to the SAR and TRM will be controlled by 10 CFR 50.59. The details relocated to the QAPM will be controlled by 10 CFR 50.54(a)(3). The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

CTS Location	New Location
4.2.4	SAR (3.2.4)
4.2.5	QAPM
6.12.2	TRM

5.5.2
5.5.3

Not addressed by TSIP.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10CFR73.55 (51 FR 27817 and 27822) and to the authority of 10CFR50.90 and 10CFR50.54(p). The plan, which contains Safeguards Information protected under 10CFR73.21, is entitled: "Arkansas Nuclear One Industrial Security Plan," with revisions submitted through August 4, 1995. The Industrial Security Plan also includes the requirements for guard training and qualification in Appendix A and the safeguards contingency events in Chapter 7. Changes made in accordance with 10CFR73.55 shall be implemented in accordance with the schedule set forth therein.

5.5.2 (5) Systems Integrity Not used.

EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

LA4
SAR

5.5.3 (6) Iodine Monitoring Not used.

EOI shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration ~~in vital areas~~ under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

5.5.10

5.5.10 (7) Secondary Water Chemistry Monitoring Not used.

This program provides controls for

A ~~A~~ secondary water chemistry monitoring ~~program shall be implemented~~ to ~~minimize~~ inhibit steam generator tube degradation. This program shall include:

A1

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

2. Identification of the procedures used to measure the values of the critical ~~parameters~~ variables;

3. Identification of process sampling points;

4. Procedures for the recording and management of data;

5. Procedures defining corrective actions for off-control point chemistry conditions; and

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

Not addressed by TSIP.

(8) FIRE PROTECTION

EOI shall implement and maintian in effect all provisions of the approved Fire Protection Program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992, subject to the following provision:

> The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license is effective as of the date of issuance and shall expire at midnight, May 20, 2014.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by:
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendices A and B - Technical Specifications

Date of Issuance: May 21 1974

5.5.15

<LATER> (3.8)

3.7.3 Both 125 VDC electrical power subsystems shall be operable when the unit is above the cold shutdown condition.

LATER

A. With one 125 VDC electrical power subsystem inoperable:

5.5.15

1. verify that there are no inoperable safety related components associated with the operable 125 VDC electrical subsystem which are redundant to the inoperable 125 VDC electrical power subsystem, (L4)

2. verify the operability of the diesel generator associated with the operable 125 VDC electrical subsystem immediately, and

3. restore the 125 VDC electrical subsystem to operable status within 8 hours.

<LATER> (3.8)

B. With one 125 VDC electrical power subsystem inoperable, and unable to satisfy the requirements or allowable outage times of 3.7.3.A.1, 3.7.3.A.2, or 3.7.3.A.3, the unit shall be placed in hot shutdown within 12 hours and in cold shutdown within an additional 24 hours.

3.7.4 Battery cell parameters shall be within limits when the associated DC electrical power subsystems are required to be operable.

A. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits:

1. Within 1 hour, verify pilot cell electrolyte level and float voltage meet Table 4.6-1 Category C limits,

2. Within 24 hours and once per 7 days thereafter, verify battery cell parameters meet Table 4.6-1 Category C limits, and

3. Within 31 days, restore battery cell parameters to Table 4.6-1 Category A and B limits.

B. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits and unable to satisfy the requirements or allowable outage times of 3.7.4.A.1, 3.7.4.A.2, or 3.7.4.A.3, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

C. With one or more batteries with electrolyte temperature of the pilot cell not within the limits of Specification 4.6.2.8, electrolyte temperature of representative cells not within the limits of Specification 4.6.2.6 or with one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category C limits, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

Bases

The electrical system is designed to be electrically self-sufficient and provide adequate, reliable power sources for all electrical equipment during startup, normal operation, safe shutdown and handling of all emergency situations. To prevent the concurrent loss of all auxiliary power, the various sources of power are independent of and isolated from each other.

LATER

5.5.11

<ADD PROGRAM DESCRIPTION>

<LATER> (3.7)

3.13 PENETRATION ROOM VENTILATION SYSTEM

Applicability

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

Objective

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

Specification

3.13.1 Two independent circuits of the penetration room ventilation system shall be operable whenever reactor building integrity is required with the following performance capabilities:

5.5.11.a.1 5.5.11.b.1 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.

5.5.11.c.1 b. The results of laboratory carbon sample analysis from the charcoal adsorber banks shall show the methyl iodide penetration less than 5.0% at velocity within ± 20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

5.5.11.a.1 5.5.11.b.1 5.5.11.d — c. Fans shall be shown to operate within ± 10% of design flow.

5.5.11.d d. The pressure drop across ~~the combined HEPA filters~~ other filters in the system and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (± 10%). M5

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. LA3 VFTP

<LATER> (3.7)

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

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.

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 within 36 hours.

LATER

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

A10

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 indicate 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.

5.5.11

<LATER> (3.7)

~~3.15 FUEL HANDLING AREA VENTILATION SYSTEM~~ LATER

~~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~~

~~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 following performance capabilities:~~

5.5.11.a.1 5.5.11.b.1 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.

5.5.11.c.2 b. The results of laboratory carbon sample analysis shall show the methyl iodide penetration less than 5.0% at a velocity within ±20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

5.5.11.a.1 5.5.11.b.1 5.5.11.d c. Fans shall be shown to operate within ±10% design flow.

5.5.11.d d. The pressure drop across ~~the combined HEPA filters~~ , other filters in the system, and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%). M5

~~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.~~ LA3

<LATER> (3.7) ~~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 Specification 3.0.3 are not applicable.~~ <LATER>

~~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.~~ A2

5.0

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 indicate 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.

A2

5.5.12

<Add Program description>

LA5

EG&STRMP

5.5.12

3.24 EXPLOSIVE GAS MIXTURE

Applicability

Applies to the Waste Gas System hydrogen/oxygen analyzers.

Objective

To prevent accumulation of explosive mixture in the waste gas system.

Specification

3.24.1 The Concentration of hydrogen/oxygen shall be limited in the waste gas decay tanks to Region "A" of Figure 3.24-1.

3.24.2 When the hydrogen/oxygen concentration in any of the decay tanks enters Region "B" of Figure 3.24-1, corrective action shall be taken to return the concentration values to Region "A" within 24 hours.

3.24.3 The provisions of Specification 3.0.3 are not applicable.

A2

Bases

These hydrogen/oxygen limits provide reasonable assurance that no hydrogen/oxygen explosion could occur to allow rupture of the waste gas decay tanks. The hydrogen and oxygen limits are based on information in NUREG/CR-2726 "Light Water Reactor Hydrogen Manual".

<Add SR 3.0.2 & SR 3.0.3 applicability statement> A3

5.5.12

<ADD Program Description>

3.25 RADIOACTIVE EFFLUENTS

3.25.1 Radioactive Liquid Holdup Tanks

Applicability: At all times.

Objective: To ensure that the limits of 10 CFR 20 are not exceeded.

Specifications:

3.25.1 A. The quantity of radioactive material contained in each unprotected* outside temporary radioactive liquid storage tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

B. With the quantity of radioactive material exceeding the above limit, immediately suspend all additions of radioactive material to the affected tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.6.

C. The provisions of Specification 3.0.3 are not applicable.

LA5

EG+STRMP

Bases:

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank* the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

*Tanks included in this specification are those outdoor temporary tanks that 1) are not surrounded by liners, dikes, or walls capable of holding the tank contents, and 2) do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

A2

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

5.5.12

<ADD Program Description>

LA5

EG+STRMP

3.25.2 **Radioactive Gas Storage Tanks**

Applicability: At all times

Objective: To restrict the amount of activity in a radioactive gas holdup tank.

Specifications:

3.25.2 A. The quantity of radioactivity contained in each gas storage tank shall be limited to 300,000 curies noble gases (Xe-133 equivalent).

B. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.4.

C. The provisions of Specification 3.0.3 are not applicable.

Bases:

The value of 300,000 curies is a suitable fraction of the quantity of radioactive material which if released over a 2-hour period, would result in a total body exposure to a member of the public at the exclusion area boundary of 500 mrem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

A2

<ADD SR 3.0.2 + SR 3.0.3 applicability statement>

A3

5.5.8

SURVEILLANCE REQUIREMENTS

~~4.0.1 Surveillance Requirements shall be met during the operational modes or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.~~

LATER (3.0) ~~4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.~~ LATER

~~4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the Action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The time at which the Action is taken may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the Action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.~~

~~4.0.4 Entry into an operational mode or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational modes as required to comply with Action requirements.~~

A5

5.5.8 4.0.5 Surveillance Requirements for inservice ~~inspection and~~ testing of ASME Code Class 1,2, and 3 components shall be applicable as follows:

a. ~~Inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i),~~ A5

5.5.8

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

5.5.8

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice ~~inspection and~~ testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: A5

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice ~~inspection and~~ testing activities	Required frequencies for performing inservice ~~inspection and~~ testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days
<ADD: Every 9 mo>	
<ADD: "Biennially">	

A5 A7

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ test activities. A5

~~d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements~~ A1

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification. <INSERT SR 3.0.3 appl.> A3

4.1 OPERATIONAL SAFETY ITEMS

<LATER>
(3.3A)
(3.3B)
(3.3C)
(3.3D)

~~Applicability~~

~~Applies to items directly related to safety limits and limiting conditions for operation.~~

~~Objective~~

~~To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.~~

~~Specification~~

~~a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.~~

LATER

5.5.8

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

⟨LATER⟩ (3.3A) (3.3B) (3.3C) (3.3D)

b. ~~Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.~~

c. ~~Discrepancies noted during surveillance testing will be corrected and recorded.~~

LATER

⟨LATER⟩ (3.2)

d. ~~A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.~~

LATER

BASES

A2

~~4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):~~

~~"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."~~

~~4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.~~

5.5.8

BASES (continued)

includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of mode changes imposed by Action requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the Action requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the Action requirements are applicable at the time that the surveillance is terminated. If the Action requirements are greater than 24 hours, sufficient time exists to complete the surveillance.

⟨LATER⟩ (3.0) LATER

Surveillance Requirements do not have to be performed on inoperable equipment because the Action requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

4.0.4 Establishes the requirement that all applicable surveillances must be met before entry into an operational mode or other condition of operation specified in the Specification. The purpose of this Specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a mode or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational modes or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with Action requirements, the provision of Specification 4.0.4 do not apply because this would delay placing the facility in a lower mode of operation.

A2

4.0.5 Establishes the requirment that inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

A2

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

A2 & R TRM

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

A2

5.0

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
	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.	L5
<LATER> (3.7)	13. Main steam isolation valves	a. Exercise through approximately 10% travel b. Cycle	a. Quarterly b. Every 18 months	LATER
	14. Main feedwater isolation valves	a. Exercise through approximately 5% travel b. Cycle	a. Quarterly b. Every 18 months	
<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

Amendment No. ~~4~~, ~~21~~, ~~23~~, ~~Order dated 4/20/81~~, ~~91~~, 152

5.0

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

A1

Applicability

Applies to the surveillance of the reactor coolant system pressure boundary.

Objective

To assure the continued integrity of the reactor coolant system pressure boundary.

Specification

A6

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and baseline data for future inspections.

A5

4.2.2 Post-operational inspections of components shall be made in accordance with the methods and intervals indicated in Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

5.0

A5

4.2.3 ~~The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated.~~

LA6
SAR

4.2.4 ~~To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support, shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.~~

LA6
QAPM

4.2.5 ~~Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.~~

5.5.7

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

A3

<INSERT 77>

A2

~~**Bases**~~

~~The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10CFR50.55a, to the extent practicable within limitations of design, geometry and materials of construction.~~

<CTS INSERT 77>

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

5.0

A5

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

<LATER> (2.0)

LATER

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2155 psig, prior to the reactor being made critical, in accordance with the ASME Boiler and Pressure Vessel Codes, Section XI; IWA-5000.

A5

4.3.3 The limitations of Specification 3.1.2 shall apply.

A2

Bases

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section XI.

For normal opening, the integrity of the Reactor Coolant System in terms of strength, is unchanged. The ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000 requires a system leak test at nominal operating pressure (2155 psig) following system opening. At the end of refueling outages, this test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components.

REFERENCES

(1) FSAR, Section 4
(2) ASME Boiler and Pressure Vessel Code, Section XI

5.5.13

4.6 AUXILIARY ELECTRICAL SYSTEM TESTS

Applicability

Applies to the periodic testing and surveillance requirements of the auxiliary electrical system to ensure it will respond promptly and properly when required.

Specification

4.6.1 Diesel Generators

(LATER) (3.8)

1. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 15 seconds. The signal initiating the start of the diesel shall be varied from one test to another (start with handswitch at control room panel and at diesel local control panel) to verify all starting circuits are operable. The generator shall be synchronized from the control room and loaded to full rated load and allowed to run until diesel generator operating temperatures have stabilized. — LATER

2. A test shall be conducted once every 18 months to demonstrate the ability of the diesel generators to perform as designed by:

   a. simulating a loss of off-site power,

   b. simulating of loss of off-site power in conjunction with an ESF signal,

   c. simulating interruption of off-site power and subsequent reconnection of the on-site power source to their respective busses, and

   d. operating the diesel generator for ≥1 hour after operating temperatures have stabilized.

3. Each diesel generator shall be given an inspection once every 18 months following the manufacturer's recommendations for this class of standby service. (A one-time extension of this interval is allowed so that these may be performed during the 1R9 refueling outage, and completed no later than December 1, 1990.)

5.5.13 & LATER (3.8)

4. During the monthly diesel generator test specified in paragraph 1 above, the following shall be performed: — L8 & LATER

(LATER) (3.8)

   a. The diesel generator starting air compressors shall be checked for operation and their ability to recharge the air receivers.

   b. The diesel oil transfer pumps shall be checked for operability and their ability to transfer oil to the day tank. — LATER

   c. The day tank fuel level shall be verified.

   d. The emergency storage tank fuel level shall be verified.

5.5.13 <ADD: Diesel Fuel Oil Testing Program description> LA3

<ADD: SR 3.0.2 & SR 3.0.3 Applicability statement> A3

<ADD: New fuel oil testing> M9

e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits ~~specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment.~~ LG

~~5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.~~

~~Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.~~

<LATER> (3.8)

~~4.6.2 DC Sources and Battery Cell Parameters~~ LATER

~~1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.~~

~~2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.~~

~~3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.~~

~~4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.~~

~~5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.~~

~~6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.~~

~~7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.~~

~~8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.~~

~~4.6.3 Emergency Lighting~~

~~The correct functioning of the emergency lighting system shall be verified once every 18 months.~~ R TRM

5.5.11

4.10 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM SURVEILLANCE

LATER

Applicability

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

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:

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

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.

⟨LATER⟩ (3.7)

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:

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.

5.5.11 ⟨LATER⟩ (3.7) 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: A8 ⟨ LATER

5.5.11.a.2
5.5.11.b.2
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%. A8

5.5.11.c.3
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: LA3 VFTP

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

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

5.5.11.a.2
5.5.11.b.2
5.5.11.d
3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

5.5.11
5.5.11.c.3

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: A8 LA3

& <LATER> (3.7)

1. ≤ 2.5% for 2 inch charcoal adsorber beds, or

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

5.5.11
5.5.11.d

d. At least once per 18 months by: A8

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%. other filters in the system M5

<LATER> (3.7)

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. LATER

5.5.11
5.5.11.a.2

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%. A8

& <LATER> (3.7)

5.5.11
5.5.11.b.2

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%. A8

~~Bases~~

~~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.~~ A2

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

5.5.11

A.2

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.

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.

5.5.11

4.11 PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE

⟨LATER⟩ (3.7)

Applicability

Applies to the surveillance of the penetration room ventilation system. — LATER

Objective

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

Specification

LA3 VFTP & LATER

5.5.11.d & ⟨LATER⟩ (3.7)

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%). , other filters in the system, — MS

5.5.11 & ⟨LATER⟩ (3.7)

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. — LA3 VFTP & LATER

⟨LATER⟩ (3.7)

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

5.5.11 & ⟨LATER⟩ (3.7)

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.

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.

LA3 VFTP & LATER

⟨LATER⟩ (3.7)

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. — LATER

5.5.11

4.17 FUEL HANDLING AREA VENTILATION SYSTEM SURVEILLANCE

⟨LATER⟩ (3.7)

Applicability — LATER

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

5.5.11 5.5.11.d & LATER (3.7)

4.17.1 ~~At intervals not to exceed 18 months,~~ pressure drop across the combined HEPA filters, and charcoal adsorber banks , other filters in the system, shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%). LA3 VFTP & LATER M5

5.5.11 & LATER (3.7)

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. LA3 VFTP & LATER

5.5.11 & LATER (3.7)

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. LA3 VFTP & LATER

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.

⟨LATER⟩ (3.7)

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. — LATER

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

A1

5.5.9 4.18 STEAM GENERATOR (SG) TUBE ~~TUBING~~ SURVEILLANCE PROGRAM

~~Applicability~~ A1

~~Applies to the surveillance of tubing of each steam generator.~~

~~Objective~~

This program provides controls to ensure the A1

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

<INSERT 110jA> A12

~~Specification~~

~~4.18.1 Baseline Inspection~~

a. The first steam generator tubing 5.5.9.b and 5.5.9.c.1 inspection performed ~~according to~~ in accordance with ~~Specifications 4.18.2 and 4.18.3.a~~ shall be considered as constituting the baseline condition for subsequent inspections.

b. ~~4.18.2~~ Examination Methods:

1 ~~a~~. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end. A1

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

c. ~~4.18.3~~ Selection and Testing.

The steam generator sample size is specified in Table ~~4.18-1~~ 5.5.9-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.18-2~~ 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in ~~Specification 4.18.4~~ 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of ~~Specification 4.18.5~~ 5.5.9.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except: A1

<CTS INSERT 110jA>

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

5.5.9

<All changes> — A1

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

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

ii ~~b.~~ At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per ~~Specification 4.18.3.a.8.~~ 5.5.9.c.1.iii

A tube inspection (pursuant to ~~Specification 4.18.5.a.9~~ 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

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

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

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

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure ~~4.18.1~~ 5.5.9-1

iv ~~d.~~ Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to ~~4.18.4.~~ 5.5.9.d The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category of the OTSG.

v ~~e.~~ Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to ~~4.18.4.~~ 5.5.9.d Tubes with ODIGA identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with topical report BAW-10235P, Revision 1.

5.5.9

2. ~~1~~. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

(A1)

3. ~~2~~. The second and third sample inspections during each inservice inspection as required by Table ~~4.18-2~~ 5.5.9-2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

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

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

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

(2) Where special inspections are performed pursuant to 5.5.9.c.1.iii ~~4.18.3.a.3~~, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

(3) Where special inspections are performed pursuant to 5.5.9.c.2 ~~4.18.3.b~~ defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection.

(A1)

5.5.9

(ALL CHANGES) A1

~~4.18.4 Inspection Intervals~~

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

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

2. ~~b.~~ If the results of the inservice inspection of a steam generator performed in accordance with Table ~~4.18-2~~ 5.5.9-2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in ~~4.18.4.a~~ 5.5.9.d.1 and the interval can be extended to 40 months.

3. ~~c.~~ Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   i. ~~1.~~ Primary-to-secondary leakage in excess of the limits of Specification ~~3.1.6.3.b~~ 3.4.13 (Inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification ~~4.18.3.a.3~~ 5.5.9.c.1.iii, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.

   If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2.

   ii. ~~2.~~ A seismic occurrence greater than the Operating Basis Earthquake,

   iii. ~~3.~~ A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   iv. ~~4.~~ A main steam line or feedwater line break.

*A group of tubes means: (a) All tubes inspected pursuant to ~~4.18.3.a.3~~ 5.5.9.c.1.iii or
(b) All tubes in a steam generator less those inspected pursuant to ~~4.18.3.a.3~~

5.5.9

e.

~~4.18.5~~ **Acceptance Criteria**

1 ~~a.~~ As used in this specification:

i ~~1.~~ Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

ii ~~2.~~ Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

iii ~~3.~~ Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

iv ~~4.~~ Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve.

v ~~5.~~ % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

vi ~~6.~~ Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

vii ~~7.~~ Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply during Cycle 16 to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with topical report BAW-10235P, Revision 1.

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

The reroll repair process will only be used to repair tubes with defects in the upper tubesheet area. The reroll repair process will be performed only once per steam generator tube using a 1 inch roll length. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. The reroll repair process is described in the topical report, BAW-10232P, Revision 00.

viii ~~8.~~ Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, ~~or a steam line or feedwater~~ line break as specified in ~~Specification 4.18.4.c~~ 5.5.9.d.3 (A1)

ix ~~9.~~ Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the upper tubesheet, that portion of the tube above the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

5.5.9
5.6.7

unless otherwise noted: < ALL CHANGES = A1 >

2. ~~b.~~ The steam generator shall be determined operable after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table ~~4.18-2~~ 5.5.9-2

~~Tube 110/60 in the "A" steam generator contains indications in the upper roll transition that exceed the plugging limit. Tube 110/60 may remain in service with these indications for the duration of cycle 10 without rendering the "A" steam generator inoperable.~~ A16

5.6.7 ~~4.18.6~~ ~~Reports~~ Steam Generator Tube Surveillance

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

a. Number and extent of tubes inspected;

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

c. Identification of tubes plugged and tubes sleeved;

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

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

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

~~This report shall be in addition to a Special Report (per Specification 6.12.5.d) required for the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 4.18-2.~~ A4 The Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 5.5.9-2 prior to resumption of plant operation. The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.5.9

Bases A2

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

In general, steam generator tubes that are degraded beyond the repair limit can either be plugged, sleeved, or rerolled. The steam generator (SG) tubes that are plugged are removed from service by the installation of plugs at both ends of the associated tube and thus completely removing the tube from service. When the tube end cracking (TEC) alternate repair criteria is applied, axially-oriented indications found not to extend from the tube sheet cladding region into the carbon steel region may be left in service under the guidelines of topical report BAW-2346P, Rev. 0. When the upper tubesheet outer diameter intergranular attack (ODIGA) alternate repair criteria is applied, indications found within the defined region of the upper tubesheet may be left in service under the guidelines of topical report BAW-10235P, Revision 1. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. Following a SG inspection, an operational assessment is performed to ensure primary-to-secondary leak rates will be maintained within the assumptions of the accident analysis.

Degraded steam generator tubes can also be repaired by the installation of sleeves which span the area of degradation and serve as a replacement pressure boundary for the degraded portion of the tube, thus permitting the tube to remain in service.

Degraded steam generator tubes can also be repaired by the rerolling of the tube in the upper tubesheet to create a new roll area and pressure boundary for the tube. The rerolling methodology establishes a new pressure boundary below the degradation, thus permitting the tube to remain in service. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed in the upper tubesheet. The rerolling repair process will only be used to repair defects in the upper tubesheet in accordance with BAW-10232P, Revision 00.

All tubes which have been repaired using the reroll process will have the new roll area inspected during future inservice inspections. Defective or degraded tube indications found in the new roll and any indications found in the original roll need not be included in determining the Inspection Results Category for the generator inspection.

The reroll repair process will only be used to repair tubes with defects in the upper tubesheet area. The reroll repair process will be performed only once per steam generator tube using a 1 inch roll length. Thus, multiple applications of the reroll process to any individual tube is not acceptable. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. After the new roll area is initially deemed acceptable, future degradation in the new roll area will be analyzed to determine if the tube is defective and needs to be removed from service. The reroll repair process is described in the topical report, BAW-10232P, Revision 00.

5.5.9

TABLE ~~4.18-1~~ 5.5.9-1

A1

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

A1

TABLE ~~4.18-2~~ 5.5.9-2

STEAM GENERATOR TUBE INSPECTION [2,3]

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug, reroll, or sleeve defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug, reroll, or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G. ~~Special Report to NRC pursuant to 6.12.5.d~~	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes. ~~Special Report to NRC pursuant to 6.12.5.d~~	N/A	N/A

A4

NOTES: [1] $S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection. 5.5.9.c.1.iii

[2] For tubes inspected pursuant to ~~4.18.3.a.3~~: No action is required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a ~~Special~~ Report to NRC pursuant to ~~6.12.5.d~~ 5.6.7

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

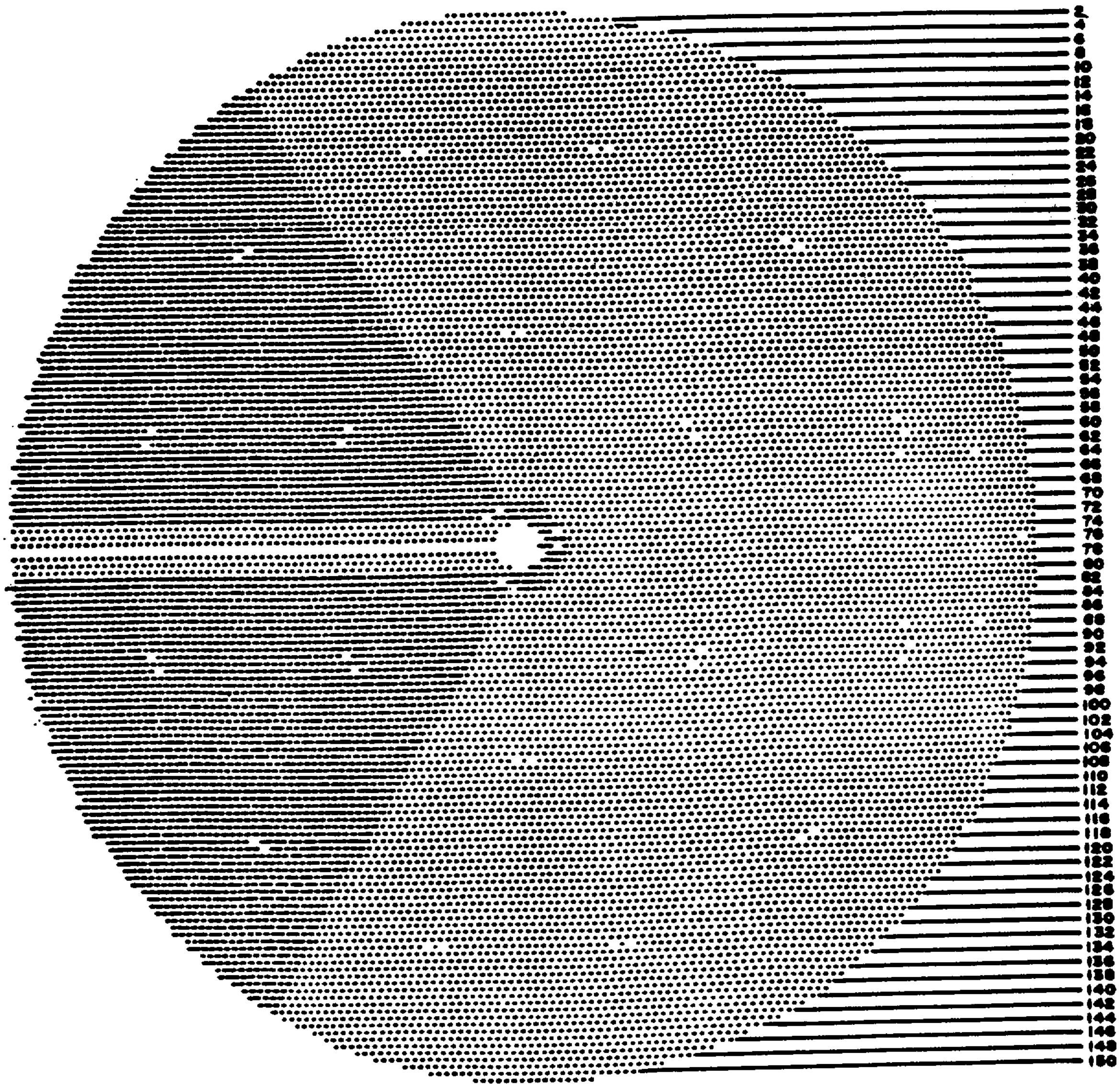
A1

5.5.9

5.5.9

Figure ~~4.18.1~~ 5.5.9-1

Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per Specification ~~4.18.3.a.8~~ 5.5.9.c.1.iii



DESCRIPTION	**TUBE COUNT**
GROUP A - 1: Lane region tubes as defined in ~~4.18.3.a.3(1)~~ 5.5.9.c.1.iii(1)	**382**
GROUP A - 3: Wedge shaped group depicted by darkened region of figure	**4880**

5.5.16

A1

4.26 REACTOR BUILDING PURGE VALVES

Applicability

This specification applies to the reactor building purge supply and exhaust isolation valves.

Objective

To assure reactor building integrity.

Specification

LATER

<LATER> (3.6) 4.26.1 The reactor building purge supply and exhaust isolation valves shall be determined closed at least once per 31 days when containment integrity is required by TS 3.6.1.

M8

5.5.16 4.26.2 Prior to exceeding conditions which require establishment of reactor building integrity per TS 3.6.1, the leak rate of the purge supply and exhaust isolation valves shall be verified to be within acceptable limits per TS 4.4.1, unless the test has been successfully completed within the last three months.

A2

Bases

Determination of reactor building purge valve closure will ensure that reactor building integrity is not unintentionally breached.

As a result of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," it was concluded that excess leakage past valve resilient seals is typically caused by severe environmental conditions and/or wear due to use. Recommended leak test frequencies of three months are deemed to be adequate to detect seal degradation of resilient seals.

The three month test need not be conducted with the precision of the Type C 10CFR50, Appendix J criteria, however the test must be sufficient to detect degradation.

5.0

4.27 DECAY HEAT REMOVAL

APPLICABILITY

<LATER> (3.4A)

Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal. — LATER

OBJECTIVE

To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.

SPECIFICATION

4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability.

4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2. — A5

<LATER> (3.4A)

4.27.3 The required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥20 inches on the startup range at least once per 12 hours. — LATER

4.27.4 The required reactor coolant loop(s) shall be determined operable by verifying the required loop(s) to be in operation and circulating reactor coolant at least once per 12 hours.

<LATER> (3.4A & 3.9)

4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours. — LATER

5.5.12

LA5

EGASTEMP

4.28 EXPLOSIVE GAS MIXTURE

Applicability

Applies to the Waste Gas System hydrogen/oxygen analyzers.

Objective

To prevent accumulation of explosive mixtures in the waste gas system.

Specification

4.28.1 The concentration of hydrogen/oxygen in the waste gas system shall be monitored continuously by either the primary or redundant waste gas analyzer during waste gas compressing operations to the waste gas decay tanks.

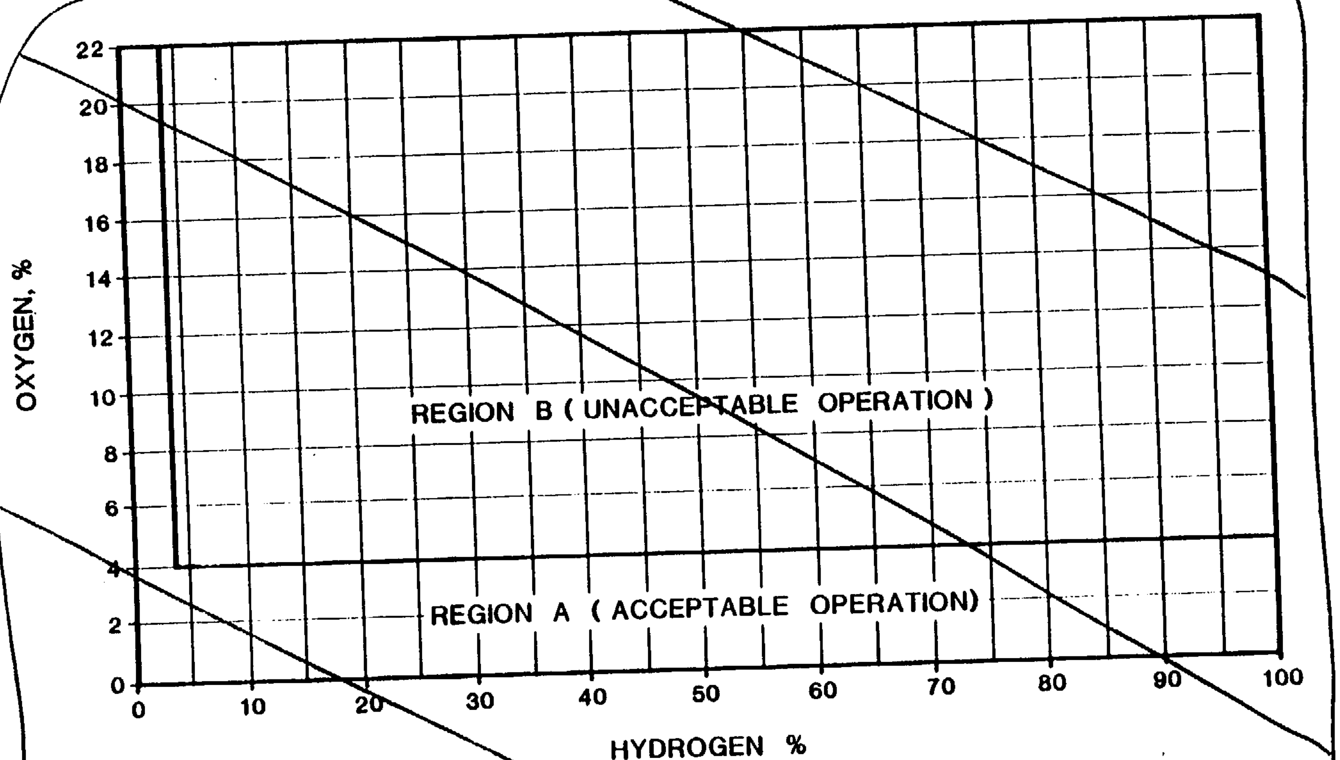
4.28.2 During waste gas system operation, with no H_2/O_2 analyzer in service, without delay suspend all additions of waste gas to the decay tanks or take grab samples for analysis every 4 hours during degassing operations, daily during other operations. The analysis of these samples shall be completed within 8 hours of taking the sample.

A2

Bases

This specification is to assure that the hydrogen/oxygen concentration will be kept within the limits in Figure 3.24-1 and therefore not enter the flammable region concentrations in the waste gas decay tanks.

Grab samples are to be taken every 4 hours during degassing operations when both hydrogen/oxygen analyzers are out of service. These samples are to be analyzed within 8 hours to assure that the hydrogen/oxygen concentration is within the limits in Figure 3.24-1. During other Waste Gas compressor operations, the hydrogen/oxygen concentration is not as subject to change, therefore grab samples are to be taken every 24 hours.



HYDROGEN – OXYGEN LIMITS FOR ANO – 1 WASTE GAS SYSTEM

FIGURE 3.2 4-1

LA5

EG&STRMP

5.5.12

5.5.12

LA5
EGrESTRPD

4.29 RADIOACTIVE EFFLUENTS

4.29.1 Radioactive Liquid Holdup Tanks

Applicability: At all times

Objective: To ensure that the limits of 10 CFR 20 are not exceeded.

Specification:

4.29.1 The quantity of radioactive material contained in an outside temporary radioactive liquid storage tank shall be determined to be within the limit of Specification 3.25.1 by analyzing a representative sample of the contents of the tank at least once per 7 days when radioactive materials are being added to the tank.

Bases:

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

A2

4.29.2 Radioactive Gas Storage Tanks

Applicability: At all times

Objective: To ensure meeting the requirements of Specification 3.25.2.

Specification:

4.29.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.25.2 at least once per 24 hours when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.1.4.1.b.

LAS

EGG STKMP

Bases:

This specification is provided so that the requirements of Specification 3.25.2 are met..

A2

5.1
5.2.1
5.2.2

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

5.1.1 ~~6.1.1~~ The ANO-1 plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 ~~6.1.2~~ An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is above the Cold Shutdown condition. With the unit not above the Cold Shutdown condition, an individual with an active SRO license or Reactor Operator license shall be designated as responsible for the control room command function. MODES 1, 2, 3, 4 A1

6.2 ORGANIZATION

5.2.1 ~~6.2.1~~ OFFSITE AND ONSITE ORGANIZATIONS

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

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

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

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

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

UNIT STAFF

5.2.2.f ~~6.2.2~~ The operations manager or the assistant operations manager shall hold a senior reactor operator license. ~~Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.~~

A5

5.2.2
5.3.1

5.2.2.e ~~6.2.2.1~~ Administrative controls shall be established to limit the amount of overtime worked by plant staff performing safety-related functions. These administrative controls shall be in accordance with the guidance provided by the NRC Policy Statement on working hours (Generic Letter 82-12).

6.3. FACILITY STAFF QUALIFICATIONS

ANS 3.1-1978

5.3.1 ~~6.3.1.~~ Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI ~~N18.1-1971~~ for comparable position, except for the designated radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. M1

~~6.4 DELETED~~

~~6.5 DELETED~~

A1

5.2.2

Table 6.2-1

ARKANSAS NUCLEAR ONE

MINIMUM SHIFT CREW COMPOSITION #

UNIT 1 (A1)

LICENSE CATEGORY	ABOVE COLD SHUTDOWN	COLD AND REFUELING SHUTDOWNS
5.2.2.b SOL	2	1* (A5)
OL	2	1
5.2.2.a NON-LICENSED	~~3~~ 2	1 (LA2) SAR

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operations after the initial fuel loading. (A5)

5.2.2.c #Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. (A1)

Additional Requirements:

1. At least one licensed Operator shall be in the control room when fuel is in the reactor. (A5)

2. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

5.2.2.d ~~3.~~ An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

4. All refueling operations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. (A5)

5.2.2.g 5. ~~When the unit is above the Cold Shutdown condition~~ an individual shall provide advisory technical support for the unit operations shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. (A1)

In MODES 1, 2, 3, and 4

5.4.1
5.5.1
5.5.3

~~6.6 DELETED~~ (A1)

~~6.7 SAFETY LIMIT VIOLATION~~

~~6.7.1 The following actions shall be taken in the event a Safety Limit is violated:~~ LATER

<LATER> (2.0)

~~a. The facility shall be placed in at least hot shutdown within one hour.~~

~~b. The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.~~

6.8 PROCEDURES AND PROGRAMS

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, ~~November, 1972~~. Rev 2, Feb 1978 (M3)

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety related equipment.~~ (A1)

~~d. (Deleted)~~

~~e. (Deleted)~~

5.4.1.c f. Fire Protection Program Implementation.

~~g. New and spent fuel storage.~~ (A1)

5.4.1.d (5.5.1) h. Offsite Dose Calculation Manual ~~and Process Control Program~~ implementation at the site. (A1)

<ADD: 5.4.1.d for "other programs">
<ADD: 5.4.1.b for EOPs per GL 82-33> (M3)

(A15)

LAR

5.5.16

6.8.2 (Deleted) A1

6.8.3 (Deleted)

5.5.16 ~~6.8.4~~ The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained: A1

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

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

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of ~~containment~~ reactor building air weight per day at P_a. A1

Reactor building leakage rate acceptance criteria is ≤ 1.0 L_a. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are ≤ < 0.60 L_a for the Type B and Type C tests and ≤ < 0.75 L_a for Type A tests. A11

The provisions of ~~Specification 4.0.2~~ SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program. A1

The provisions of ~~Specification 4.0.3~~ SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program. A1

5.5.4 6.8.5 **The Radioactive Effluent Controls Program shall be established, implemented, and maintained:**

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

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

b. **Limitations on the concentrations of radioactive material released ~~in liquid effluents to unrestricted~~ areas conforming to 10 CFR 20, Appendix B, Table II, Column 2;** ⟨INSERT 127a A⟩ A17

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

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

e. **Determination of cumulative ~~and projected~~ dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;** ⟨INSERT 127a B⟩ A17

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

g. **Limitations on ~~the dose rate~~ from the site resulting from radioactive material released in gaseous effluents to areas at or beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;~~** A1 ⟨INSERT 127a C⟩ A17

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

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

j. **Limitations on ~~the annual dose or dose commitment~~ to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.** A1

⟨INSERT 127a D⟩ A17

<CTS INSERT 127aA>

... ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.

<CTS INSERT 127aB>

. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM ...

<CTS INSERT 127aC>

... shall be in accordance with the following:

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

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

<CTS INSERT 127aD>

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

5.0

6.9 DELETED

(A1)

5.7

A5

~~6.10 RADIATION PROTECTION PROGRAM~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

5.7 **6.11 HIGH RADIATION AREA**

5.7.1
5.7.1.a
5.7.1.b
<INSERT 129 A>
<INSERT 129 B>
5.7.1.d

6.11.1 In lieu of the "control device" or "alarm signal" required by ~~paragraph 20.203(c)(2) of~~ 10CFR20, each high radiation area (as defined in 20.202(b)(3) of 10CFR20) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and shall be controlled by requiring the issuance of a radiation work permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following: A1 L2

5.7.1.d.1 ~~a.~~ A radiation monitoring device which continuously indicates the radiation dose rate in the area.

5.7.1.d.2 ~~b.~~ A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them. L2 A1

<INSERT 129 C>
5.7.1.e
<INSERT 129D>
only

5.7.1.d.4.i ~~c.~~ An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation work permit.

<Note: this text is repeated in insert 129C>

<CTS INSERT 129A>

Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

<CTS INSERT 129B>

... or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

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

<CTS INSERT 129C>

..., with an appropriate alarm setpoint, or

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

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

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

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

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, ...

<CTS INSERT 129D>

... These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7

5.7.2 ~~6.11.2~~ ~~The requirements of 6.11.1 above, shall also apply to each~~ high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and access to these areas shall be maintained under the administrative control of the shift supervisor on duty and/or the designated radiation protection manager.

<INSERT 129a A>

L2

<CTS INSERT 129aA>

... at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

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

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

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

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

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

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

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

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

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

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

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

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

<CTS INSERT 129aA>
(continued)

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

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

5.6
5.6.1

6.12 REPORTING REQUIREMENTS

6.12.1 ~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the appropriate NRC Regional Office unless otherwise noted.~~ A5

6.12.2 Routine Reports

~~6.12.2.1 Startup Report~~

~~A summary report of plant startup and power escalation testing shall be submitted following 1) receipt of an operating license, 2) amendment to the license involving a planned increase in power level, 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and 4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~Startup reports shall be submitted within 1) 90 days following completion of the startup test program, 2) 90 days following resumption or commencement of commercial power operation, or 3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

LA6
TRM

5.6.1 6.12.2.2 Occupational Exposure Data Report 1/ L3a

An Occupational Exposure Data Report for the previous calendar year shall be submitted ~~prior to March 1~~ by April 30 of each year. The report shall contain a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) , for whom monitoring was performed, receiving ~~exposures~~ an annual deep dose equivalent greater than 100 mrem/~~yr~~ and their associated ~~man~~ rem ~~exposure~~ collective deep dose equivalent (reported in person-rem) according to work and job functions, 2/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling.

A13

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of ~~20.407 of~~ 10 CFR ~~Part~~ 20.2206.

4| 5.6.1 Note....

5.6.1
5.6.2
5.6.3

ionization chamber electronic dosimeter

5.6.1 The dose assignments to various duty functions may be estimates based on pocket ~~dosimeter~~, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total ~~whole body dose~~ received from external sources shall be assigned to specific major work functions. A13

deep dose equivalent

5.6.4 6.12.2.3 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis by the 15th of each month following the calendar month covered by the report.

~~6.12.2.4 Annual Report~~

~~All challenges to the pressurizer electromatic relief valve (ERV) and pressurizer safety valves shall be reported annually.~~ L3b

5.6.2 6.12.2.5 Annual Radiological Environmental Operating Report *

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

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

in the previous year

5.6.3 6.12.2.6 Radioactive Effluent Release Report **

prior to May 1 of each year

The Radioactive Effluent Release Report covering the operation of the unit shall ~~be submitted~~ in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1. A14

5.6.2 NOTE * A single submittal may be made for ANO. The submittal should combine those sections that are common to both units.

shall

5.6.3 NOTE ** A single submittal may be made for ANO. The submittal ~~should~~ combine those sections that are common to both units. The submittal shall specify the releases of radioactive material from each unit.

5.6.5

5.6.5 6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

2.1	Safety Limits, Reactor Core - Axial Power Imbalance protective limits and Variable Low RCS Pressure-Temperature Protective Limits
2.3.1	Reactor Protection System trip setting limits - Protection System Maximum Allowable Setpoints for Axial Power Imbalance and Variable low RC system pressure
3.1.8.3	Minimum Shutdown Margin for Low Power Physics Testing
3.5.2.1	Allowable Shutdown Margin limit during Power Operation
3.5.2.2	Allowable Shutdown Margin limit during Power Operation with inoperable control rods
3.5.2.4	Quadrant power Tilt limit
3.5.2.5	Control Rod and APSR position limits
3.5.2.6	Reactor Power Imbalance limits

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A5

Note: Next non-blank page is 146

5.6.6

5.6.6 ~~6.12.4~~ Reactor Building Inspection Report

~~6.12.4.1~~ Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The reportshall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

5.6.7

5.6 6.12.5 Special Reports

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. A5

a. Deleted A1

<LATER> (3.3D) b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LATER

c. Deleted A1

5.6.7 d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification ~~4.2.~~ 5.5.9

<LATER> (3.7) e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2. LATER

f. Deleted

g. Deleted

h. Deleted

i. Deleted A1

<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.

5.5.14
5.5.15

6.13 (DELETED) A1

Add ITS 5.5.14,
"Tech Spec Bases Control Program" M7

Add ITS 5.5.15,
"Safety Function Determination Program M7

5.5.1

5.5.1

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports ~~required by Specifications 6.12.2.5 and 6.12.2.6~~. A1

Licensee initiated changes to the ODCM:

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

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

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

b. Shall become effective after approval of the ~~General Manager, Plant Operations~~ PNO; and LA1 SAR

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

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 5.0: Administrative Controls

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:

5.0 L1 Not used

5.0 L2

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The controls for access to a high radiation area are not considered as initiators, nor as a mitigation factor, in any previously evaluated accident. Therefore, the 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?

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the 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 requirements for control of high radiation areas provide for the use of alternates to the "control device" or "alarm signal" requirements of 10 CFR 20.1601. This change provides such alternative methods for controlling access. These methods and additional administrative requirements have been determined to provide adequate controls to prevent unauthorized and inadvertent access to such areas. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>5.0 L3</u>

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 date for 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.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

5.0 L4

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The DC Sources are used to support mitigation of the consequences of an accident. Equipment powered by the DC Sources continues to be evaluated for loss of function, and previously determined appropriate ACTIONS for such inoperabilities continue to be required. Experience with the reliability of the DC sources indicates that the proposed increase in the Completion Time will not significantly increase the probability of a loss of electric power accident or of any other accident previously evaluated. The proposed ACTION continues to provide adequate assurance of OPERABLE required equipment and therefore, 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 corrective actions are taken to restore plant systems to OPERABLE status, as assumed 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?

This change does not involve a significant reduction in a margin of safety since the OPERABILITY of the equipment and loss of function continue to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the equipment to OPERABLE status, rather than requiring a shutdown transient.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

5.0 L5

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

CTS Table 4.1-2, item 12 requires verification of a flow limiting gap that exists between the main feedwater line pipe and an annulus attached to the reactor building penetration on a periodic Frequency. The circular plate which provides the flow limitation is welded to the penetration and not subject to fluctuation except due to radial expansion during heatup which is considered in the design. Therefore, a change in the Frequency to require this verification following any modifications which may affect the required gap continues to provide adequate assurance of this design feature. Additionally, this verification is removed from the Technical Specifications since it is a specific design feature of a structure which is only subject to change via the design change process. As such, the "post-modification" verifications are also required by the design change process, and as with other post-modification type requirements, can be removed from the Technical Specifications without a significant impact on safety. This change does not result in any changes in hardware or methods of operation. Neither the flow limiting gap, nor the hardware which provides the gap is assumed to initiate an accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated. The design does provide for mitigation of a design basis pipe break to limit the consequences. However since the gap is provided by a design feature which is only subject to change by the design change process, periodic verification is unnecessary. Further, removal of this requirement from the Technical Specifications does not change the hardware, nor remove the design controls in place. 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 impact only the administrative requirements for periodic verification of a design feature 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 periodic verification of this structural design feature since it is only subject to change by the design change process. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

5.0 L6

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The testing of diesel generator fuel oil is not considered an initiator, or a mitigating factor, in any previously evaluated accident. Therefore, the 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?**

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the 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 testing of stored diesel generator fuel oil is revised to require the periodic testing of the stored fuel oil only for particulates (replacing the periodic testing per ASTM-D975) once every 31 days. The change reflects industry-standard acceptable DG fuel oil testing programs. Over the storage life of ANO-1 DG fuel oil, the properties tested by ASTM-D975 are not expected to change and performing these tests once on the new fuel oil (see DOC M9) provides adequate assurance of the proper initial quality of fuel oil. The periodic testing for particulates monitors a parameter that reflects degradation of fuel oil and can be trended to provide increased confidence that the stored DG fuel oil will support DG operability. Therefore, this change does not involve a significant reduction in a margin of safety.

ITS DISCUSSION OF DIFFERENCES

ITS Section 5.0: Administrative Controls

1 NUREG 5.1.1, 5.2.1, 5.2.2, & 5.5.1 - Incorporates TSTF-065, Rev 1.

Unit specific changes consistent with current license basis include:

1) The ANO-1 unit specific designator is added to clearly establish the separate requirements that exist for ANO-1 and ANO-2. This prevents possible misinterpretation that the same individual may occupy this position for both ANO units.
2) Current Technical Specifications (CTS) Table 6.2-1 "additional requirement" number 3 is retained in Improved Technical Specifications (ITS) 5.2.2.c as "an individual qualified in radiation protection procedures." This maintains the greater flexibility provided by the CTS for fulfilling this position requirement.

2 NUREG 5.2.2 - In the discussion of Unit Staff (ITS 5.2.2), plant specific clarifications are provided to reflect the station two unit design, and that the two units share a common control room envelope, but the control rooms are separated. Unit specific terminology is incorporated to clarify applicability of requirements on a unit specific basis since the unit operations staff is assigned in this manner (i.e., to either ANO-1 or ANO-2), and a specific identification is provided for the applicable column of the table in 10 CFR 50.54(m)(2)(i). The shift manning requirements for "one unit, one control room" are considered to be applicable to each unit at ANO on an individual basis due to the dissimilarity of design of the units. ANO does not attempt to license individuals on both units simultaneously. These changes are consistent with current license basis.

3 NUREG 5.4.1 - An additional clarification is provided in proposed ITS 5.4.1.b to identify the appropriate discussions of emergency operating procedure requirements in Generic Letter 82-33. This change involves no revision of the actual requirements since Section 7 is the only portion of the identified Generic Letter which requires upgrades to the emergency operating procedures. Rather the change provides an editorial clarification to prevent possible misinterpretation of requirements to provide emergency operating procedures for all items identified in the Generic Letter. This change is consistent with current license basis.

4 Not used.

5 NUREG 5.4.1 - The NUREG 5.4 requirements to establish, implement and maintain written procedures covering the activity of "quality assurance for effluent and environmental monitoring" are not adopted. Procedures for effluent and environmental monitoring are required by 10 CFR 50 and Appendix I of Part 50. The QAPM is considered applicable to the implementation procedures for effluent and environmental monitoring for the station. Further, this activity is appropriately addressed in the station Environmental Report (ER) with the following statements: "Radiological analytical methods used in the radiological monitoring program are described in approved procedures as required by the Quality Assurance program for operations.

Additionally, procedure implementation and records are subject to periodic audit by the Quality Assurance Organization." (Ref. ANO-2 ER Section 6.1; Note that the ER is applicable to the site and thus appropriate for both units 1 and 2.) This periodic audit function continues to be implemented through the current QAPM Section 18.3.2.f which provides for periodic audits of the Radiological Environmental Monitoring Program. These controls are considered sufficient since they are not directly pertinent to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Since these details are also not necessary to adequately describe the pertinent regulatory requirement, they are not mandated by 10 CFR 50.36, and they do not meet the criteria in 10 CFR 50.36, they can be appropriately retained in licensee controlled documents without a significant impact on safety. Retaining these requirements in controlled documents also provides adequate assurance that they will be maintained. Changes to the QAPM are controlled by 10 CFR 50.54. Since the controls are consistent with the QA controls for other activities, the specific listing for effluent and environmental monitoring is unnecessary.

6 NUREG 5.3.1 – NUREG 5.3.1 was revised to reflect CTS 6.3.1 requirements for staff qualification. These changes are consistent with current license basis and QAPM.

7 NUREG 5.5.1 - The RSTS cross reference to other Specifications is not adopted in ITS 5.5.1.b. This is a simple editorial change in presentation which has no impact on the actual requirement. Typically, cross references are not provided in the ITS, and this change is made to provide consistency both within the proposed Specifications and with the previously approved ITS for other EOI stations, i.e., Grand Gulf and River Bend.

8 NUREG 5.5.5 - The program identified in NUREG 5.5.5, "Component Cyclic or Transient Limit," is not adopted for the ANO-1 ITS. This program is currently administratively controlled (Procedure 1010.002) and the limits are addressed in the SAR (and therefore changes are controlled pursuant to 10 CFR 50.59). This is considered adequate for these design limits, and they are therefore, proposed to continue to be so controlled. This change is consistent with current license basis.

9 NUREG 5.5.7 – This change incorporates the CTS 4.2.6 requirements for the Reactor Coolant Pump Flywheel Inspection Program as ITS 5.5.7. ANO-1 is not committed to the requirements of Regulatory Guide 1.14, Revision 1, as stated in the NUREG. Therefore, the current ANO-1 surveillance requirements have been retained. In addition, an SR 3.0.2/SR 3.0.3 applicability statement has been added. The current ANO-1 requirements allow the application of the CTS 4.0.2 and CTS 4.0.3 provisions (which correspond to SR 3.0.2 and SR 3.0.3, respectively) to the CTS 4.2.6 requirements. These changes maintain the current ANO-1 licensing basis in ITS 5.5.7.

10 NUREG 5.5.16 – This change incorporates the CTS 6.8.4 requirements for the Reactor Building Leakage Rate Testing Program as ITS 5.5.16. The ITS Program is virtually identical to CTS requirements with the exception of the following:

1) A minor change was made to correct the reference to ITS SR 3.0.2 and SR 3.0.3 in lieu of CTS 4.0.2 and 4.0.3.
2) The CTS 4.26.2 requirement for leak rate testing of the reactor building purge valves was inserted into the ITS 5.5.16 program. This action consolidates CTS requirements for leak rate testing.

These changes are either editorial or are consistent with current license basis.

11 NUREG 5.5.8 – Incorporates TSTF-279.

12 NUREG 5.5.10 - The Secondary Water Chemistry Program is proposed to be revised to be consistent with the content of the current Operating License Condition 2.C(7) which does not include evaluation of the chemistry results for potential low pressure turbine disc stress corrosion cracking. An evaluation of the secondary water chemistry to maximize the turbine availability is currently accomplished under administrative controls (Procedure 1000.042) and is proposed to continue to be so controlled. This change is consistent with current license basis.

13 NUREG 5.5.11 - The Ventilation Filter Testing Program is proposed to be revised to be consistent with the content of the CTS for testing of HEPA and charcoal filters in safety related ventilation systems. Additionally, item e of the NUREG is not adopted since no heaters are provided in the design of these systems. These changes are consistent with current license basis.

14 NUREG SR 3.8.3.2 Bases – The discussion of the new fuel oil testing referencing "clear and bright" is revised. ANO fuel oil is supplied with added dye, which precludes appropriate "clear and bright" testing. In its place is supplied a reference to the currently utilized "water and sediment" testing of ASTM-D975.

15 Not used.

16 NUREG 5.5.9 & 5.5.13 – Incorporates TSTF-118.

17 NUREG Section 5.6 leads in with a statement about making submittals in accordance with 10 CFR 50.4. Many of the reports addressed are submitted in accordance with Part 20 and are not governed by 50.4. Since this statement is not part of CTS, it is removed from ITS.

18 NUREG 5.6.6 - The NUREG report for the reactor coolant system pressure and temperature limits is not adopted for the ITS. These limits will continue to be provided in the appropriate Limiting Conditions for Operation (LCO) (refer to ITS 3.4.3, "RCS Pressure and Temperature (P/T) Limits"). This change is consistent with current license basis.

19 NUREG 5.6.7 - Incorporates TSTF-037, Rev. 2.

20 NUREG 5.6.8 - The NUREG 5.6.8 reporting requirements related to post accident monitor inoperability are not proposed to be specifically identified in Section 5.0 of the ITS. A Special Report will continue to be required by the ACTIONS for the Post Accident Monitoring Instrumentation LCO (ITS 3.3.15 Required Action B.1), but details for content of the report will be provided only in the associated Bases for the Required Action. These controls are considered sufficient since they are not directly pertinent to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Since the details of the report are also not necessary to fulfill the pertinent regulatory requirement, they are not mandated by 10 CFR 50.36, and they do not meet the criteria in 10 CFR 50.36, they can be appropriately retained in licensee controlled documents without a significant impact on safety. Retaining these requirements in controlled documents also provides adequate assurance that they will be maintained. Changes to the Bases are controlled by the proposed program in the Administrative Controls Section of the ITS. Additionally, this change is consistent with previously approved ITS for other EOI stations, i.e., Grand Gulf and River Bend.

21 Not used.

The example provided in the NUREG 5.7.1 of individuals qualified in radiation protection procedures "(e.g., Health Physics Technicians)" is not incorporated. This example is unnecessary and is considered likely to be interpreted as more limiting than intended since other individuals may be qualified in radiation protection procedures. This change is consistent with current license basis.

NUREG 5.7.1.c is revised to retain the CTS 6.11.1.c requirements by deleting reference to the Radiation Protection Manager. This change is consistent with current license basis.

22 NUREG 5.5.2 - The listing of systems which are considered Primary Coolant Sources Outside Containment (NUREG 5.5.2) is not incorporated. The systems to which the program is applied have been previously identified in response to NUREG-0578 item 2.1.6.a. The application is adequately controlled through the design modification process and application of 10 CFR 50.59. Therefore, the list of systems to which the program is applied is not included in the CTS and is proposed to continue to be administratively controlled. This change is consistent with current license basis.

23 NUREG 5.1.1 - The requirement for approval of each proposed test, experiment, or modification to systems or equipment that affect nuclear safety by the [Plant Superintendent] is not adopted. CTS Sections 6.5 and 6.8 were previously revised (Amendment No. 179 dated April 25, 1995) to eliminate this detail. Approval requirements for such procedures and modifications are delineated in the QAPM as discussed in the request for and approval of this recent amendment. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

24 NUREG 5.2.2 - The requirements of 10 CFR 50.54(m)(2)(iii) and 50.54(k) adequately provide for this shift manning. These regulations, 50.54(m)(2)(iii), require "when a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's Technical Specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." Further, 10 CFR 50.54(k) requires "an operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility." The NUREG 5.2.2.b requirements will be met through compliance with these regulations and is not required to be re-iterated in the ITS. This change is consistent with TSTF-258, Rev 4, with one exception. 10 CFR 55.4 provides a definition for the phrase "actively performing the functions of an operator or senior operator," for the purposes of operator proficiency, as "an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications,..." Since this 10 CFR 55.4 definition appears to require a facility to define those positions on the shift crew that are credited for gaining or maintaining the skills associated with performing licensed activities, a statement requiring adherence to the minimum shift composition of 10 CFR 50.54(m)(2)(i) has been added.

25 NUREG 5.5.10 - The specific identification of the ITS 5.5.10.c requirement to include "monitoring the discharge of the condensate pumps for evidence of condenser inleakage" is not adopted. The program can adequately control these details as demonstrated by the implementation of the current Operating License Condition 2.C(7). This change is consistent with current license basis.

26 NUREG 5.6.1 & 5.6.3 – Incorporates TSTF-152.

27 NUREG 5.2.2 - The ITS 5.2.2.g requirements for a Shift Technical Advisor (STA) on the unit staff are clarified to indicate that the STA is only required in MODES 1, 2, 3, and 4. This is consistent with CTS Table 6.2-1. This change is consistent with current license basis.

28 NUREG 5.2.2 - The introductory phrase "The unit staff organization shall include the following:" is omitted. This phrase provides no requirements or clarification, and implies that "the following" is intended to be a listing of required organizational elements. However, also included are general requirements for the staff, e.g., absence and overtime limitations, etc. Therefore, the introductory phrase is not appropriate.

29 NUREG 5.1.2 - The identification of the "Shift Supervisor" as responsible for the control room command function is not consistent with the current practice as ANO and is not adopted. The "command and control" functions are currently assigned to a Control Room Supervisor who is not limited to the area of the control room envelope. A Shift Superintendent is also provided who implements many of the functions of the NUREG "Shift Supervisor" and who typically remains in the control room. Further, the command structure is adequately controlled by procedures and "turnover" requirements in the ITS are unnecessary. These changes are consistent with the current license basis.

30 Not used.

31 NUREG 5.5.3 is modified to reflect CTS requirements for sampling of radioactive "iodine".

32 Not used.

33 NUREG 5.2.1 & 5.2.2 - A change similar to TSTF-065, Rev 1, and portions of TSTF-258, Rev 4, is included for the "specified corporate executive position" in ITS 5.2.1.c. Also, the ITS 5.2.2.g discussion is revised so that it does not imply that the STA and the "shift supervisor" must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the shift supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. However, the NUREG wording of "the STA shall provide ... support to the Shift Supervisor..." is considered to be easily misinterpreted to require separate individuals. Therefore, the wording is revised so that the STA function may be provided by either a separate individual or the individual who also fulfills another role in the shift command structure. This is consistent with CTS Table 6.2-1. This change is consistent with current license basis.

34 NUREG 5.5.12 – NUREG 5.5.12 is revised to match the CTS 3.25.1 and 4.29.1 which address only temporary outdoor liquid radwaste tanks. Additionally, an editorial clarification is made in the description of the limits for these tanks to match the Bases for the CTS, i.e., the radioactivity must be "less than the amount that would result in concentrations equal to the limits...," rather than an amount that would be "less than the amount that would result in concentrations less than the limits...." These revisions result in no functional differences in the requirements. Note that this entire Program is bracketed in NUREG-1430.

ITS DISCUSSION OF DIFFERENCES

35 NUREG 5.5.9 – The CTS 4.18 requirements for Steam Generator (SG) Tube Inspection are incorporated into the ITS as specified in the NUREG 5.5.9 Reviewer's Note. The following discuss the required changes:

1) Minor reformatting was necessary to establish consistency with the NUREG.
2) A note that reporting requirements were relocated from CTS 4.18.6 to ITS 5.6.7 was added for clarification.
3) TSTF-118 was incorporated which added a statement that ITS SR 3.0.2 is applicable to the SG Tube Surveillance Program inspection Frequencies.

These changes were in accordance with NUREG guidance, TSTF-118 or were editorial with no change in license basis requirements.

36 NUREG 5.6.10 – The CTS 4.18.6 and CTS 6.12.5.d requirements for the Steam Generator Tube Surveillance Report were incorporated into ITS 5.6.7 as directed the NUREG 5.6.10 Reviewer's Note. Minor reformatting was necessary and cross-reference numbers were changed to accurately reflect the ITS location of the requirements. No relaxation of requirements exists as a result of this change. This change is consistent with NUREG-1430 direction and current license basis.

37 NUREG 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," is not incorporated in the ANO-1 ITS. The license amendment #199 revised the reactor building structural integrity requirements to relocate this program from the ANO-1 CTS. NUREG 5.6.9 is revised to reflect the Reactor Building Inspection Report consistent with CTS 6.12.4.

38 NUREG 5.5.13 – Incorporates TSTF-106, Rev 1.

39 NUREG 5.5.15 – Incorporates TSTF-273, Rev 2.

40 NUREG 5.5.2 – Incorporates TSTF-299.

41 NUREG 5.5.4 – Incorporates TSTF-308.

42 NUREG 5.5.4, 5.6.4 & 5.7 – Incorporates TSTF-258, Rev 4. Two editorial changes are reflected in the markup of the Section 5.7 Insert (which is from TSTF-258, Rev 4). The addition of a comma in 5.7.1.b clarifies that the added detail applies to the "equivalent" means. Paragraph 5.7.2.d.3(ii) ends with phrasing that is editorially different than the same requirement found in paragraph 5.7.1.d.4(ii). The ending phrasing used in 5.7.1 is utilized in 5.7.2.

CTS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

ANO-1 plant manager

5.1.1 The ~~[Plant Superintendent]~~ shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. 1 6.1.1

~~The [Plant Superintendent] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.~~ 23

5.1.2 ~~The [Shift Supervisor (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room~~ while the unit is in MODE 1, 2, 3, or 4, ~~an~~ An individual with an active Senior Reactor Operator (SRO) license shall be designated ~~to assume~~ as responsible for the control room command function. ~~During any absence of the [SS] from the control room while the unit is in MODE 5 or 6, an~~ With the unit not in MODE 1, 2, 3 or 4, an individual with an active SRO license or Reactor Operator license shall be designated ~~to assume~~ as responsible for the control room command function. 6.1.2 29

CTS

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

6.2.1

5.2.1 Onsite and Offsite Organizations

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

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the ~~[FSAR]~~; Safety Analysis Report (SAR) — 6.2.1.a — 1

INSERT 5.0-2A

b. The ~~[Plant Superintendent]~~ ANO-1 plant manager shall be responsible for overall safe operation of the ~~plant~~ unit and shall have control over those onsite activities necessary for safe operation and maintenance of the ~~plant~~; unit — 6.2.1.b edit

c. ~~The~~ [a] A specified corporate executive ~~position~~ shall have corporate responsibility for overall ~~plant~~ unit nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, ~~maintaining, and~~ providing technical support to the ~~plant~~ to ensure nuclear safety~~; and~~. — 6.2.1.c edit — 33

INSERT 5.0-2B

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures. — 6.2.1.d

5.2.2 Unit Staff

~~The unit staff organization shall include the following:~~ — 28

a. A non-licensed operator shall be ~~assigned to each reactor containing~~ on site when fuel is in the reactor, and an additional non-licensed operator — 6.2.2 Table 6.2-1 — 2

(continued)

<INSERT 5.0-2A>

, including the unit specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications,

<INSERT 5.0-2B>

The specified corporate executive shall be identified in the SAR; and

CTS

5.2 Organization

5.2.2 Unit Staff (continued)

on site when the

shall be ~~assigned for each control room from which a~~ reactor is ~~operating~~ in MODES 1, 2, 3, or 4.

~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~

2

b. ~~At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~

<INSERT 5.0-3A>

24

2

6.2.2 Table 6.2-1

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

for one unit, one control room,

1

6.2.2 Table 6.2-1

d. ~~A Health Physics Technician~~ shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

An individual qualified in radiation protection procedures

e. ~~Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).~~

~~Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~

(continued)

<INSERT 5.0-3A>

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

CTS

5.2 Organization

5.2.2 Unit Staff (continued)

1. ~~An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;~~

2. ~~An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;~~

3. ~~A break of at least 8 hours should be allowed between work periods, including shift turnover time;~~

4. ~~Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.~~

~~Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.~~

~~Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.~~

~~OR~~

e. The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). 6.2.2.1

f. The ~~[~~Operations ~~M~~anager or ~~A~~ssistant ~~O~~perations ~~M~~anager~~]~~ shall hold an SRO license. 6.2.2 1

g. In MODES 1, 2, 3 or 4, an individual ~~The Shift Technical Advisor (STA)~~ shall provide advisory technical support for the unit operations shift crew ~~to the Shift Supervisor (SS)~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~In addition, the STA~~ This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. 27 33 6.3.1 Table 6.2-1

CTS

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

~~Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8]~~ ANSI ANS 3.1-1978 for comparable positions except for the designated radiation protection manager, who shall meet or exceed the minimum qualifications of ~~[Regulations,~~ Regulatory Guide~~s, or ANSI Standards acceptable to NRC staff]~~ 1.8, September 1975. 6.3.1

6

CTS

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; 6.8.1.a,b,c,g N/A

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33; N/A 3

~~c. Quality assurance for effluent and environmental monitoring;~~ 5

c. Fire Protection Program implementation; and 6.8.1.f

d. All programs specified in Specification 5.5. 6.8.1.h N/A

5.0 ADMINISTRATIVE CONTROLS

CTS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained. 6.14

5.5.1 Offsite Dose Calculation Manual (ODCM)

edit

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

edit

~~b.~~ The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports ~~required by Specification [5.6.2] and Specification [5.6.3]~~. 7

Licensee initiated changes to the ODCM:

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

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

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

b. Shall become effective after the approval of the ~~[Plant Superintendent]~~ ANO general manager; and 1

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the

(continued)

CTS

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

LC 2.C(5)

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. [The systems include [Low Pressure Injection, Reactor Building Spray, Makeup and Purification, and Hydrogen Recombiner].] The program shall include the following: (22)

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at ~~refueling cycle intervals or less~~ least once per 18 months. The provisions of SR 3.0.2 are applicable. (40)

LC 2.C(6)
6.8.1.i

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodine ~~gases~~, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following: (31)

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

6.8.5

5.5.4 Radioactive Effluent Controls Program

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

(continued)

5.5 Programs and Manuals

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5.5.4 Radioactive Effluent Controls Program (continued) 6.8.5

be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM; 6.8.5.a

ten times the consentration values in

to 10 CFR 20.1001 – 20.2402

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20~~ Appendix B, Table 2, Column 2; 6.8.5.b 42

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM; 6.8.5.c

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I; 6.8.5.d

e. Determination of cumulative ~~and projected~~ dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days; 6.8.5.e 41

<INSERT 5.0-9A>

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I; 6.8.5.f

from the site   at or

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1~~; 6.8.5.g 42

<INSERT 5.0-9B>

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; 6.8.5.h

(continued)

<INSERT 5.0-9A>

Determination of projected dose contributions from the radioactive effluents in accordance with the methodology in the ODCM ...

<INSERT 5.0-9B>

... shall be in accordance with the following:

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

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

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, ~~conforming to 10 CFR 50, Appendix I~~; and 6.8.5 i

j. Limitations on the ~~annual dose or dose commitment~~ (beyond the site boundary) to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to ~~40~~ CFR 190. 6.8.5 j 42

<INSERT 5.0-10A>

5.5.5 ~~Component Cyclic or Transient Limit~~ (Not used.) 8

~~This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

5.5.6 ~~[Pre-Stressed Concrete Containment Tendon Surveillance Program~~ (Not used.) 37

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]~~

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel ~~per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~ 9

<INSERT 5.0-10B>

(continued)

<INSERT 5.0-10A>

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

<INSERT 5.0-10B>

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

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

CTS

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program 4.0.5

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components, ~~including applicable supports.~~ The program shall include the following: 11

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

edit

ASME ~~Boiler and Pressure Vessel~~ Code ~~and applicable Addenda~~ terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

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

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

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

5.5.9 Steam Generator (SG) Tube Surveillance Program 4.18

~~Reviewer's Note: The Licensees current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

<INSERT 5.0-11A> 35 16

(continued)

<Insert 5.0-11A (SG Tube Inspection Program)>

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

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Tube Surveillance Program inspection frequencies. 16

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

b. Examination Methods:

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

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

c. Selection and Testing

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

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

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

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

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

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

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

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

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

iv. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to 4.18.4. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category of the OTSG.

v. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to 4.18.4. Tubes with ODIGA identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with topical report BAW-10235P, Revision 1.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

iv. A main steam line or feedwater line break.

e. Acceptance Criteria:

1. Terms as used in this program:

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

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

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

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

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

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

vii. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply during Cycle 16 to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with topical report BAW-10235P, Revision 1.

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

The reroll repair process will only be used to repair tubes with defects in the upper tubesheet area. The reroll repair process will be performed only once per steam generator tube using a 1 inch roll length. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. The reroll repair process is described in the topical report, BAW-10232P, Revision 00.

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

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

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

TABLE 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION [2,3]

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

NOTES:

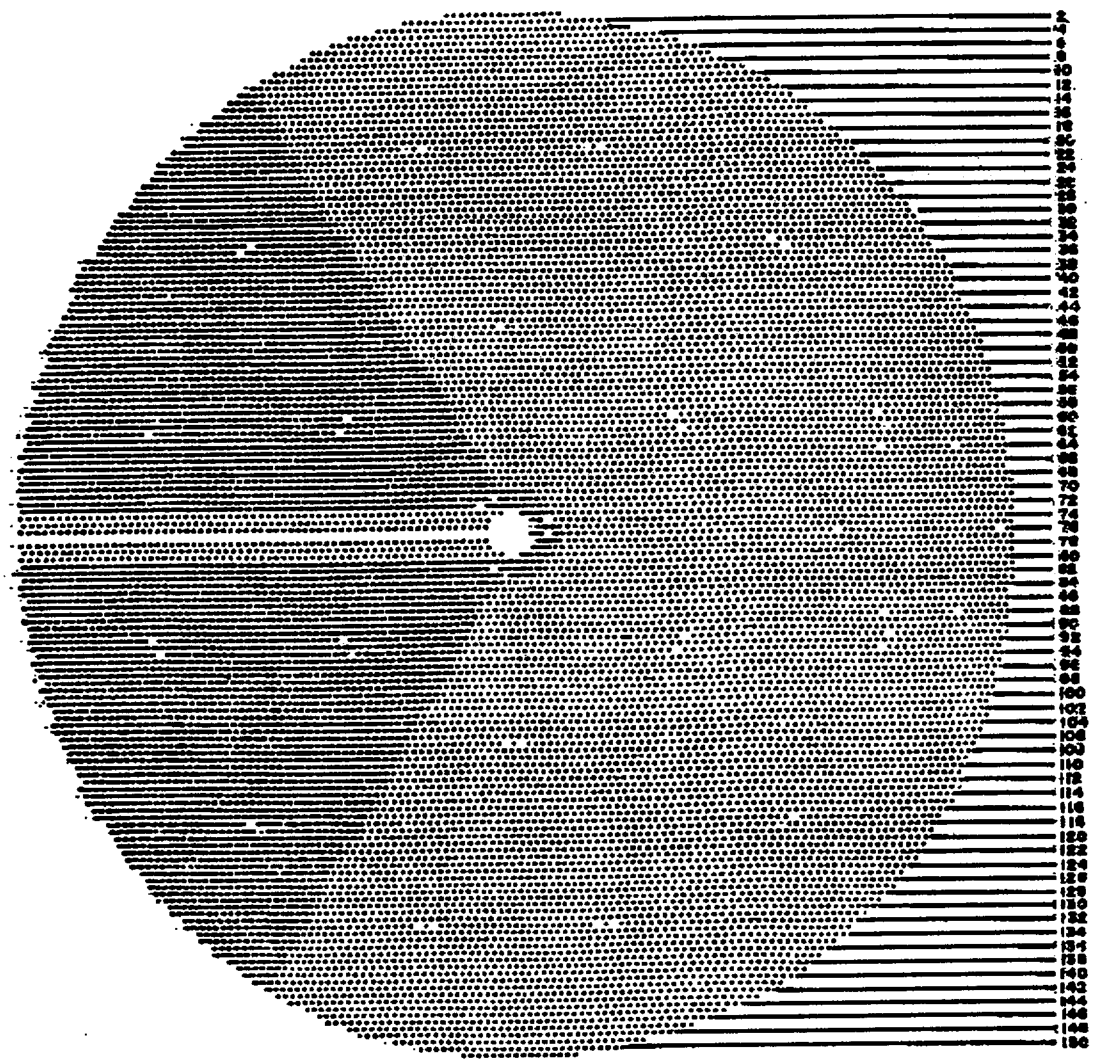
1 $S = \frac{3N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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

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

FIGURE 5.5.9-1

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



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

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5.5 Programs and Manuals (continued)

LC 2.C(7)

5.5.10 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation~~, and low pressure turbine disc stress corrosion cracking~~. The program shall include: (12)

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

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

c. Identification of process sampling points~~, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage~~; (25)

d. Procedures for the recording and management of data;

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

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

5.5.11 Ventilation Filter Testing Program (VFTP) Safeguards (ES) 4.10.2 4.11 4.17

A program shall be established to implement the following required testing of Engineered ~~Safety Feature (ESF) filter~~ ventilation systems filters at the frequencies specified in ~~[Regulatory Guide], and in accordance with~~ Regulatory Guide 1.52, Revision 2~~, ASME N510-1989, and AG-1~~.

<INSERT 5.0-12A>

~~a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < [0.05%] when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [±10%].~~ (13)

ESF Ventilation System	Flowrate
~~[ ]~~	~~[ ]~~

(continued)

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<INSERT 5.0-12A>

The VFTP is applicable to the Penetration Room Ventilation System (PRVS), the Fuel Handling Area Ventilation System (FHAVS), and the Control Room Emergency Ventilation System (CREVS).

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

   1. ≥ 99% DOP removal for the PRVS and the FHAVS when tested at the system design flowrate ± 10%; and — 3.13.1.a&c, 3.15.1.a&c

   2. ≥ 99.95% DOP removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate ± 10%. — 4.10.2.b.1, b.3, e

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

   1. ≥ 99% halogenated hydrocarbon removal for the PRVS and FHAVS when tested at the system design flowrate ± 10%; and — 3.13.1.a&c, 3.15.1.a&c

   2. ≥ 99.95% halogenated hydrocarbon removal for the CREVS when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%. — 4.10.2.b.1, b.3, f

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

   1. < 5% for the PRVS when tested at the system design flowrate ± 20%. — 3.13.1.b, LAR 99-06

   2. < 5% for the FHAVS when tested at the system design flowrate ± 20%. — 3.15.1.b

   3. when obtained as described in Regulatory Guide 1.52, Revision 2, for CREVS: — 4.10.2.b.2, c

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

      ii. ≤ 0.5% for 4 inch charcoal adsorber beds

d. Demonstrate, for the PRVS, FHAVS, and CREVS, that the pressure drop across the combined HEPA filters, other filters in the system, and the charcoal adsorbers is < 6 inches of water when tested at the system design flowrate ± 10%. — 3.13.1.c&d, 3.15.1.c&d, 4.10.2.b.3, d.1, 4.11.1, 4.17.1

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

(13)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.5]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

ESF Ventilation System	Flowrate
[ ]	[ ]

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of ≤ [30°C] and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

[Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).

Safety factor = [5] for systems with heaters.
= [7] for systems without heaters.]

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52,

(continued)

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5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued) (13)

Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

e. [Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].]

ESF Ventilation System	Wattage
[ ]	[ ]

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

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program NA

This program provides controls for potentially explosive gas mixtures contained in the [Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with ~~[Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"]~~ the ODCM. edit edit

temporary

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the [Waste Gas Holdup System] and a surveillance program to ensure the limits are maintained. Such limits shall be 3.24 4.28

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; 3.25.2 4.29.2

3.25.1

temporary

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. 34 equal to 4.29.1

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

5.5.13 Diesel Fuel Oil Testing Program 4.6.1.4.e

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

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

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

(continued)

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance Frequencies 16

5.5 Programs and Manuals

4.6.1.4.e

5.5.13 Diesel Fuel Oil Testing Program (continued)

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

3. water and sediment within limits ~~a clear and bright appearance with proper color~~; 14

b. ~~Other properties for ASTM 2D fuel oil are within limits~~ within ~~30~~ 31 days following ~~sampling and~~ addition, of the new fuel oil to storage tanks; ~~and~~ 38

(INSERT 5.0-16A)

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days ~~in accordance with~~ based on ASTM D-2276, Method A-2 or A-3; and 38

5.5.14 Technical Specifications (TS) Bases Control Program NA

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

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

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

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

2. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59. edit

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

X. Proposed changes that do not ~~meet~~ ~~the~~ this criteria ~~of 5.5.14b above~~ shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). edit

(continued)

<INSERT 5.0-16A>

..., verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil.

CTS

5.5 Programs and Manuals (continued)

N/A

5.5.15 Safety Function Determination Program (SFDP)

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

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

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

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

d. Other appropriate limitations and remedial or compensatory actions.

<INSERT 5.0 - 17A>

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

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

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

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

39

10

6.8.4
4.26.2
3.7.3A.1

<INSERT 5.0-17B>

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

<INSERT 5.0-17A>

... and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), ...

<INSERT 5.0-17B>

5.5.16 Reactor Building Leakage Rate Testing Program

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

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

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

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

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

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

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

CTS

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

~~The following reports shall be submitted in accordance with 10 CFR 50.4.~~ 17

6.12.2.2

5.6.1 Occupational Radiation Exposure Report

-----------------------------NOTE-----------------------------
A single submittal may be made for ~~a multiple unit station~~ AND. The submittal should combine sections common to ~~all~~ both units ~~at the station~~.
----

edit

INSERT 5.0-18A

~~A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following the initial criticality.]~~ 26

6.12.2.5

5.6.2 Annual Radiological Environmental Operating Report

-----------------------------NOTE-----------------------------
A single submittal may be made for ~~a multiple unit station~~ AND. The submittal should combine sections common to ~~all~~ both units ~~at the station~~.
----

edit

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual

(continued)

<INSERT 5.0-18A>

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

CTS

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued) 6.12.2.5

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. ~~[in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. 6.12.2.5

5.6.3 Radioactive Effluent Release Report 6.12.2.6

both AND

-----NOTE-----

A single submittal may be made for ~~a multiple unit station~~. The submittal ~~should~~ shall combine sections common to ~~all~~ units ~~at the station; however, for units with separate radwaste systems,~~ the submittal shall specify the releases of radioactive material from each unit. edit

26

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

CTS

5.6 Reporting Requirements (continued)

6.12.2.3

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience~~[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,]~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. 42

6.12.3

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

<INSERT 5.0-20A> ~~The individual specifications that address core operating limits must be referenced here.~~

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

<INSERT 5.0-20B> ~~Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.~~

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

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

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~ 18

~~a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic~~

(continued)

<INSERT 5.0-20A>

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

<INSERT 5.0-20B>

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

CTS

5.6 Reporting Requirements

18

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

testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: [The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: [Identify the NRC staff approval document by date.]

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

[Reviewer's Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.]

(continued)

CTS

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued) 18

[6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.]

5.6.7 EDG Failures Report 19

[If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.]

5.6.8 PAM Report 20

When a report is required by Condition B or G of LCO 3.3.[17], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

CTS

5.6 Reporting Requirements (continued)

5.6.9 ~~[Tendon Surveillance Report~~ 6.12.4

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~ (37)

INSERT 5.0-23A

5.6.7 ~~5.6.10~~ [Steam Generator Tube ~~Inspector~~ Surveillance Report] (S) 4.18.6 6.12.5.d

~~Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~ (36)

INSERT 5.0-23B

~~[Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.]~~

<INSERT 5.0-23A>

5.6.6 Reactor Building Inspection Report

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

<INSERT 5.0-23B>

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

   1. Number and extent of tubes inspected;

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

   3. Identification of tubes plugged and tubes sleeved;

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

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

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

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

CTS
6.11.1

INSERT 5.0-24A

42

5.0 ADMINISTRATIVE CONTROLS

[5.7 High Radiation Area]

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel

(continued)

<INSERT 5.0-24A>

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

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

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures. 42

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

42

[5.7 High Radiation Area]

5.7.2 (continued)

under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.