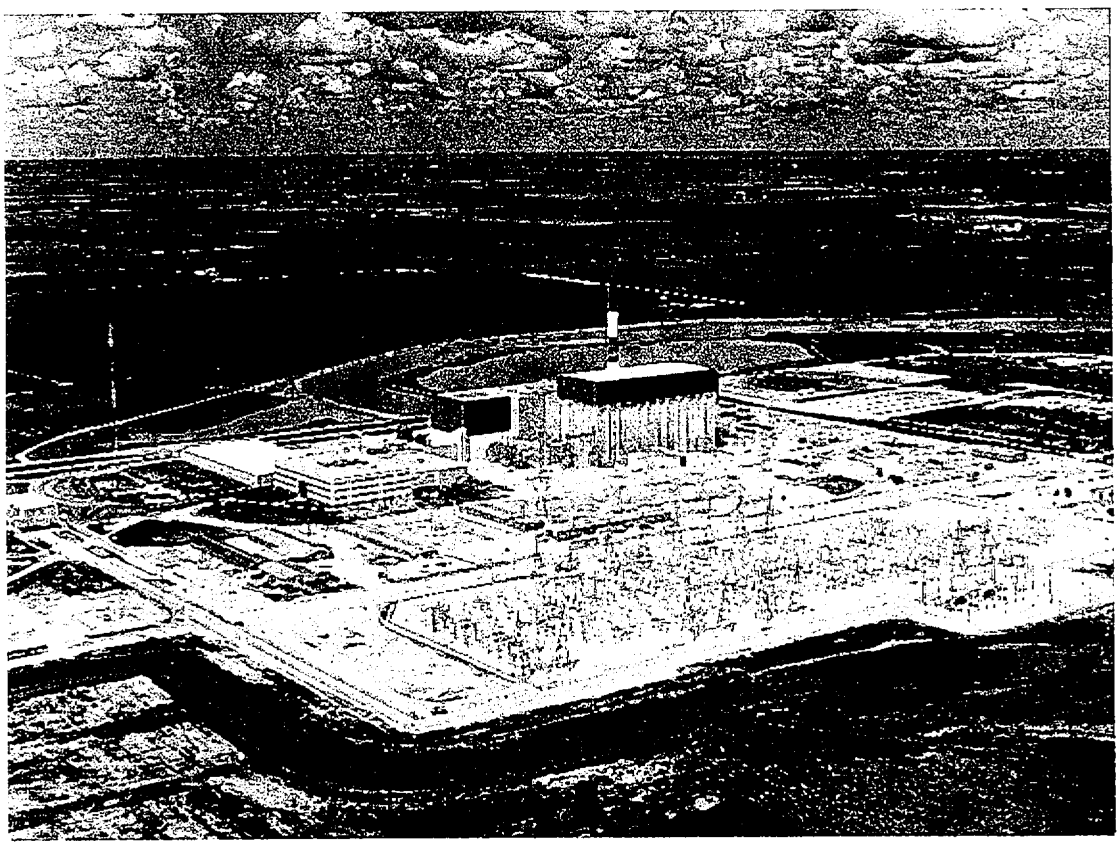
Improved Technical Specifications



LaSalle County Station

Volume 8:
Section 3.7

ComEd

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHRSW subsystem inoperable.	A.1	--------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for RHR shutdown cooling subsystem made inoperable by RHRSW System. --------------------- Restore RHRSW subsystem to OPERABLE status.	7 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Both RHRSW subsystems inoperable.	B.1	--------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.9 for RHR shutdown cooling subsystems made inoperable by RHRSW System. --------------------- Restore one RHRSW subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days

3.7 PLANT SYSTEMS

3.7.2 Diesel Generator Cooling Water (DGCW) System

LCO 3.7.2 The following DGCW subsystems shall be OPERABLE:

a. Three unit DGCW subsystems; and

b. The opposite unit Division 2 DGCW subsystem.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each DGCW subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGCW subsystems inoperable.	A.1 Declare supported component(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.2.1 Verify each required DGCW subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.2	Verify each required DGCW pump starts automatically on each required actual or simulated initiation signal.	24 months

3.7 PLANT SYSTEMS

3.7.3 Ultimate Heat Sink (UHS)

LCO 3.7.3 The Core Standby Cooling System (CSCS) pond shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CSCS pond inoperable due to sediment deposition or bottom elevation not within limit.	A.1 Restore CSCS pond to OPERABLE status.	90 days
B. Required Action and associated Completion Time of Condition A not met. OR CSCS pond inoperable for reasons other than Condition A.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the average water temperature of CSCS pond is ≤ 97°F.	24 hours
SR 3.7.3.2	Verify sediment level is ≤ 1.5 ft in the intake flume and the CSCS pond.	24 months
SR 3.7.3.3	Verify CSCS pond bottom elevation is ≤ 686.5 ft.	24 months

3.7 PLANT SYSTEMS

3.7.4 Control Room Area Filtration (CRAF) System

LCO 3.7.4 Two CRAF subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRAF subsystem inoperable.	A.1 Restore CRAF subsystem to OPERABLE status.	7 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE CRAF subsystem in pressurization mode.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two CRAF subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CRAF subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------- LCO 3.0.3 is not applicable. ----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Operate each CRAF subsystem for ≥ 10 continuous hours with the heaters operating.	31 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.2	Manually initiate flow through the CRAF recirculation filters for $\geq$ 10 hours.	31 days
SR 3.7.4.3	Perform required CRAF filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.4.4	Verify each CRAF subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.7.4.5	Verify each CRAF subsystem can maintain a positive pressure of $\geq$ 0.125 inches water gauge relative to adjacent areas during the pressurization mode of operation at a flow rate of $\leq$ 4000 cfm.	24 months

3.7 PLANT SYSTEMS

3.7.5 Control Room Area Ventilation Air Conditioning (AC) System

LCO 3.7.5 Two control room area ventilation AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room area ventilation AC subsystem inoperable.	A.1 Restore control room area ventilation AC subsystem to OPERABLE status.	30 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE control room area ventilation AC subsystem in operation.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two control room area ventilation AC subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two control room area ventilation AC subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u> E.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Monitor control room and auxiliary electric equipment room temperatures.	12 hours
SR 3.7.5.2	Verify correct breaker alignment and indicated power are available to the control room area ventilation AC subsystems.	7 days

3.7 PLANT SYSTEMS

3.7.6 Main Condenser Offgas

LCO 3.7.6 The gross gamma activity rate of the noble gases measured prior to the holdup line shall be ≤ 340,000 μCi/second after decay of 30 minutes.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated and steam jet air ejector (SJAE) in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross gamma activity rate of the noble gases not within limit.	A.1 Restore gross gamma activity rate of the noble gases to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
	B.2 Isolate SJAE.	12 hours
	<u>OR</u>	
	B.3.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.3.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	-------------------NOTE------------------- Not required to be performed until 31 days after any main steam line not isolated and SJAE in operation. -- Verify the gross gamma activity rate of the noble gases is $\leq$ 340,000 μCi/second after decay of 30 minutes.	31 days AND Once within 4 hours after a $\geq$ 50% increase in the nominal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level

3.7 PLANT SYSTEMS

3.7.7 Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

OR

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.7.1 Verify one complete cycle of each main turbine bypass valve.	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.2	Perform a system functional test.	24 months
SR 3.7.7.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

3.7 PLANT SYSTEMS

3.7.8 Spent Fuel Storage Pool Water Level

LCO 3.7.8 The spent fuel storage pool water level shall be $\geq$ 21 ft 4 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool,
During movement of new fuel assemblies in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.8.1 Verify the spent fuel storage pool water level is $\geq$ 21 ft 4 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days

B 3.7 PLANT SYSTEMS

B 3.7.1 Residual Heat Removal Service Water (RHRSW) System

BASES

BACKGROUND

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System. The RHRSW System also provides cooling water to the RHR pump seal coolers which are required for RHR pump operation during the shutdown cooling mode in MODE 3.

The RHRSW System consists of two independent and redundant subsystems. Each subsystem is made up of two pumps (together capable of providing a nominal flow of 7400 gpm), a suction source, valves, piping, heat exchanger, and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity with both pumps operating to maintain safe shutdown conditions. The two subsystems are separated from each other so that failure of one subsystem will not affect the OPERABILITY of the other subsystem. The RHRSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RHRSW System is described in the UFSAR, Section 9.2.1, Reference 1.

The RHRSW and the Diesel Generator Cooling Water subsystems are subsystems to the Core Standby Cooling System (CSCS) – Equipment Cooling Water System (ECWS). The CSCS – ECWS consists of three independent piping subsystems corresponding to essential electrical power supply Divisions 1, 2, and 3. The CSCS – ECWS subsystems take suction from the service water tunnel located in the Lake Screen House. The RHRSW subsystems are manually initiated. Cooling water is then pumped from the service water tunnel by the RHRSW pumps to the supported system and components (RHR heat exchangers and RHR pump seal coolers). After removing heat from its supported systems and components, the water from the RHRSW subsystem is discharged to the CSCS Pond (i.e., the Ultimate Heat Sink) through a discharge line that is

(continued)

BASES

BACKGROUND (continued)

common to the corresponding divisional discharge from the other unit. The discharge line terminates in the discharge structure at an elevation above the normal CSCS Pond level.

The system is initiated manually from the control room. In addition, the Division 2 RHRSW subsystem may be initiated manually from the remote shutdown panel in the auxiliary electric equipment room. If operating during a loss of offsite power, the system is automatically load shed to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time after the LOCA.

APPLICABLE SAFETY ANALYSES

The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RHRSW System to support long term cooling of the reactor or primary containment is discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The safety analyses for long term cooling were performed for various combinations of RHR System failures. The worst case single failure that would affect the performance of the RHRSW System is any failure that would disable one subsystem of the RHRSW System. As discussed in the UFSAR, Section 6.2.2.3.1 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystem and the associated RHR System is assumed to occur 10 minutes after a DBA. The RHRSW flow assumed in the analyses is 7400 gpm with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are 200°F and 30.6 psig, respectively, well below the design temperature of 275°F and maximum design pressure of 45 psig.

The RHRSW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

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

An RHRSW subsystem is considered OPERABLE when:

a. Two pumps are OPERABLE; and

b. An OPERABLE flow path is capable of taking suction from the CSCS service water tunnel and transferring the water to the associated RHR heat exchanger at the assumed flow rate.

An adequate suction source is not addressed in this LCO since the minimum net positive suction head and the maximum suction source temperature are covered by the requirements specified in LCO 3.7.3, "Ultimate Heat Sink (UHS)."

APPLICABILITY

In MODES 1, 2, and 3, the RHRSW System is required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling (LCO 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray") and decay heat removal (LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown"). The Applicability is therefore consistent with the requirements of these systems.

In MODES 4 and 5, the OPERABILITY requirements of the RHRSW System are determined by the systems it supports and therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, the LCOs of the RHR Shutdown Cooling System (LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System — Cold Shutdown," LCO 3.9.8, "Residual Heat Removal (RHR) — High Water Level," and LCO 3.9.9, "Residual Heat Removal (RHR) — Low Water Level"), which require portions of the RHRSW System to be OPERABLE, will govern RHRSW System operation in MODES 4 and 5.

(continued)

BASES (continued)

ACTIONS

A.1

Required Action A.1 is intended to handle the inoperability of one RHRSW subsystem. The Completion Time of 7 days is allowed to restore the RHRSW subsystem to OPERABLE status. With the unit in this condition, the remaining OPERABLE RHRSW subsystem is adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE RHRSW subsystem could result in loss of RHRSW function. The Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE subsystem and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.9, be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

B.1

With both RHRSW subsystems inoperable (e.g., both subsystems with inoperable pump(s) or flow paths, or one subsystem with an inoperable pump and one subsystem with an inoperable flow path), the RHRSW System is not capable of performing its intended function. At least one subsystem must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time for restoring one RHRSW subsystem to OPERABLE status, is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.9, be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual, power operated, and automatic valve in each RHRSW subsystem flow path provides assurance that the proper flow paths will exist for RHRSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be realigned to its accident position. This is acceptable because the RHRSW System is a manually initiated system.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

REFERENCES

1. UFSAR, Section 9.2.1.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. UFSAR, Section 6.2.2.3.1.

B 3.7 PLANT SYSTEMS

B 3.7.2 Diesel Generator Cooling Water (DGCW) System

BASES

BACKGROUND

The DGCW System is designed to provide cooling water for the removal of heat from the standby diesel generators, low pressure core spray (LPCS) pump motor cooling coils, and Emergency Core Cooling System (ECCS) cubicle area cooling coils that support equipment required for a safe reactor shutdown following a design basis accident (DBA) or transient.

The DGCW System consists of three independent cooling water headers (Divisions 1, 2, and 3), and their associated pumps, valves, and instrumentation. The pump and header for the Division 1 DGCW subsystem is common to both units (and supplies cooling to equipment on both units). The other divisions have independent pumps and suction headers.

The following combinations of DGCW pumps are sized to provide sufficient cooling capacity to support the required safety related systems during safe shutdown of the unit following a loss of coolant accident (LOCA):

a. The unit Division 1 and 2 DGCW pumps;

b. The unit Division 1 and 3 DGCW pumps and opposite unit Division 2 DGCW pump; or

c. The unit Division 2 and 3 DGCW pumps.

The unit Division 1 DGCW subsystem services its associated Diesel Generator (DG) and ECCS cubicle area coolers, and the LPCS pump motor cooler. The unit Division 2 DGCW subsystem services its associated DG and ECCS cubicle area cooler. The unit Division 3 DGCW subsystem services the High Pressure Core Spray (HPCS) DG and its associated ECCS cubicle area cooler. The opposite unit Division 2 DGCW subsystem services its associated DG for support of systems required by both units.

(continued)

BASES

BACKGROUND (continued)	The DGCW and the Residual Heat Removal Service Water (RHRSW) subsystems are subsystems to the Core Standby Cooling System (CSCS) – Equipment Cooling Water System (ECWS). The CSCS – ECWS consists of three independent piping subsystems corresponding to essential electrical power supply Divisions 1, 2, and 3. The CSCS – ECWS subsystems take a suction from the service water tunnel located in the Lake Screen House. Each DGCW pump auto-starts upon receipt of a diesel generator (DG) start signal when power is available to the pump's electrical bus or on start of ECCS cubicle area coolers. The Division 1 DGCW pump also auto-starts upon receipt of a start signal for the LPCS pump. Cooling water is then pumped from the service water tunnel by the DGCW pumps to the supported systems and components (i.e., the DGs, LPCS pump motor cooler, and the ECCS cubicle area coolers). After removing heat from these systems and components, the water from the DGCW subsystem is discharged to the CSCS pond (i.e., the Ultimate Heat Sink) through a discharge line that is common to the corresponding divisional discharge from the other unit. The discharge line terminates in the discharge structure at an elevation above the normal CSCS Pond level. A complete description of the DGCW System is presented in the UFSAR, Section 9.2.1 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The ability of the DGCW System to provide adequate cooling to the DGs, LPCS pump motor cooling coils and ECCS cubicle area cooling coils is an implicit assumption for the safety analyses presented in UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). The ability to provide onsite emergency AC power is dependent on the ability of the DGCW System to cool the DGs. The DGCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The unit's Division 1, 2, and 3, and the opposite unit's Division 2 DGCW subsystems are required to be OPERABLE to ensure the effective operation of the DGs, the LPCS pump motor, and the ECCS equipment supported by the ECCS cubicle area coolers during a DBA or transient. The OPERABILITY of

(continued)

BASES

LCO (continued)

each DGCW subsystem is based on having an OPERABLE pump and an OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring cooling water to the associated diesel generator, LPCS pump motor cooling coils, and ECCS cubicle area cooling coils, as required.

An adequate suction source is not addressed in this LCO since the minimum net positive suction head of the DGCW pump and the maximum suction source temperature are covered by the requirements specified in LCO 3.7.3, "Ultimate Heat Sink (UHS)."

APPLICABILITY

In MODES 1, 2, and 3, the DGCW subsystems are required to support the OPERABILITY of equipment serviced by the DGCW subsystems and required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the DGCW subsystems are determined by the systems they support. Therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, the LCOs of the systems supported by the DGCW subsystems will govern DGCW System OPERABILITY requirements in MODES 4 and 5.

ACTIONS

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

A.1

If one or more DGCW subsystems are inoperable, the associated DG(s) and ECCS components supported by the affected DGCW loop, including LPCS pump motor cooling coils or ECCS cubicle area cooling coils, as applicable, cannot perform their intended function and must be immediately declared inoperable. In accordance with LCO 3.0.6, this

(continued)

BASES

ACTIONS

A.1 (continued)

also requires entering into the Applicable Conditions and Required Actions for LCO 3.8.1, "AC Sources—Operating," and LCO 3.5.1, "Emergency Core Cooling Systems (ECCS)—Operating."

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

Verifying the correct alignment for manual, power operated, and automatic valves in each required DGCW subsystem flow path provides assurance that the proper flow paths will exist for DGCW subsystem operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet be considered in the correct position provided it can be automatically realigned to its accident position, within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

SR 3.7.2.2

This SR ensures that each required DGCW subsystem pump will automatically start to provide required cooling to the associated DG, LPCS pump motor cooling coils, and ECCS cubicle area cooling coils, as applicable, when the associated DG starts and the respective bus is energized or on start of the applicable ECCS cubicle area cooler. For the Division 1 DGCW subsystem, this SR also ensures the DGCW pump automatically starts on receipt of a start signal for the unit LPCS pump. These starts may be performed using actual or simulated initiation signals.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.2.2 (continued) Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based at the refueling cycle. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.
REFERENCES	1. UFSAR, Section 9.2.1. 2. UFSAR, Chapter 6. 3. UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.3 Ultimate Heat Sink (UHS)

BASES

BACKGROUND	The UHS (i.e., the Core Standby Cooling System (CSCS) Pond) consists of the volume of water remaining in the cooling lake following the failure of the main dike. This water has a depth of approximately 5 feet and a top water elevation established at 690 feet. The volume of the remaining water in the cooling lake is sufficient to permit a safe shutdown and cooldown of the station for 30 days with no water makeup for both accident and normal conditions (Regulatory Guide 1.27, Ref. 1). The CSCS Pond provides a source of water to the service water tunnel from which the Residual Heat Removal Service Water (RHRSW) and Diesel Generator Cooling Water (DGCW) pumps take suction. The service water tunnel is filled from the CSCS Pond by six inlet lines which connect to the circulating water pump forebays. Prior to entering the service water tunnel inlet pipes, the water is strained by the Lake Screen House traveling screens to prevent large pieces of debris from entering the system and blocking flow or damaging equipment. However, because the traveling screens are not safety related, a 54-inch bypass line around the screens, isolated by a normally closed manual valve, is provided to assure a continuous supply of CSCS Pond water to the service water tunnel. Additional information on the design and operation of the CSCS Pond is provided in UFSAR, Sections 9.2.1 and 9.2.6 (Refs. 2 and 3). The excavation slopes of the CSCS Pond and flume are designed to be stable under all conditions of emergency operation while providing the capability to supply adequate cooling water to equipment required for safe reactor shutdown.
APPLICABLE SAFETY ANALYSES	The volume of the CSCS pond is sized to permit the safe shutdown and cooldown of the units for a 30 day period with no additional makeup water source available for both normal and accident conditions (Ref. 2). The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

OPERABILITY of the UHS is based on a maximum water temperature of 97°F and a minimum pond water level at or above elevation 690 ft mean sea level. In addition, to ensure the volume of water available in the CSCS pond is sufficient to maintain adequate long term cooling, sediment deposition (in the intake flume and in the pond) must be ≤ 1.5 ft and CSCS pond bottom elevation must be ≤ 686.5 ft.

APPLICABILITY

In MODES 1, 2, and 3, the UHS is required to be OPERABLE to support OPERABILITY of the equipment serviced by the UHS, and is required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the UHS is determined by the systems it supports. Therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. The LCOs of the systems supported by the UHS will govern UHS OPERABILITY requirements in MODES 4 and 5.

ACTIONS

A.1

If the CSCS pond is inoperable, due to sediment deposition > 1.5 ft (in the intake flume, CSCS pond, or both) or the pond bottom elevation > 686.5 ft, action must be taken to restore the inoperable UHS to an OPERABLE status within 90 days. The 90 day Completion Time is reasonable based on the low probability of an accident occurring during that time, historical data corroborating the low probability of continued degradation (i.e., further excessive sediment deposition or pond bottom elevation changes) of the CSCS pond during that time, and the time required to complete the Required Action.

B.1 and B.2

If the CSCS pond cannot be restored to OPERABLE status within the associated Completion Time, or the CSCS pond is determined inoperable for reasons other than Condition A (e.g., inoperable due to CSCS pond average water temperature

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

> 97°F), the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Verification of the CSCS pond temperature ensures that the heat removal capabilities of the RHRSW System and DGCW System are within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.3.2

This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the sediment level in the intake flume and the CSCS pond is ≤ 1.5 feet. Sediment level is determined by a series of sounding cross-sections compared to as-built soundings. The 24 month Frequency is based on historical data and engineering judgement regarding sediment deposition rate.

SR 3.7.3.3

This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the CSCS pond bottom elevation is ≤ 686.5 feet. The 24 month Frequency is based on historical data and engineering judgement regarding pond bottom elevation changes.

REFERENCES

1. Regulatory Guide 1.27, Revision 2, January 1976.
2. UFSAR, Section 9.2.1.
3. UFSAR, Section 9.2.6.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Area Filtration (CRAF) System

BASES

BACKGROUND

The CRAF System provides a radiologically controlled environment (control room and auxiliary electric equipment room) from which the unit can be safely operated following a Design Basis Accident (DBA). The Control Room Area Heating Ventilation and Air Conditioning (HVAC) System is comprised of the Control Room HVAC System and the Auxiliary Electric Equipment Room (AEER) HVAC System. The Control Room HVAC System is common to both units and serves the control room, main security control center, and the control room habitability storage room (toilet room). The AEER HVAC System is common to both units and services the auxiliary electrical equipment rooms. The control room area is comprised of the areas covered by the Control Room and AEER HVAC Systems.

The safety related function of the CRAF System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems (i.e., the emergency makeup air filter units (EMUs) for treatment of outside supply air). Recirculation filters are also provided for treatment of recirculated air. Each EMU subsystem consists of a demister, an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork, dampers, and instrumentation and controls. Demisters remove water droplets from the airstream. The electric heater reduces the relative humidity of the air entering the EMUs. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay. Each Control Room and AEER Ventilation System has a charcoal recirculation filter in the supply of the system that is normally bypassed. In addition, the OPERABILITY of the CRAF System is dependent upon portions of the Control Room Area HVAC System, including the control room and auxiliary electric equipment room outside air intakes, supply fans, ducts, dampers, etc.

(continued)

BASES

BACKGROUND (continued)

In addition to the safety related standby emergency filtration function, parts of the CRAF System that are shared with the Control Room Area HVAC System are operated to maintain the control room area environment during normal operation. Upon receipt of a high radiation signal from the outside air intake (indicative of conditions that could result in radiation exposure to control room personnel), the CRAF System automatically isolates the normal outside air supply to the Control Room Area HVAC System, and diverts the minimum outside air requirement through the EMUs before delivering it to the control room area. The recirculation filters for the control room and AEER must be manually placed in service within 4 hours of receipt of any control room high radiation alarm.

The CRAF System is designed to maintain the control room area environment for a 30 day continuous occupancy after a DBA, without exceeding a 5 rem whole body dose or its equivalent to any part of the body. CRAF System operation in maintaining the control room area habitability is discussed in the UFSAR, Sections 6.4, 6.5.1, and 9.4.1 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CRAF System to maintain the habitability of the control room area is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 6 and 15 (Refs. 4 and 5, respectively). The pressurization mode of the CRAF System is assumed to operate following a loss of coolant accident, main steam line break, fuel handling accident, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 5. No single active failure will cause the loss of outside or recirculated air from the control room area.

The CRAF System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two redundant subsystems of the CRAF System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

(continued)

BASES

LCO (continued)

The CRAF System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated EMU is OPERABLE and the associated charcoal recirculation filters for the control room and AEER are OPERABLE. An EMU is considered OPERABLE when its associated:

a. Fan is OPERABLE;

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

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation through the EMU can be maintained.

Additionally, the portions of the Control Room Area HVAC System that supply the outside air to the EMUs are required to be OPERABLE. This includes the outside air intakes, associated dampers and ductwork.

In addition, the control room area boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that the pressurization limit of SR 3.7.4.5 can be met. However, it is acceptable for access doors to be open for normal control room area entry and exit and not consider it to be a failure to meet the LCO.

APPLICABILITY

In MODES 1, 2, and 3, the CRAF System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRAF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of irradiated fuel assemblies in the secondary containment;

(continued)

BASES

APPLICABILITY (continued)

b. During CORE ALTERATIONS; and

c. During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

A.1

With one CRAF subsystem inoperable, the inoperable CRAF subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CRAF subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of CRAF System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CRAF subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

C.1, C.2.1, C.2.2, and C.2.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CRAF subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRAF subsystem may be placed in the pressurization mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room area. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

D.1

If both CRAF subsystems are inoperable in MODE 1, 2, or 3, the CRAF System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

E.1, E.2, and E.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CRAF subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation for $\geq$ 10

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1 (continued)

continuous hours during system operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.4.2

This SR verifies that flow can be manually realigned through the CRAF System recirculation filters and maintained for ≥ 10 hours. Standby systems should be checked periodically to ensure that they function. Monthly operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and two subsystem redundancy available.

SR 3.7.4.3

This SR verifies that the required CRAF testing is performed in accordance with Specification 5.5.8, "Ventilation Filter Testing Program (VFTP)." The CRAF filter tests are in accordance with ANSI/ASME N510-1989 (Ref. 5). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.4.4

This SR verifies that each CRAF subsystem automatically switches to the pressurization mode of operation on an actual or simulated air intake radiation monitors initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 overlaps this SR to provide complete testing of the safety function. Operating experience has shown that these components normally pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.5

This SR verifies the integrity of the control room area and the assumed inleakage rates of potentially contaminated air. The control room area positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CRAF System. During the pressurization mode of operation, the CRAF System is designed to slightly pressurize the control room area to ≥ 0.125 inches water gauge positive pressure with respect to adjacent areas to prevent unfiltered inleakage. The CRAF System is designed to maintain this positive pressure at a flow rate of ≤ 4000 cfm to the control room area in the pressurization mode. This test also requires manual initiation of flow through the control room and AEER recirculation filters line when the CRAF System is in the pressurization mode of operation. The Frequency of 24 months is consistent with industry practice and other filtration system SRs.

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Section 6.5.1.
3. UFSAR, Section 9.4.1.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. ANSI/ASME N510-1989.

B 3.7 PLANT SYSTEMS

B 3.7.5 Control Room Area Ventilation Air Conditioning (AC) System

BASES

BACKGROUND

The Control Room Area Ventilation AC System provides temperature control for the control room area. The control room area is comprised of the control room and the Auxiliary Electric Equipment Rooms (AEERs).

The Control Room Area Ventilation AC System is comprised of two independent, redundant subsystems that provide cooling and heating of control room air and the auxiliary electric equipment rooms air. Each Control Room Area Ventilation AC subsystem consists of a Control Room AC subsystem and an AEER AC subsystem. The associated Control Room AC and AEER AC subsystems share a common outside air intake with a common emergency makeup air filter unit. The Control Room AC System is common to both units and serves the control room, main security control center, and the control room habitability storage room (toilet room). The AEER AC System is common to both units and services the AEERs.

Each Control Room Area Ventilation AC subsystem is powered from a Division 2 power source. One subsystem is powered from Unit 1 Division 2 and the other subsystem is powered from Unit 2 Division 2.

Each control room AC and AEER AC subsystem consists of a supply air filter, supply and return air fans, direct expansion cooling coils, an air-cooled condenser, a refrigerant compressor and receiver, heating coils, ductwork, dampers, and instrumentation and controls to provide temperature control for their respective areas. However, the heating coils are not safety related.

The Control Room Area Ventilation AC System is designed to provide a controlled environment under both normal and accident conditions. A single control room area ventilation AC subsystem provides the required temperature control to maintain a suitable control room and AEER environment for a sustained occupancy of at least the required normal and emergency shift crew complements. The design conditions for

(continued)

BASES

BACKGROUND (continued)	habitability of the control room and AEER environment are 65°F to 85°F and a maximum of 50% relative humidity. The Control Room Area Ventilation AC System operation in maintaining the temperatures of the control room and AEERs is discussed in the UFSAR, Sections 6.4 and 9.4.1 (Refs. 1 and 2, respectively).
APPLICABLE SAFETY ANALYSES	The design basis of the Control Room Area Ventilation AC System is to maintain temperatures of the control room and AEERs for a 30 day period after a Design Basis Accident (DBA). The Control Room Area Ventilation AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room Area Ventilation AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room and AEERs. A single active failure of a component of the Control Room Area Ventilation AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room and AEERs temperature control. The Control Room Area Ventilation AC System is designed in accordance with Seismic Category I requirements, with exceptions described in UFSAR Section 9.4.1.1.1.1 (Ref. 3). The Control Room Area Ventilation AC System is capable of removing sensible and latent heat loads from the control room and AEERs, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY. The Control Room Area Ventilation AC System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant subsystems of the Control Room Area Ventilation AC System are required to be OPERABLE to ensure that at least one subsystem is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

(continued)

BASES

LCO (continued)	The Control Room Area Ventilation AC System is considered OPERABLE when the individual components necessary to maintain the control room and AEERs temperatures are OPERABLE in both subsystems. These components include the supply and return air fans, direct expansion cooling coils, an air-cooled condenser, a refrigerant compressor and receiver, ductwork, dampers, and instrumentation and controls.
APPLICABILITY	In MODE 1, 2, or 3, the Control Room Area Ventilation AC System must be OPERABLE to ensure that the control room and AEERs temperatures will not exceed equipment OPERABILITY limits during operation of the Control Room Area Filtration (CRAF) System in the pressurization mode. In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room Area Ventilation AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated: a. During movement of irradiated fuel assemblies in the secondary containment; b. During CORE ALTERATIONS; and c. During operations with a potential for draining the reactor vessel (OPDRVs).
ACTIONS	<u>A.1</u> With one control room area ventilation AC subsystem inoperable, the inoperable control room area ventilation AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room area ventilation AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room area ventilation air conditioning function. The 30 day Completion Time is based

(continued)

BASES

ACTIONS

A.1 (continued)

on the low probability of an event occurring requiring operation of the CRAF System in the pressurization mode and the consideration that the remaining subsystem can provide the required protection.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable control room area ventilation AC subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

C.1, C.2.1, C.2.2, and C.2.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE control room AC subsystem may be placed immediately in operation.

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

D.1

If both control room area ventilation AC subsystems are inoperable in MODE 1, 2, or 3, the Control Room Area Ventilation AC System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

The Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

OPDRVs with two control room area ventilation AC subsystems inoperable, action must be taken to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR monitors the control room and AEER temperatures for indication of Control Room Area Ventilation AC System performance. Trending of control room area temperature will provide a qualitative assessment of refrigeration unit OPERABILITY. Limiting the average temperature of the Control Room and AEER to less than or equal to 85°F provides a threshold beyond which the operating control room area ventilation AC subsystem is no longer demonstrating capability to perform its function. This threshold provides margin to temperature limits at which equipment qualification requirements could be challenged. Subsystem operation is routinely alternated to support planned maintenance and to ensure each subsystem provides reliable service. The 12 hour Frequency is adequate considering the continuous manning of the control room by the operating staff.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	<u>SR 3.7.5.2</u> Verifying proper breaker alignment and power available to the control room area ventilation AC subsystems provides assurance of the availability of the system function. The 7 day Frequency is appropriate in view of other administrative controls that assure system availability.
REFERENCES	1. UFSAR, Section 6.4. 2. UFSAR, Section 9.4.1. 3. UFSAR, Section 9.4.1.1.1.1.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND	During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases. The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and water separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the water separator prior to entering the holdup line.
APPLICABLE SAFETY ANALYSES	The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event as discussed in the UFSAR, Section 15.7.1.1 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2). The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/Mwt-second after decay of 30 minutes. The LCO is conservatively established based on the safety analysis discussed in Reference 1.

(continued)

BASES (continued)

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, main steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment considering the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of a Main Condenser Offgas System rupture occurring.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits within the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from significant sources of radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in each drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR, on a 31 day Frequency, requires an isotopic analysis of a representative offgas sample taken prior to the holdup line to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-135M, Xe-138, Kr-85M, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted (as indicated by the offgas pre-treatment noble gas activity monitor), to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. UFSAR, Section 15.7.1.

2. 10 CFR 100.

B 3.7 PLANT SYSTEMS

B 3.7.7 Main Turbine Bypass System

BASES

BACKGROUND	The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is approximately 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of five valves mounted on a valve manifold connected to the main steam lines between the main steam isolation valves and the main turbine stop valves. Each of these valves is sequentially operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the UFSAR, Section 7.7.5.2 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves, directing all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass valve outlet manifold, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser (Ref. 2).
APPLICABLE SAFETY ANALYSES	The Main Turbine Bypass System is assumed to function during the turbine trip, turbine generator load rejection, and feedwater controller failure maximum demand transients, described in the UFSAR, Sections 15.2.3, 15.2.2A, and 15.1.2A (Refs. 3, 4, and 5, respectively). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty. The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii)

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, such that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow continued operation.

An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Refs. 3, 4, and 5). The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR.

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during the turbine trip, feedwater controller failure maximum demand, and turbine generator load rejection transients. As discussed in the Bases for LCO 3.2.2 sufficient margin to these limits exists < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), and the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status and the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine trip, turbine generator load rejection, and feedwater controller failure maximum demand transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 7 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required simulated system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME, as defined in the transient analysis inputs for the cycle, is in compliance with the assumptions of the appropriate safety analyses. The response time limits are specified in the Technical Requirements Manual (Ref. 6). The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 7.7.5.2.
2. UFSAR, Section 10.4.4.
3. UFSAR, Section 15.2.3.
4. UFSAR, Section 15.2.2A.
5. UFSAR, Section 15.1.2A.
6. Technical Requirements Manual.

B 3.7 PLANT SYSTEMS

B 3.7.8 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Sections 9.1.2 and 15.7.4 (Refs. 1 and 2, respectively).
APPLICABLE SAFETY ANALYSES	The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG-0800, Section 15.7.4, Ref. 3) of the 10 CFR 100 (Ref. 4) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 5). The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are less severe than those of the fuel handling accident over the reactor core (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident. The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The specified water level preserves the assumption of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

APPLICABILITY

This LCO applies whenever movement of irradiated fuel assemblies occurs in the spent fuel storage pool or whenever movement of new fuel assemblies occurs in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since fuel assembly movement can occur in MODES 1, 2, or 3, Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of fuel assembly movement are not postponed due to entry into LCO 3.0.3.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. With the spent fuel storage pool level less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.7.8.1 This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable and water level changes are controlled by unit procedures.
REFERENCES	1. UFSAR, Section 9.1.2. 2. UFSAR, Section 15.7.4. 3. NUREG-0800, Section 15.7.4, Revision 1, July 1981. 5. 10 CFR 100. 6. Regulatory Guide 1.25, March 1972.

A.1

3/4.7 PLANT SYSTEMS

3/4.7.1 CORE STANDBY COOLING SYSTEM-EQUIPMENT COOLING WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.1 3.7.1.1 Two independent residual heat removal service water (RHRSW) system subsystems shall be OPERABLE, ~~with each subsystem comprised of:~~ LA.1

~~a. Two OPERABLE RHRSW pumps, and~~

~~b. An OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring the water through the associated RHR heat exchanger.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5. LA.2

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:

ACTION A — 1. With one RHRSW subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. L.1 ACTION C

ACTION B — 2. With both RHRSW subsystems inoperable, restore one inoperable subsystem in 8 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours. L.2 A.2 ACTION C

Notes to Required Actions A.1 and B.1 — b. In OPERATIONAL CONDITION 3 or 4 with the RHRSW subsystem(s)# inoperable which is associated with an RHR shutdown cooling mode loop(s) required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, as applicable, declare the associated RHR shutdown cooling mode loop(s) inoperable and take the ACTION required to Specification 3.4.9.1 or 3.4.9.2, as applicable. LA.2

~~c. In OPERATIONAL CONDITION 5 with the RHRSW subsystem# cooling mode loop(s) inoperable which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2.~~ LA.2

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 4.7.1.1 Each residual heat removal service water system subsystem shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position, or can be aligned to the correct position. A.3

~~*Whenever both RHRSW subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~ A.2

~~#Only one pump per subsystem need be OPERABLE if sufficient for decay heat removal.~~ LA.2

A.1

3/4.7 PLANT SYSTEMS

3/4.7.1 CORE STANDBY COOLING SYSTEM-EQUIPMENT COOLING WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.1 3.7.1.1 Two ~~independent~~ residual heat removal service water (RHRSW) system subsystems shall be OPERABLE, ~~with each subsystem comprised of:~~ LA.1

~~a. Two OPERABLE RHRSW pumps, and~~

~~b. An OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring the water through the associated RHR heat exchanger.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, ~~4 and 5~~. LA.2

ACTION:

Action A — a. In OPERATIONAL CONDITION 1, 2 or 3: 7 days L.1

1. With one RHRSW subsystem inoperable, restore the inoperable subsystem to OPERABLE status within ~~72 hours~~ or be in at least L.2

Action C — HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

restore one inoperable subsystem in 8 hours or

Action B — 2. With both RHRSW subsystems inoperable, be in at least HOT SHUTDOWN

Action C — within 12 hours and in COLD SHUTDOWN* within the next 24 hours. A.2

Notes to Required Actions A.1 and B.1 — b. In OPERATIONAL CONDITION 3 ~~or 4~~ with the RHRSW subsystem(s)# inoperable which is associated with an RHR shutdown cooling mode loop(s) required OPERABLE by Specification 3.4.9.1 ~~or 3.4.9.2, as applicable~~, declare the associated RHR shutdown cooling mode loop(s) inoperable and take the ACTION required to Specification 3.4.9.1 ~~or 3.4.9.2, as applicable~~. LA.2

~~c. In OPERATIONAL CONDITION 5 with the RHRSW subsystem# cooling mode loop(s) inoperable which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2.~~ LA.2

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 4.7.1.1 Each residual heat removal service water system subsystem shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position, or can be aligned to the correct position. A.3

~~*Whenever both RHRSW subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~ A.2

~~#Only one pump per subsystem need be OPERABLE if sufficient for decay heat removal.~~ LA.2

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The "*" footnote of CTS 3/4.7.1 is deleted since it provides unnecessary duplication of the ACTIONS required by proposed LCO 3.4.9, contains no additional restrictions on the operation of the plant and, in fact, could be interpreted as a relaxation of the requirements to achieve MODE 4 (COLD SHUTDOWN). The current and proposed ACTION to be taken in MODE 4 (proposed LCO 3.4.9) adequately prescribes the requirement to establish circulation by an alternate method (i.e., the duplicative requirement of the footnote). If conditions are such that MODE 4 cannot be attained, the ACTIONS remain in effect, essentially requiring that efforts to reach MODE 4 continue. Elimination of the footnote is an administrative presentation preference.

A.3 CTS 4.7.1.1 requires verification that each RHRSW subsystem valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. The RHRSW function is manually actuated (requiring valve lineup verification and repositioning as necessary and starting of the RHRSW pumps by the operator). In the CTS, it is recognized and interpreted that "in the correct position" allows the valves to be in the non-accident position provided they can be realigned to the correct position. In the ITS, the words "in the correct position" mean that the valves must be in the accident position, unless they are automatically aligned on an accident signal. Thus, to address the change in meaning, the additional words "or can be aligned to the correct position" have been added to CTS 4.7.1.1 (ITS SR 3.7.1.1) to clarify that it is permissible for the RHRSW System valves to be in the non-accident position and the subsystems to still be considered OPERABLE. Since this is only a clarification of the current requirement, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 3.7.1.1 details relating to system OPERABILITY, that the RHRSW subsystems shall be independent and that each subsystem shall have two RHRSW pumps capable of taking suction from the CSCS water tunnel and transferring the water to the associated RHR heat exchanger, are proposed to be relocated to the Bases. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. In addition, the requirements of the Surveillances will also help ensure these relocated details are maintained. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 CTS 3/4.7.1.1 provides LCO requirements, Actions, and Surveillance Requirements for the RHRSW System when in MODES 4 and 5. These requirements are proposed to be relocated to the Technical Requirements Manual (TRM). Since this system is a support system for other equipment with their own Specifications, the definition of OPERABILITY in ITS 1.1 will provide sufficient assurance the system can perform its required support function. In addition, the Bases for the supported systems will require the necessary portions of the RHRSW System to be OPERABLE. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference in the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 3.7.1.1 Action a.1 requires, when one RHRSW subsystem is inoperable, that the inoperable subsystem be restored to OPERABLE status within 72 hours. In ITS 3.7.1, when one RHRSW subsystem is inoperable, Required Action A.1 requires the inoperable subsystem to be restored to OPERABLE status in 7 days. This change provides additional time to restore the subsystem prior to requiring a plant shutdown. In this condition, the remaining OPERABLE RHRSW subsystem is capable of providing the required heat removal function. Analyses show the capacity and capability of the remaining RHRSW subsystem is such that adequate cooling is provided to each of the systems supported by the

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) RHRSW System. The proposed allowed outage time of 7 days in ITS 3.7.1 Required Action A.1 and the capability of the RHRSW System to perform its intended safety function, in the associated condition, is consistent with Technical Specification allowed outage time provided for restoration of both subsystems of RHR suppression pool cooling and both RHR suppression pool spray subsystems (systems supported by the RHRSW System in MODES 1, 2, and 3). Furthermore, since adequate RHRSW cooling is available to the supported loads (i.e., suppression pool cooling, suppression pool spray and RHR shutdown cooling) for the above described condition, this change also provides the benefit of avoiding the transient risk associated with an unnecessary plant shutdown. Therefore, the proposed change to the RHRSW System allowed outage time is acceptable.

L.2 CTS 3.7.1.1 Action a.2, when both RHRSW subsystems are inoperable, requires the plant to be placed in Hot Shutdown within 12 hours and in Cold Shutdown within the next 24 hours. ITS 3.7.1 Required Action B.1, when both RHRSW subsystems are inoperable, requires one RHRSW subsystem to be restored to OPERABLE status in 8 hours. This change provides additional time to restore one RHRSW subsystem, when both subsystems are inoperable, prior to requiring a plant shutdown. The 8 hour allowed outage time provided to restore one RHRSW subsystem to OPERABLE status is consistent with the allowed outage time provided for restoration of both subsystems of RHR suppression pool cooling and both RHR suppression pool spray subsystems (systems supported by the RHRSW System in MODES 1, 2, and 3). The allowed outage time is also acceptable due to the low probability of a DBA or transient occurring within this 8 hour period when both RHRSW subsystems are inoperable.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.7.2

PLANT SYSTEMS

DIESEL GENERATOR COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.2

3.7.1.2 The independent Unit 1 Division 1, 2 and 3 and the Unit 2 Division 2 diesel generator cooling water subsystems shall be OPERABLE ~~with each subsystem comprised of:~~ LA.1

~~a. One OPERABLE diesel generator cooling water pump, and~~

~~b. An OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring cooling water to the associated diesel generator.~~

APPLICABILITY: ~~When the diesel generator is required to be OPERABLE.~~ MODES 1, 2, and 3 LA.2

ACTION: Add proposed ACTIONS Note A.2

Action A

With one or more diesel generator cooling water subsystems inoperable, declare the associated diesel generator inoperable ~~and take the ACTION required by Specifications 3.8.1.1 or 3.8.1.2, as applicable~~. M.1 A.3 LA.2

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each of the above required diesel generator cooling water subsystems shall be demonstrated OPERABLE:

SR 3.7.2.1 a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. LD.1

SR 3.7.2.2 b. At least once per ~~18~~ 24 months by verifying that:

1. Each pump starts automatically upon receipt of a actual or simulated start signal for the associated diesel generator, and L.1 LA.3

2. The Division 1 pump each required starts automatically upon receipt of a actual or simulated start signal for the LPCS pump in Unit 1. LA.3 LA.3 L.1

Page 1 of 2

A.1 ITS 3.7.2

PLANT SYSTEMS

DIESEL GENERATOR COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.2 3.7.1.2 The ~~Independent~~ Unit 2 Division 1, 2 and 3 and the Unit 1 Division 2 diesel generator cooling water subsystems shall be OPERABLE ~~with each subsystem comprised of:~~ LA.1

~~a. One OPERABLE diesel generator cooling water pump, and~~

~~b. An OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring cooling water to the associated diesel generator.~~ LA.2

APPLICABILITY: ~~When the diesel generator is required to be OPERABLE.~~ MODES 1, 2, and 3

ACTION: Add proposed ACTIONS Note A.2

Action A With one or more diesel generator cooling water subsystems inoperable, declare the associated diesel generator inoperable ~~and take the ACTION required by Specifications 3.8.1.1 or 3.8.1.2, as applicable.~~ M.1 A.3 LA.2

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each of the above required diesel generator cooling water subsystems shall be demonstrated OPERABLE:

SR 3.7.2.1 a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SR 3.7.2.2 b. At least once per 18 24 months by verifying that: LD.1

1. Each pump starts automatically upon receipt of a actual or simulated start signal ~~for the associated diesel generator~~, and LA.3 L.1

2. The ~~Division 1~~ pump starts automatically upon receipt of a each required actual or simulated start signal ~~for the LPCS pump in Unit 2~~. LA.3 L.1

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Technical Specification (ISTS)).

A.2 An ITS 3.7.2 ACTIONS Note is proposed allowing separate Condition entry for each DGCW subsystem in order to provide more explicit instructions within the ITS format consistent with the existing CTS 3.7.1.2 Action for one or more inoperable DGCW subsystems. This change is intended to ensure that each occurrence of an inoperable DGCW subsystem be assessed in accordance with the applicable Conditions and Required Actions of the supported components. This is consistent with the intent of the CTS 3.7.1.2 Action. Since this change only provides more explicit direction of the current interpretation of the existing Specification, this change is considered administrative.

A.3 The CTS 3.7.1.2 Action requires action to be taken per CTS 3.8.1.1 when the diesel generator(s) are declared inoperable due to inoperable DGCW subsystem(s). The format of the ITS does not include providing "cross references." CTS 3.8.1.1 (ITS 3.8.1) adequately prescribes the necessary conditions for compliance without such references. Therefore, the existing reference to take the ACTION required by Specification 3.8.1.1 in the CTS 3.7.1.2 Action serves no functional purpose, and its removal is purely an administrative difference in presentation.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The current Applicability is whenever a diesel generator is required to be OPERABLE. The Applicability has been revised to MODES 1, 2, and 3 consistent with the Applicability of proposed ITS 3.8.1, "AC Sources-Operating," and ITS 3.5.1, "Emergency Core Cooling System (ECCS)-Operating." (The change to the DGCW requirements in MODES or conditions other than MODES 1, 2, and 3 is addressed in Discussion of Change LA.2.) This change is necessary since the unit DGCW subsystems support the OPERABILITY of the ECCS by cooling the ECCS cubicle area cooling coils as

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 (cont'd) well as the associated diesel generator. A commensurate change is also made to the CTS 3.7.1.2 Action for one or more DGCW subsystems inoperable. In this same condition, ITS 3.7.2 Required Action A.1 requires each of the components supported by the inoperable DGCW subsystem to be declared inoperable, not just the associated diesel generator.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.7.1.2 relating to system OPERABILITY (in this case that the DGCW subsystems will be independent and each subsystem shall have one OPERABLE DGCW pump, and an OPERABLE flow path capable of taking suction from the CSCS water tunnel and transferring water to the associated diesel generator) are proposed to be relocated to the Bases. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. In addition, the requirements of the Surveillance will also help ensure these relocated details are maintained. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program.

LA.2 CTS 3.7.1.2 provides LCO requirements, Actions, and Surveillance Requirements for the DGCW System when the diesel generator is required to be OPERABLE. These requirements, when in MODES or conditions other than MODE 1, 2, or 3, are proposed to be relocated to the Technical Requirements Manual (TRM). Since this system is a support system for other equipment with their own Specifications, the definition of OPERABILITY in ITS 1.1 will provide sufficient assurance the system can perform its required support function. In addition, the Bases for the supported systems will require the necessary portions of the DGCW System to be OPERABLE. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference in the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.3 CTS 4.7.1.2.b.1 requires verification that each pump starts automatically upon receipt of a start signal for the associated diesel generator. CTS 4.7.1.2.b.2 requires verification that the Division 1 DGCW pump starts automatically upon receipt of a start signal for the LPCS pump. ITS SR 3.7.2.2 simply requires the verification of the capability of each DGCW pump to start upon each required initiation signal. The details regarding the specific start signals to be used during the Surveillance are relocated to the Bases. ITS 3.7.2.2 will continue to ensure that each of the DGCW pumps is capable of actuating on each required start signal. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. The Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LD.1 The CTS 4.7.1.2.b (proposed SR 3.7.2.2) Frequency for performing the DGCW automatic start surveillance is proposed to be extended from 18 months to 24 months. ITS SR 3.7.2.2 verifies each DGCW pump starts automatically on each required actual or simulated initiation signal. Extending this surveillance is acceptable in part because this requirement is also verified on a more frequent basis, e.g., every 31 days when performing SR 3.8.1.2 during diesel generator start testing and every 92 days during LPCS pump start for the Inservice Testing Program. This testing will detect significant failures affecting system operation that would be detected by conducting the 24 month surveillance test. Reviews of historical maintenance and surveillance data have shown that this test normally passes its Surveillance at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to extended Surveillance Frequency will be minimal. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis. This evaluation is consistent with the requirements of Generic Letter 91-04, which provided NRC guidance on extending Surveillance Frequencies from 18 months to 24 months to accommodate longer fuel cycles.

"Specific"

L.1 The phrase "actual or simulated" in reference to CTS 4.7.1.2.b.1 and b.2 requirements for a start signal, is proposed to be added to ITS SR 3.7.2.2 for verifying that the DGCW System actuates on each of the required start signals. This allows actual or simulated automatic DGCW System actuations to be used to fulfill the Surveillance Requirement. OPERABILITY is adequately demonstrated in either case since the DGCW System cannot discriminate between "actual" or "simulated" start signals, and ensures that the required automatic start capability is demonstrated.

RELOCATED SPECIFICATIONS

None

ITS 3.7.3

ULTIMATE HEAT SINK A.1

LIMITING CONDITION FOR OPERATION

LCO 3.7.3 3.7.1.3 The CSCS pond shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, ~~4, 5, and *~~ LA.1

Action A ACTION: With the CSCS pond inoperable due to sediment deposition in excess of limit or pond bottom elevation greater than limit, restore the pond to OPERABLE status within 90 days or: M.1

Action B a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Add 2nd portion of Condition B M.1

~~b. In OPERATIONAL CONDITION 4, 5, or *, declare the RHRSW system and the diesel generator cooling water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.~~ LA.1

SURVEILLANCE REQUIREMENTS

4.7.1.3 The CSCS pond shall be determined OPERABLE at least once per 18 24 months by determining that: LD.1

SR 3.7.3.2 a. No sediment deposition in excess of 1.5 foot has occurred in the intake flume or in the CSCS pond ~~as determined by a series of sounding cross-sections compared to as-built soundings~~. LA.2

SR 3.7.3.3 b. The pond bottom elevation is less than or equal to 686.5 feet.

Add proposed SR 3.7.3.1 M.1

~~*When handling irradiated fuel in the secondary containment.~~ LA.1

Page 1 of 2

ULTIMATE HEAT SINK

A.1

ITS 3.7.3

LIMITING CONDITION FOR OPERATION

LCO 3.7.3 3.7.1.3 The CSCS pond shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, ~~4, 5, and *~~. LA.1

M.1

ACTION A ACTION: With the CSCS pond inoperable, due to sediment deposition in excess of limit or pond bottom elevation greater than limit restore the pond to OPERABLE status within 90 days or:

ACTION B a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Add 2nd portion of Condition B M.1

~~b. In OPERATIONAL CONDITION 4, 5, or *, declare the RHRSW system and the diesel generator cooling water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.~~ LA.1

SURVEILLANCE REQUIREMENTS

4.7.1.3 The CSCS pond shall be determined OPERABLE at least once per 18 24 months by determining that: LD.1

SR 3.7.3.2 a. No sediment deposition in excess of 1.5 foot has occurred in the intake flume or in the CSCS pond ~~as determined by a series of sounding cross-sections compared to as-built soundings~~. LA.2

SR 3.7.3.3 b. The pond bottom elevation is less than or equal to 686.5 feet.

Add proposed SR 3.7.3.1 M.1

~~*When handling irradiated fuel in the secondary containment.~~ LA.1

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A new Surveillance Requirement (ITS SR 3.7.3.1) is added to CTS 4.7.1.3 to require verification that the UHS (CSCS pond) average water temperature is ≤ 97°F every 24 hours. This maximum UHS temperature is assumed in the LaSalle design basis accident (DBA) analyses. The addition of this Surveillance Requirement represents an additional restriction on plant operation necessary to help ensure the OPERABILITY of the UHS and the heat removal capabilities of the Residual Heat Removal Service Water System and the Diesel Generator Cooling Water System are maintained within the assumptions of the DBA analyses.

When the CSCS pond is inoperable, the Action of CTS 3.7.1.3 provides a 90 day period to restore the CSCS pond to OPERABLE status. In ITS 3.7.3, the 90 day period for restoration of the CSCS pond has been maintained when the inoperability is due to sediment deposition exceeding the required limit or pond bottom depth exceeding the limit. For other inoperabilities of the CSCS pond (e.g., average water temperature not within limit), ITS 3.7.3, Required Action B.1 and B.2 will require the plant to be in MODE 3 within 12 hours and in MODE 4 within 36 hours. This change to the actions associated with an inoperable CSCS pond represents an additional restriction on operation necessary to help ensure that actions taken in the event of a loss of function associated with the Ultimate Heat Sink are maintained consistent with the actions required for a loss of function associated with the systems and components supported by the CSCS pond.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 3/4.7.1.3 provides LCO requirements, Actions, and Surveillance Requirements for the CSCS pond when in MODES 1, 2, 3, 4, and 5, and when handling irradiated fuel in the secondary containment. These requirements, when in MODES or conditions other than MODE 1, 2, or 3, are proposed to be relocated to the Technical Requirements Manual (TRM). Since the CSCS pond supports the OPERABILITY of other equipment with their own Specifications, the definition of OPERABILITY in ITS 1.1 will provide sufficient assurance the CSCS pond can perform its required support function. In addition, the Bases for the supported systems will require the CSCS pond (i.e., the Ultimate Heat Sink) to be OPERABLE. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference in the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The CTS 4.7.1.3.a (ITS SR 3.7.3.2) details of the methods for determining the level of sediment deposition in the CSCS pond (by a series of sounding cross-sections compared to as-built soundings) are to be relocated to the Bases. These details are not necessary to ensure the OPERABILITY of the CSCS pond. The requirements of ITS 3.7.3 and associated Surveillance Requirements are adequate to ensure the CSCS pond is maintained OPERABLE. Therefore, these details are not required to be in the Technical Specifications to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

LD.1 The Frequencies for performing CTS 4.7.1.3.a and 4.7.1.3.b (ITS SRs 3.7.3.2 and 3.7.3.3, respectively) have been extended from 18 to 24 months. The determination of sediment deposition and the pond bottom elevation ensure that the volume of water in the CSCS pond will be adequate to support long term cooling for a 30 day period after a DBA. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed Specification 3.0.2). This proposed change was evaluated in accordance with the guidance provided in

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillance at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be small. Reviews of historical maintenance and surveillance data have shown that these tests pass their surveillance at the current frequency. A hydrographic survey of the UHS was performed in 1997. This survey found that amount of sediment that accumulated from the time of original construction to the survey date as being negligible. This means that negligible sediment accumulated over 15 years (The LaSalle Unit 1 Operating License was issued in April 1982, the survey was performed in November 1997).

Furthermore, both units of LaSalle were shutdown for approximately two years. During this time cooling water flow through the CSCS pond was minimal. Minimal flow corresponds to minimal flow velocity. Since the capability of the flow to transport suspended solids is dependent upon the density of the suspended solids and flow velocity, maximum sediment accumulation is expected to occur when flow velocity is minimum. Another hydrographic survey of the UHS was performed in 1999 and found virtually no difference in the volume of the UHS from the 1997 survey. From this it is concluded that the sediment deposition during the extended shutdown period was negligible.

In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

"Specific"

None

RELOCATED SPECIFICATIONS

None

3/4.7.2 CONTROL ROOM AND AUXILIARY ELECTRIC EQUIPMENT ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

LA.1

LCO 3.7.4 3.7.2 Two independent control room and auxiliary electric equipment room emergency filtration system trains shall be OPERABLE[#] A.2 L.1

APPLICABILITY: All OPERATIONAL CONDITIONS and *. MODES 1, 2, and 3 During CORE ALTERATIONS During OPDRVs

ACTION:

Action A a. With one emergency filtration system train inoperable, restore the inoperable train to OPERABLE status within 7 days or:

Action B 1. In OPERATIONAL CONDITIONS 1, 2, 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

During CORE ALTERATIONS, OPDRVs L.1

Action C 2. ~~In OPERATIONAL CONDITION 4, 5~~ or *, initiate and maintain operation of the OPERABLE emergency filtration system in the pressurization mode of operation. Add proposed Required Actions C.2.1, C.2.2, + C.2.3

Add proposed ACTION D

A.3

Action E b. With both emergency filtration system trains inoperable ~~in OPERATIONAL CONDITION 4, 5~~ or *, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. During CORE ALTERATIONS, OPDRVs, L.1 initiate action to suspend A.4

Note to Action E c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room and auxiliary electric equipment room emergency filtration system train shall be demonstrated OPERABLE:

a. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ L.2

SR 3.7.4.1 1. Operate each Control Room and Auxiliary Electric Equipment Room Emergency Filter System for greater than or equal to 10 continuous hours with the heaters operating, and

SR 3.7.4.2 2. Manually initiating flow through the control room and auxiliary electric equipment room recirculation filters for at least 10 hours.

Applicability *When irradiated fuel is being handled in the secondary containment.

~~#The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4, 5 or *.~~ A.2

A.1

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.4.3 b. Perform required control room and auxiliary electric equipment room filter testing in accordance with, and at the frequency specified by, the Ventilation Filter Testing Program.

c. Deleted. 24 LD.1

SR 3.7.4.4 SR 3.7.4.5 d. At least once per 18 months by:

1. Deleted.

SURVEILLANCE REQUIREMENTS (Continued)

actual or L.3 LA.2 LA.4

SR 3.7.4.4 2. Verifying that on each of the below pressurization mode actuation test signals, the emergency train ~~automatically switches to the pressurization mode of operation.~~ Manually initiate flow through the control room and auxiliary electric equipment room recirculation filters line and then verify that the control room LA.3
SR 3.7.4.5 and auxiliary electric equipment rooms are maintained at a positive pressure of greater than or equal to 1/8 inch W.G. relative to the adjacent areas during emergency train operation at a flow rate less than or equal to 4000 cfm: actuates

a) Outside air smoke detection, and LA.2

b) Air intake radiation monitors. LA.4

3. Deleted.

e. Deleted.

f. Deleted.

3/4.7.2 CONTROL ROOM AND AUXILIARY ELECTRIC EQUIPMENT ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.4 3.7.2 Two ~~independent~~ control room and auxiliary electric equipment room emergency filtration system trains shall be OPERABLE.[@] LA.1 A.2

APPLICABILITY: ~~All OPERATIONAL CONDITIONS~~ and *. MODES 1, 2, and 3 During CORE ALTERATIONS During OPDRVs L.1

ACTION:

ACTION A a. With one emergency filtration system train inoperable, restore the inoperable train to OPERABLE status within 7 days or:

ACTION B 1. In OPERATIONAL CONDITIONS 1, 2, 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

During CORE ALTERATIONS, OPDRVs, L.1

ACTION C 2. ~~In OPERATIONAL CONDITION 4, 5~~ or *, initiate and maintain operation of the OPERABLE emergency filtration system in the pressurization mode of operation. Add proposed Required Actions C.2.1, C.2.2 + C.2.3

Add proposed ACTION D A.3

ACTION E b. With both emergency filtration system trains inoperable, ~~in OPERATIONAL CONDITION 4, 5~~ or *, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. initiate action to suspend A.4

NOTE TO ACTION E c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room and auxiliary electric equipment room emergency filtration system train shall be demonstrated OPERABLE:

a. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ L.2

SR 3.7.4.1 1. Operate each Control Room and Auxiliary Electric Equipment Room Emergency Filter System for greater than or equal to 10 continuous hours with the heaters operating, and

SR 3.7.4.2 2. Manually initiating flow through the control room and auxiliary electric equipment room recirculation filters for at least 10 hours.

Applicability *When irradiated fuel is being handled in the secondary containment.

~~@The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4, 5 or *.~~ A.2

A.1

ITS 3.7.4

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.4.3 b. Perform required control room and auxiliary electric equipment room filter testing in accordance with, and at the frequency specified by, the Ventilation Filter Testing Program.

c. Deleted. 24 LD.1

SR 3.7.4.4
SR 3.7.4.5 d. At least once per 18 months by:

1. Deleted.

 A.1 ITS 3.7.4

SURVEILLANCE REQUIREMENTS (Continued)

L.3 actual or LA.2

SR 3.7.4.4 2. Verifying that on each of the below pressurization mode actuation test signals, the emergency train ~~automatically switches to the pressurization mode of operation.~~ Manually initiate flow through the control room and auxiliary electric equipment room recirculation filters and then verify that the control room and auxiliary electric equipment rooms are maintained at a positive pressure of SR 3.7.4.5 greater than or equal to 1/8 inch W.G. relative to the adjacent areas during emergency train operation at a flow rate less than or equal to 4000 cfm: LA.4 LA.3 actuates LA.4

a) ~~Outside air smoke detection, and~~ LA.2

b) ~~Air intake radiation monitors.~~ LA.4

3. Deleted.

e. Deleted.

f. Deleted.

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Technical Specification (ISTS)).

A.2 CTS 3.7.2 footnote #, which provides a reference that the normal or emergency power source may be inoperable in Operational Condition 4, 5, or when irradiated fuel is being handled in Secondary Containment, has been deleted. This reference is an explicit part of the definition of OPERABLE-OPERABILITY, as defined in ITS 1.1, "Definitions." There is no need to duplicate this reference in ITS 3.7.4. Therefore, deletion of CTS 3.7.2 footnote # is an administrative change.

A.3 In CTS 3.7.2, no Actions are provided for when two CRAF subsystems are inoperable in MODES 1, 2, and 3. Therefore, CTS 3.0.3 would be applicable and would be required to be entered. In ITS 3.7.4, a new ACTION D has been added to direct entry into LCO 3.0.3 if both CRAF subsystems are inoperable in MODE 1, 2, or 3. This avoids confusion as to the proper ACTION if in MODE 1, 2, or 3 and simultaneously in a special condition, such as handling irradiated fuel assemblies in the secondary containment. Since this ACTION results in the same ACTION as the current Technical Specifications, this change is administrative.

A.4 CTS 3.7.2 ACTION b to "suspend...operations with a potential for draining the reactor vessel" may not be possible for all plant conditions. In such a condition, the existing ACTION results in "non-compliance with the Technical Specifications" and a requirement for an LER. The intent of the ACTION is more appropriately presented in ITS 3.7.4 Required Action E.3. With the proposed Required Action, a requirement to immediately initiate action to suspend OPDRVs is imposed. Included in this Required Action is the understanding that best efforts to suspend OPDRVs must continue until they are suspended, which is how the current ACTION is implemented. However, with this Required Action, if the suspension of OPDRVs cannot be accomplished immediately, no LER will be required.

This interpretation of the ACTIONS intent is supported by the BWR Standard Technical Specifications, NUREG-1434, Rev. 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail of CTS 3.7.2 relating to system design (that the CRAF subsystems are "independent") is proposed to be relocated to the Bases. This is a design detail that is not necessary to be included in the Technical Specifications to ensure the OPERABILITY of the CRAF System since OPERABILITY requirements are adequately addressed in ITS 3.7.4. As such, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 CTS 4.7.2.d.2 provides verification that the CRAF System automatically switches to the pressurization mode of operation on detection of smoke in an outside air intake. This requirement is proposed to be relocated to the Technical Requirements Manual (TRM). The CRAF System actuation on detection of smoke in an outside air intake functions to permit continuous occupancy of the control room area during an external smoke event. However, this smoke protection mode of the CRAF System is not assumed to mitigate a DBA or transient since smoke intrusion is not a DBA or transient. None of the four NRC Policy Statement criteria are applicable to this requirement. Therefore, moving these requirement to the TRM is appropriate and consistent with the NRC Policy Statement and 10 CFR 50.36. As a result, these requirements are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference into the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled in accordance with the provisions of 10 CFR 50.59.

LA.3 CTS 4.7.2.d.2 contains details regarding the methodology for performing a surveillance to verify the ability of each CRAF System to maintain a positive pressure in the control room and auxiliary electric equipment rooms (i.e., control room area) relative to the adjacent areas during emergency train operation at a specified flow rate. This information is to be relocated to the Bases. ITS SR 3.7.4.5 will continue to ensure that the test is performed, and the applicable acceptance criteria met. Thus, these details for performing the surveillance are not required to be maintained in the Technical Specifications to protect public health and safety. Changes to the Bases will be controlled in accordance with the Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.4 The CTS 4.7.2.d.2 (ITS SR 3.7.4.4) details of the methods for performing the CRAF System actuation test (the source of the signal used for automatic actuation and that the CRAF subsystems automatically switch to the pressurization mode of operation) are to be relocated to the Bases. These details are not necessary to ensure the OPERABILITY of the CRAF System. The requirements of ITS 3.7.4 and SR 3.7.4.4, which verifies that each CRAF subsystem actuates on an actual or simulated signal, are adequate to ensure the CRAF System is maintained OPERABLE. Therefore, these details are not required to be in the Technical Specifications to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

LD.1 The Frequency for performing CTS 4.7.2.d.2 (proposed SR 3.7.4.4 and SR 3.7.4.5) has been extended from 18 months to 24 months. This SR ensures that each CRAF subsystem is capable of automatic initiation and that the mechanical components operate as designed on system actuation (e.g., fans start, valves and dampers open or close as required), and that the control room area boundary leakage is within the capacity of the CRAF System by demonstrating that control room area can be maintained at a positive pressure with respect to adjacent areas when in the pressurization mode of operation.

The proposed change will allow this Surveillance to extend the Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24-month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed Specification 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that this test normally passes the Surveillance at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be small.

The CRAF System will be tested every 31 days according to proposed SR 3.7.4.1 and SR 3.7.4.2, therefore, any significant mechanical component failures will be detected and repaired during plant operation. This more frequent testing although it does not test the actual initiation signal verifies the OPERABILITY of the majority of the CRAF System circuitry. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) related to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3, surveillance intervals from 18 to 24 months:

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd)

"Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

Extending the surveillance interval for this verification of control room area boundary integrity is acceptable because the control room area boundary is maintained at a positive pressure during normal operation. Therefore, any substantial degradation of the boundary that would prevent maintaining the control room area at the required pressure during an accident will be evident prior to the scheduled performance of these tests.

Based on the results of the review of the historical maintenance and surveillance data and the ability to detect significant failures during plant operation, the impact, if any, of this change on system availability is minimal. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis.

"Specific"

L.1 The Applicability of CTS 3.7.2 is revised from All Operational Conditions (i.e., Operational Conditions 1, 2, 3, 4, and 5) and during movement of irradiated fuel assemblies in secondary containment to MODES 1, 2, and 3; during movement of irradiated fuel assemblies in secondary containment; during CORE ALTERATIONS; and during operations with the potential for draining the reactor vessel (OPDRVs) in ITS 3.7.4. The CRAF System is required to be OPERABLE to control operator exposure during and following a design basis accident, since a design basis accident could lead to a fission product release. When the plant is in MODE 4 or 5, the probability and consequences of a design basis accident are reduced due to the temperature and pressure limitations in these MODES. However, in MODE 4 or 5, activities are conducted for which significant releases of radioactivity are postulated. Therefore, the CRAF System is only required to be OPERABLE in MODE 4 or 5, when activities are in progress which could, if an event occurs, result in significant releases of

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) radioactivity (i.e., during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs). This change modifies the CTS 3.7.2 Mode 4 and 5 Applicability to include only these activities. This change is considered acceptable since it is consistent with the intent of CTS 3.7.2 ACTION b (in Mode 4 and 5 with two CRAF subsystems inoperable, CTS 3.7.2 ACTION b requires suspension of those activities for which significant releases of radioactivity are postulated). This change allows operations that do not have a potential for a significant radioactive release to be performed without requiring the CRAF System to be OPERABLE and provides additional scheduling flexibility during plant refueling outages. In addition, due to this change, CTS 3.7.2 Action a.2 has been modified to allow exiting the new Applicability (by suspending these activities) in lieu of operating an OPERABLE CRAF subsystem in the pressurization mode.

L.2 CTS 4.7.2.a requires the CRAF System to be operated every 31 days on a STAGGERED TEST BASIS. Proposed SR 3.7.4.1 and SR 3.7.4.2 do not include the STAGGERED TEST BASIS requirement. The intent of a requirement for staggered testing is to increase reliability of the component/system being tested. A number of reviews/evaluations have been performed which have demonstrated that staggered testing has negligible impact on component reliability. As a result, it has been determined that staggered testing 1) is operationally difficult, 2) has negligible impact on component reliability, 3) is not as significant as initially thought, and 4) has no impact on failure frequency. Therefore, the CRAF staggered testing requirements have been deleted. Since the Frequency is not affected, i.e., both CTS and ITS require monthly testing for each subsystem, and staggered testing has a negligible impact on component reliability, this requirement has been deleted.

L.3 The phrase "actual or," in reference to the actuation test signal in CTS 4.7.2.d.2, has been added to proposed SR 3.7.4.4, which verifies that each CRAF subsystem actuates on an actuation test signal. This allows satisfactory automatic CRAF System initiations for other than surveillance purposes to be used to fulfill the Surveillance Requirement. Operability is adequately demonstrated in either case since the CRAF subsystem itself cannot discriminate between "actual" or "test" signals.

RELOCATED SPECIFICATIONS

None

Insert New Specification 3.7.5

Insert new Specification 3.7.5, "Control Room Area Ventilation Air Conditioning (AC) System," as shown in proposed ITS 3.7.5. M.1

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Proposed ITS 3.7.5 is a new Specification that defines requirements for OPERABILITY of the Control Room Area Ventilation Air Conditioning (AC) System. This system is comprised of two subsystems, each containing a Control Room AC subsystem and the Auxiliary Electric Equipment Room (AEER) AC subsystem. The Specification requires both sets of associated subsystems to be OPERABLE in MODES 1, 2, and 3, during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, and during OPDRVs. This change is based on ISTS 3.7.4. However, the ISTS SR 3.7.4.1 is not adopted. The LaSalle control room and AEER AC subsystems have air cooled condensers and refrigerant compressors. While an appropriate testing methodology has been developed for systems with water cooled chillers, the Nuclear HVAC Utilities Group (NHUG) has not yet developed a capacity verification test methodology for systems with air cooled condensers. Therefore, alternate testing is proposed similar to testing previously approved for ComEd's Zion Nuclear Power Station. This new Specification imposes additional restrictions upon plant operations adequate to ensure the OPERABILITY of components in the control room in a post-accident environment.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

None

RELOCATED SPECIFICATIONS

None

A.1

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

after decay of 30 minutes A.2

LCO 3.7.6 3.11.2.2 The release rate of the sum of the activities from the noble gases measured prior to the holdup line shall be limited to less than or equal to 3.4 x 10^5 microcuries/second.

and L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3. with any main steam line not isolated and steam jet air ejector (SJAE) in operation

ACTION:

ACTION A With the release rate of the sum of the activities of the noble gases prior to the holdup line exceeding 3.4 x 10^5 microcuries/second restore the release rate to within its limit within 72 hours or
ACTION B be in at least STARTUP with the main steam isolation valves closed within the next ~~6~~ 12 hours. L.2

Add proposed Required Action B.2 L.1

Add proposed Required Actions B.3.1 and B.3.2 L.3

SURVEILLANCE REQUIREMENTS

~~4.11.2.2.1 The radioactivity rate of noble gases prior to the holdup line shall be continuously monitored in accordance with the ODCM.~~ LA.1

SR 3.7.6.1 4.11.2.2.2 The release rate of the sum of the activities from noble gases ~~prior to the holdup line~~ shall be determined to be within the limits of Specification 3.11.2.2 at the following frequencies ~~by performing an isotopic analysis of a representative sample of gases taken prior to the holdup line~~. LA.2

a. At least once per 31 days.

b. Within 4 hours following an increase, ~~as indicated by the off gas pre-treatment Noble Gas Activity Monitor,~~ of greater than or equal to 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant. LA.2 M.1

Add proposed SR 3.7.6.1 Note L.4

Page 1 of 2

RADIOACTIVE EFFLUENTS

A.1

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

after decay of 30 minutes A.2

LCO 3.7.6 3.11.2.2 The release rate of the sum of the activities from the noble gases measured prior to the holdup line shall be limited to less than or equal to 3.4 x 10^5 microcuries/second.

and L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3. with any main steam line not isolated and steam jet air ejector (SJAE) in operation

ACTION:

ACTION A — With the release rate of the sum of the activities of the noble gases prior to the holdup line exceeding 3.4 x 10^5 microcuries/second restore the release rate to within its limit within 72 hours or be in at least STARTUP with the

ACTION B — main steam isolation valves closed within the next ~~6~~ 12 hours. L.2

Add proposed Required Action B.2 L.1

SURVEILLANCE REQUIREMENTS

Add proposed Required Actions B.3.1 and B.3.2 L.3

~~4.11.2.2.1 The radioactivity rate of noble gases prior to the holdup line shall be continuously monitored in accordance with the ODCM.~~ LA.1

SR 3.7.6.1 4.11.2.2.2 The release rate of the sum of the activities from noble gases prior to the holdup line shall be determined to be within the limits of Specification 3.11.2.2 at the following frequencies ~~by performing an isotopic analysis of a representative sample of gases taken prior to the holdup line.~~ LA.2

a. At least once per 31 days.

b. Within 4 hours following an increase~~, as indicated by the off gas pre-treatment Noble Gas Activity Monitor,~~ LA.2 of greater than or equal to M.1 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

Add proposed SR 3.7.6.1 Note L.4

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Technical Specification (ISTS)).

A.2 CTS 3.11.2.2 requires the radioactivity rate of noble gases downstream of the recombiner to be ≤ 340,000 microcuries/second. The accident analysis (UFSAR, Section 15.7) that this radioactivity rate is based on also assumes that the radioactivity rate is after a 30 minute decay period. Therefore, addition of the 30 minute decay period in the LCO, only provides clarification to the parameters in use, and as such, this change is considered administrative only.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.11.2.2.2.b requires verification that the release rate of noble gases prior to the holdup line is within limits within 4 hours following an increase of > 50%. The amount of increase is changed to include an increase equal to 50% in ITS SR 3.7.6.1. This is an inconsequential change that is considered more restrictive because technically it increases the range of releases to be considered. However, no additional performances of the Surveillance would be expected since the increase is insignificant.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 4.11.2.2.1 requirement to continuously monitor radioactivity rate of noble gases prior to the holdup line is proposed to be relocated to the Offsite Dose Calculation Manual (ODCM). This relocated requirement is not necessary to be included in the Technical Specifications to assure that main condenser offgas activity rate is within limits. Proposed SR 3.7.6.1 provides adequate assurance the main condenser offgas activity rate is within limits. The ODCM currently contains requirements to provide this monitoring of the main condenser air ejector activity release rate. As such, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the ODCM will be controlled by the provisions of the ODCM Control Process described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The CTS 4.11.2.2.2 details defining the methods for performing this Surveillance, the location of the sample, and method for determining when an increase has occurred are proposed to be relocated to the Bases. These details are not necessary to ensure the main condenser offgas activity rate limits are maintained. The requirements of ITS 3.7.6 and SR 3.7.6.1 are adequate to ensure the main condenser offgas activity rate is maintained within limits. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 The Applicability of CTS LCO 3.11.2.2 is OPERATIONAL CONDITIONS 1, 2, and 3. In the event the requirement of CTS LCO 3.11.2.2 is not met, the Action requires compliance be restored within 72 hours, or the plant placed in at least STARTUP (i.e., MODE 2) with the main steam isolation valves closed within the next 6 hours. Thus, the CTS actually permits operation in MODES 2 and 3 to continue as long as the main steam isolation valves are closed. The Applicability is changed to MODE 1 and MODES 2 and 3 with any main steam line not isolated and the steam jet air ejector (SJAE) in operation in proposed ITS 3.7.6. This proposed change is less restrictive, because the requirement will not be applicable in MODES 2 and 3 if the SJAE is not in operation regardless of the position of the main steam isolation valves. The main condenser offgas gross gamma activity limit is an initial condition of the main condenser offgas system failure event. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits of 10 CFR 100. With the main steam lines isolated or the SJAE not in operation, the offgas system is not being used to process the gross gamma activity; it is essentially maintained within the reactor coolant. Therefore, the event cannot occur. In addition, a new Required Action (ITS 3.7.6 Required Action B.2), which requires isolation of the air ejector has also been added consistent with this change to the Applicability.

L.2 The default action of the CTS 3.11.2.2 Action requires the main steam isolation valves to be closed in 6 hours if the main condenser offgas activity release rate for noble gases is not restored to within its limit within the Completion Time of 72 hours (ITS 3.7.6 Required Action A.1). The proposed Completion Time (ITS 3.7.6 Required Actions B.1 and B.2) to be outside the Applicability of the Specification has been extended from 6 hours to 12 hours. This proposed time is

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) required to shutdown and cooldown the unit from full power conditions and isolated the main steam isolation valves in an orderly manner and without challenging unit systems. This proposed time is considered reasonable based on operating experience and is consistent with the BWR ISTS, NUREG-1434, Rev. 1. Allowing 12 hours to complete the Required Actions is an acceptable exchange in risk; the risk of an event occurring during the additional period provided to exit the Applicability, versus the potential risk of unit upset that could challenge safety systems resulting from a rapid shutdown.

L.3 The CTS 3.11.2.2 Action requires the plant to be in at least STARTUP with the main steam isolation valves closed within 6 hours if the main condenser offgas activity is not restored to within limits within 72 hours. Alternative default Required Actions have been added to place the plant in a condition outside the Applicability of the Specification. ITS 3.7.6 Required Actions B.3.1 and B.3.2 will require the plant to be in MODE 3 in 12 hours and MODE 4 in 36 hours. This change is less restrictive since it provides optional actions to be taken for placing the plant in a condition that is outside the Applicability. In addition, the time to place the plant in a condition outside the Applicability is 36 hours instead of 6 hours as currently required by the CTS 3.11.2.2 Action (see Discussion of Change L.2 for further changes to the 6 hour Completion Time). This Specification is not required in MODE 4 since the main steam is not being exhausted to the main condenser, therefore the assumptions of a Main Condenser Offgas System failure event will still be bounded by the current analyses. Therefore, the proposed Required Action to be in MODE 4 is acceptable since the assumptions of the accident analysis will be preserved. The proposed Completion Times are consistent with other Specifications which require the plant to be in MODE 3 then MODE 4. The Completion Times are acceptable, based on operating experience, to reach the required plant conditions from full power conditions in a orderly manner and without challenging plant systems.

L.4 CTS 4.11.2.2.2 requires the main condenser offgas activity to be periodically determined. Proposed ITS SR 3.7.6.1 requires the performance of this Surveillance at the same Frequency however it is proposed to allow the Surveillance to not be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. This determination is only meaningful with one or more main steam lines not isolated and the SJAE in operation. Only in this condition can radioactive gases be in the Main Condenser Offgas System at significant rates. The 31 day period is an acceptable time to establish conditions appropriate for data collection and evaluation and is considered acceptable given the availability of instrumentation to monitor the offgas activity release rate.

RELOCATED SPECIFICATIONS

None

A.1

PLANT SYSTEMS

3/4.7.10 MAIN TURBINE BYPASS SYSTEM

Add proposed 2nd Part of LCO 3.7.7 A.2

LIMITING CONDITION FOR OPERATION

LCO 3.7.7 3.7.10 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. A.3

ACTION:

With the main turbine bypass system inoperable:

ACTION A
1. If at least four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:
   a) Within 2 hours, either:
      1) Restore the system to OPERABLE status, or

LCO 3.7.7
      2) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) to the main turbine bypass inoperable value per Specification 3.2.3.

ACTION B
   b) Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

ACTION A
2. If less than four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:

restore the system to OPERABLE status or A.4

LCO 3.7.7
   a) Within 2 hours increase the MCPR LCO to the main turbine bypass inoperable value per Specification 3.2.3, and
   ~~b) Within the next 12 hours restore the system to OPERABLE status~~ L.1

ACTION B
   c) Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

SR 3.7.7.1
   a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel.

PLANT SYSTEMS (Continued)

A.1

3/4.7.10 MAIN TURBINE BYPASS SYSTEM

SURVEILLANCE REQUIREMENTS

b. ~~18~~ 24 months by: LD.1

SR 3.7.7.2 1. Performing a system functional test ~~which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position~~. LA.1

SR 3.7.7.3 2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME is within limits ~~to be less than or equal to 200 milliseconds~~ LA.2

PLANT SYSTEMS

A.1 ITS 3.7.7

3/4.7.10 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.7 3.7.10 The main turbine bypass system shall be OPERABLE. Add proposed 2nd part of LCO 3.7.7

APPLICABILITY: ~~OPERATIONAL CONDITION 1,~~ when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. A.2 A.3

ACTION:

With the main turbine bypass system inoperable:

ACTION A
1. If at least four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:

   a) Within 2 hours, either:

      1) Restore the system to OPERABLE status, or

LCO 3.7.7
      2) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) to the main turbine bypass inoperable value per Specification 3.2.3.

ACTION B
   b) Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

ACTION A
2. If less than four bypass valves are capable of accepting steam flow per Surveillance 4.7.10.a:

   a) Within 2 hours restore the system to OPERABLE status or increase the MCPR LCO to the main turbine bypass inoperable value per Specification 3.2.3, and A.4

LCO 3.7.7

   ~~b) Within the next 12 hours restore the system to OPERABLE status.~~ L.1

ACTION B
   c) Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

SR 3.7.7.1
   a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel.

   b. ~~18~~ 24 months by: LD.1

SR 3.7.7.2
      1. Performing a system functional test ~~which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position~~. LA.1

SR 3.7.7.3
      2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME ~~to be less than or equal to 200 milliseconds~~ is within limits LA.2

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Technical Specification (ISTS)).

A.2 CTS 3.7.10 requires the Main Turbine Bypass System to be OPERABLE. The purpose of the Main Turbine Bypass System is to help ensure a MCPR Safety Limit Violation will not occur due to a feedwater transient. Therefore, an additional LCO option has been added to CTS 3.7.10 to permit a MCPR penalty to be applied in lieu of maintaining the Main Turbine Bypass System OPERABLE. This is consistent with the current licensing basis as indicated in CTS 3.7.10, Actions 1.a)2) and 2.a). The MCPR penalty is specified in the COLR, similar to other MCPR penalties. This change in format is consistent with the BWR ISTS, NUREG-1434, Rev. 1.

A.3 The Applicability of CTS 3.7.10 is "OPERATIONAL CONDITION 1 when THERMAL POWER is 25% or more of RATED THERMAL POWER." With THERMAL POWER ≥ 25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state in the Applicability of CTS 3.7.10 (ITS 3.7.7).

A.4 In the event less than four main turbine bypass valves are capable of accepting steam flow, Action 2.a) of CTS 3.7.10 requires the MINIMUM CRITICAL POWER RATIO to be increased to the main turbine bypass value per CTS 3.2.3 within 2 hours. The option to restore the Main Turbine Bypass System to an OPERABLE status within 2 hours has been added in proposed ITS 3.7.7, Action A (i.e., satisfy the requirements of the LCO). This change is considered to be administrative since restoring compliance with the LCO (per CTS 3.0.2) is always an option, whether or not it is specifically stated in the Actions.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details relating to methods of performing CTS 4.7.10.b.1, the main turbine bypass system functional test (e.g., simulated automatic actuation) are proposed to be relocated to the Bases. These proposed relocated details are not necessary to ensure the OPERABILITY of the Main Turbine Bypass System. The requirements of ITS 3.7.7 and SR 3.7.7.2 are adequate to ensure the Main Turbine Bypass System is maintained OPERABLE. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The CTS 4.7.10.b.2 details of the actual TURBINE BYPASS SYSTEM RESPONSE TIME are proposed to be relocated to the Technical Requirements Manual (TRM). Testing of the response time is provided by a specific Surveillance Requirement (SR 3.7.7.3) and is an integral part of the OPERABILITY of the Main Turbine Bypass System. As such, the requirements of ITS 3.7.7 and SR 3.7.7.3 are adequate to ensure the Main Turbine Bypass System response times are maintained within limits and the Main Turbine Bypass System is maintained OPERABLE. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference into the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

LD.1 The Frequency for performing CTS 4.7.10.b.1, the system functional test and CTS 4.7.10.b.2, the TURBINE BYPASS SYSTEM RESPONSE TIME test (proposed SR 3.7.7.2 and 3.7.7.3), has been extended from 18 months to 24 months. These SRs ensure that the Main Turbine Bypass System will function with the required response as assumed in the transient analysis such as the turbine generator load rejection and feedwater transients in order to mitigate the increase in reactor vessel pressure, which reduces the MCPR during the transient. The proposed change will allow these Surveillances to extend their Surveillance Frequencies from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed Specification 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,"

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be small. The main turbine bypass logic which is being tested is part of the Main Turbine Control System which is in continuous operation at power. Most malfunctions anticipated during power operations that would impact the Main Turbine Bypass System performance would also impact the operation of the entire Main Turbine Control System, which in most cases would be readily apparent to plant operators. In addition the weekly test of the turbine bypass valves (SR 3.7.7.1) will also detect problems since the test uses a fast open signal for the last 10% of valve travel.

Based on the above discussion, the impact of this change on system availability, if any, is minimal.

In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

"Specific"

L.1 In the event less than four main turbine bypass valves are capable of accepting steam flow, Action 2.b) of CTS 3.7.10 requires the Main Turbine Bypass System to be restored to an OPERABLE status within 12 hours following completion of Action 2.a) of CTS 3.7.10. This requirement has not been retained in proposed ITS 3.7.7. Analyses have been performed assuming the Main Turbine Bypass System is out of service (i.e., all five bypass valves are inoperable). These analyses confirmed that continued plant operation with the Main Turbine Bypass System out of service was acceptable with the application of a specific cycle-dependent MCPR value, as specified in the COLR, for the inoperable Main Turbine Bypass System. Thus, requiring the Main Turbine Bypass System to be restored to an OPERABLE status following the application of the specific MCPR penalty imposes an unnecessary restraint upon operation. Additionally, this change is consistent with ISTS 3.7.6 of NUREG-1434, Revision 1.

RELOCATED SPECIFICATIONS

None

REFUELING OPERATIONS

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

LCO 3.7.8 3.9.9 At least ~~23 feet~~ 21 ft 4 inches of water shall be maintained over the ~~top of active fuel in~~ irradiated fuel assemblies seated in the spent fuel storage pool racks. A.3

A.2 APPLICABILITY: ~~Whenever~~ During movement of irradiated fuel assemblies ~~are~~ in the spent fuel storage pool. During movement of new fuel assemblies in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool.

ACTION:

ACTION A With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area ~~after placing the fuel assemblies~~ and crane load ~~in a safe condition~~. The provisions of Specification 3.0.3 are not applicable. LA.2 LA.1

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

page 1 of 2

A.1

ITS 3.7.8

REFUELING OPERATIONS

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

21 ft 4 inches

LCO 3.7.8 3.9.9 At least ~~22 feet~~ of water shall be maintained over the ~~top of active fuel in~~ irradiated fuel assemblies seated in the spent fuel storage pool racks. A3

A.2 APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool. During movement of

During movement of new fuel assemblies in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool.

ACTION:

ACTION A With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area ~~after placing the fuel assemblies~~ and crane load ~~in a safe condition~~. The provisions of Specification 3.0.3 are not applicable. LA.2

LA.1

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Technical Specification (ISTS)).

A.2 CTS 3.9.9, which requires the spent fuel pool water level to be within limit, has an Applicability of "whenever irradiated fuel assemblies are in the spent fuel storage pool." However, the CTS 3.9.9 Action only requires suspension of fuel movement and crane operations with loads. (In addition, the relocation of crane operations with loads is specifically discussed in Discussion of Change LA.1 below). Thus, the spent fuel pool water level is not required to be maintained within the limit as long as fuel movement is suspended. With fuel movement suspended, fuel pool level can be outside the limits for an unlimited amount of time. The Applicability of ITS 3.7.8 is limited to circumstances when irradiated fuel assemblies are being moved in the spent fuel storage pool or when new fuel is being moved in the spent fuel storage pool with irradiated fuel assemblies in the spent fuel storage pool. This is acceptable since the purpose of ITS 3.7.8 is to ensure sufficient water is above the irradiated fuel assemblies to meet the assumptions of a fuel handling accident. With no fuel being handled, a fuel handling accident cannot occur. Therefore, since CTS 3.9.9 already allows continued operation with the spent fuel pool water level not within the limit (provided fuel handling is suspended), this change is considered administrative.

A.3 CTS 3.9.9 requires that 23 feet of water shall be maintained over the "top of active fuel" in irradiated fuel assemblies seated in the spent fuel pool storage racks. Proposed ITS 3.7.8 provides an equivalent requirement, that is stated in terms of the depth of water that shall be maintained over the "irradiated fuel assemblies" seated in the spent fuel pool storage racks. While the CTS requirement establishes the top of active fuel as the reference point for measuring spent fuel pool depth, the ITS requirement uses the top of the fuel bundle - which is located at the top of the fuel bundle bail handle. Maintaining 21 ft 4 inches of water over the "irradiated fuel assemblies" seated in the spent fuel pool storage racks is equivalent to maintaining 23 feet of water over the "top of active fuel" in irradiated fuel assemblies seated in the spent fuel pool storage racks. Consequently, the spent fuel pool water level LCO of proposed ITS 3.7.8 has been revised to require 21 ft 4 inches of water above the irradiated fuel assemblies seated in the spent fuel pool storage racks. Since the depth of water that must be maintained over the top of irradiated fuel has not changed, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 3.9.9 Action requires suspension of crane operations with loads when the spent fuel storage pool water level is not within the limit. The requirement is proposed to be relocated to the UFSAR since the movement of loads other than fuel assemblies is administratively controlled based on the heavy loads analysis. The bounding design basis fuel handling accident assumes an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies seated within the RPV. The movement of other loads over irradiated fuel assemblies is administratively controlled based on available analysis for the individual load. The load analysis methodology and crane operation which dictate the controls are described in UFSAR, Sections 9.1.2.1.2, 9.1.2.1.3, and 9.1.4. As such, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

LA.2 Details of the methods for performing the CTS 3.9.9 Action (after placing the fuel assemblies in a safe condition) are proposed to be relocated to the Bases. The allowance to place fuel assemblies in a safe condition prior to suspending fuel movement is not necessary for assuring, in the case of spent fuel water level not within limits, actions are taken to preclude a spent fuel handling accident from occurring. ITS 3.7.8 Required Action A.1 is adequate to preclude a spent fuel handling accident from occurring. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 3/4.7.4

3/4.7.4 SEALED SOURCE CONTAMINATION

R.1

LIMITING CONDITION FOR OPERATION

3.7.4 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

   a. The licensee, or

   b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.4.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

   a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

      1. With a half-life greater than 30 days, excluding Hydrogen 3, and

      2. In any form other than gas.

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R.1

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.4.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

PLANT SYSTEMS

3/4.7.4 SEALED SOURCE CONTAMINATION

R.1

LIMITING CONDITION FOR OPERATION

3.7.4 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

   a. The licensee, or

   b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.4.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

   a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

      1. With a half-life greater than 30 days, excluding Hydrogen 3, and

      2. In any form other than gas.

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R.1

SURVEILLANCE REQUIREMENTS (Continued)

b. ~~Stored sources not in use - Each sealed source and fission detector~~ shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.4.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.7.4, which provides requirements for sealed source contamination, does not identify a parameter which is an initial condition assumption for a DBA or transient, does not identify a significant abnormal degradation of the reactor coolant pressure boundary, does not provide any mitigation of a design basis event, and is not a structure, system, or component which operating experience or PRA has shown to be significant to public health and safety. Therefore, the requirements specified in CTS 3/4.7.4 did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the LaSalle 1 and 2 Technical Specifications and will be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

CTS 3/4.7.7

PLANT SYSTEMS

3/4.7.7 AREA TEMPERATURE MONITORING

R.1

LIMITING CONDITION FOR OPERATION

3.7.7 The temperature of each area of Unit 1 and Unit 2 shown in Table 3.7.7-1 shall be maintained within the limits indicated in Table 3.7.7-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.7-1:

a. For more than 8 hours, in lieu of any Licensee Event Report, prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.7 The temperature in each of the above required areas shown in Table 3.7.7-1 shall be determined to be within its limit at least once per 24 hours.

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R.1

TABLE 3.7.7-1

AREA TEMPERATURE MONITORING

AREA			TEMPERATURE LIMIT (°F)
A.	Unit 1 Area Temperature Monitoring		
	1.	Control Room	50-104
	2.	Auxiliary Electric Equipment Room	50-104
	3.	Diesel Generator Room	50-122
	4.	Switchgear Room	50-104
	5.	HPCS, LPCS, RHR & RCIC Rooms	50-150
	6.	Primary Containment	
		a. Drywell	50-150
		b. Beneath Reactor Pressure Vessel	50-185
B.	Unit 2 Area Temperature Monitoring Required For Unit 1		
	1.	Auxiliary Electric Equipment Room	50-104
	2.	Diesel Generator 2A Room	50-122
	3.	Division 1 and 2 Switchgear Rooms	50-104

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CTS 3/4.7.7

R.1

3/4.7.7 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.7 The temperature of each area of Unit 1 and Unit 2 shown in Table 3.7.7-1 shall be maintained within the limits indicated in Table 3.7.7-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.7-1:

a. For more than 8 hours, in lieu of any Licensee Event Report, prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.7 The temperature in each of the above required areas shown in Table 3.7.7-1 shall be determined to be within its limit at least once per 24 hours.

R.1

TABLE 3.7.7-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
A. Unit 2 Area Temperature Monitoring	
1. Control Room	50-104
2. Auxiliary Electric Equipment Room	50-104
3. Diesel Generator Room	50-122
4. Switchgear Room	50-104
5. HPCS, LPCS, RHR & RCIC Rooms	50-150
6. Primary Containment	
a. Drywell	50-150
b. Beneath Reactor Pressure Vessel	50-185
B. Unit 1 Area Temperature Monitoring Required For Unit 2	
1. Auxiliary Electric Equipment Room	50-104
2. Diesel Generator 1A Room	50-122
3. Division 1 and 2 Switchgear Rooms	50-104

DISCUSSION OF CHANGES
CTS: 3/4.7.7 - AREA TEMPERATURE MONITORING

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.7.7 provides requirements for area temperature monitoring. This Specification does not identify a parameter which is an initial condition assumption for a DBA or transient, does not identify a significant abnormal degradation of the reactor coolant pressure boundary, does not provide any mitigation of a design basis event, and is not a structure, system, or component which operating experience or PRA has shown to be significant to public health and safety. Therefore, the requirements specified in CTS 3/4.7.7 did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the LaSalle 1 and 2 Technical Specifications, and will be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

CTS 3/4.7.8

R.1

3/4.7.8 STRUCTURAL INTEGRITY OF CLASS 1 STRUCTURES

LIMITING CONDITION FOR OPERATION

3.7.8 The structural integrity of Class 1 structures shall be verified pursuant to the requirements of Specifications 4.7.8.1 and 4.7.8.2.

APPLICABILITY: At all times.

ACTION:

With the settlement of any Class 1 structure not verified to be within the allowable final settlement value as required, submit a Special Report in accordance with Specification 6.6.C:

a. By telephone within 24 hours,

b. Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

c. In writing within 14 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.

SURVEILLANCE REQUIREMENTS

4.7.8.1 The total settlement of each Class 1 structure and the differential settlement between Class 1 structures shall be determined to the nearest 0.01 foot by measurement and calculation:

a. At least once per 31 days:

1. During the first 6 months of unit operation,

2. Until observed settlement has stabilized,* and

3. Whenever previously stabilized* settlement exceeds 0.01 feet since the previous reading.

b. At least once per 6 months.

4.7.8.2 A Special Report shall be prepared and submitted to the Commission at least once per 6 months until settlement of Class 1 structures has stabilized. The report shall include settlement and differential settlement plots versus time and a comparison of allowable and actual settlement.

*$\leq$ 0.01 feet from previous reading.

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CTS 3/4.7.8

R.1

3/4.7.8 STRUCTURAL INTEGRITY OF CLASS 1 STRUCTURES

LIMITING CONDITION FOR OPERATION

3.7.8 The structural integrity of Class 1 structures shall be verified pursuant to the requirements of Specifications 4.7.8.1 and 4.7.8.2.

APPLICABILITY: At all times.

ACTION:

With the settlement of any Class 1 structure not verified to be within the allowable final settlement value as required, submit a Special Report in accordance with Specification 6.6.C:

a. By telephone within 24 hours,

b. Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

c. In writing within 14 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.

SURVEILLANCE REQUIREMENTS

4.7.8.1 The total settlement of each Class 1 structure and the differential settlement between Class 1 structures shall be determined to the nearest 0.01 foot by measurement and calculation:

a. At least once per 31 days:

1. Until observed settlement has stabilized,* and

2. Whenever previously stabilized* settlement exceeds 0.01 foot since the previous reading.

b. At least once per 6 months.

4.7.8.2 A Special Report shall be prepared and submitted to the Commission at least once per 6 months until settlement of Class 1 structures has stabilized. The report shall include settlement and differential settlement plots versus time and a comparison of allowable and actual settlement.

*$\leq$ 0.01 foot from previous reading.

DISCUSSION OF CHANGES
CTS: 3/4.7.8 - STRUCTURAL INTEGRITY OF CLASS 1 STRUCTURES

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.7.8 provides requirements for structural integrity requirements for Class I structures. This Specification does not identify a parameter which is an initial condition assumption for a DBA or transient, does not identify a significant abnormal degradation of the reactor coolant pressure boundary, does not provide any mitigation of a design basis event, and is not a structure, system, or component which operating experience or PRA has shown to be significant to public health and safety. Therefore, the requirements specified in CTS 3/4.7.8 did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the LaSalle 1 and 2 Technical Specifications, and will be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the LaSalle 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

CTS 3/4.7.9

PLANT SYSTEMS

3/4.7.9 SNUBBERS

LA.1

LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable, on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7.9-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7.9-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment 91.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and

LA.1

SURVEILLANCE REQUIREMENTS (Continued)

(3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.9f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspection

An inspection shall be performed of all hydralic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting, or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

At least once per 18 months during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested, in accordance with Figure 4.7-1. "C" is the

LA.1

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests (continued)

total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type may be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional testing sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

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LA.1

SURVEILLANCE REQUIREMENTS (Continued)

f. Functional Testing Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement

PLANT SYSTEMS

LA.1

SURVEILLANCE REQUIREMENTS (Continued)

h. Functional Testing of Repaired and Replaced Snubbers (Continued)

snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.5B.

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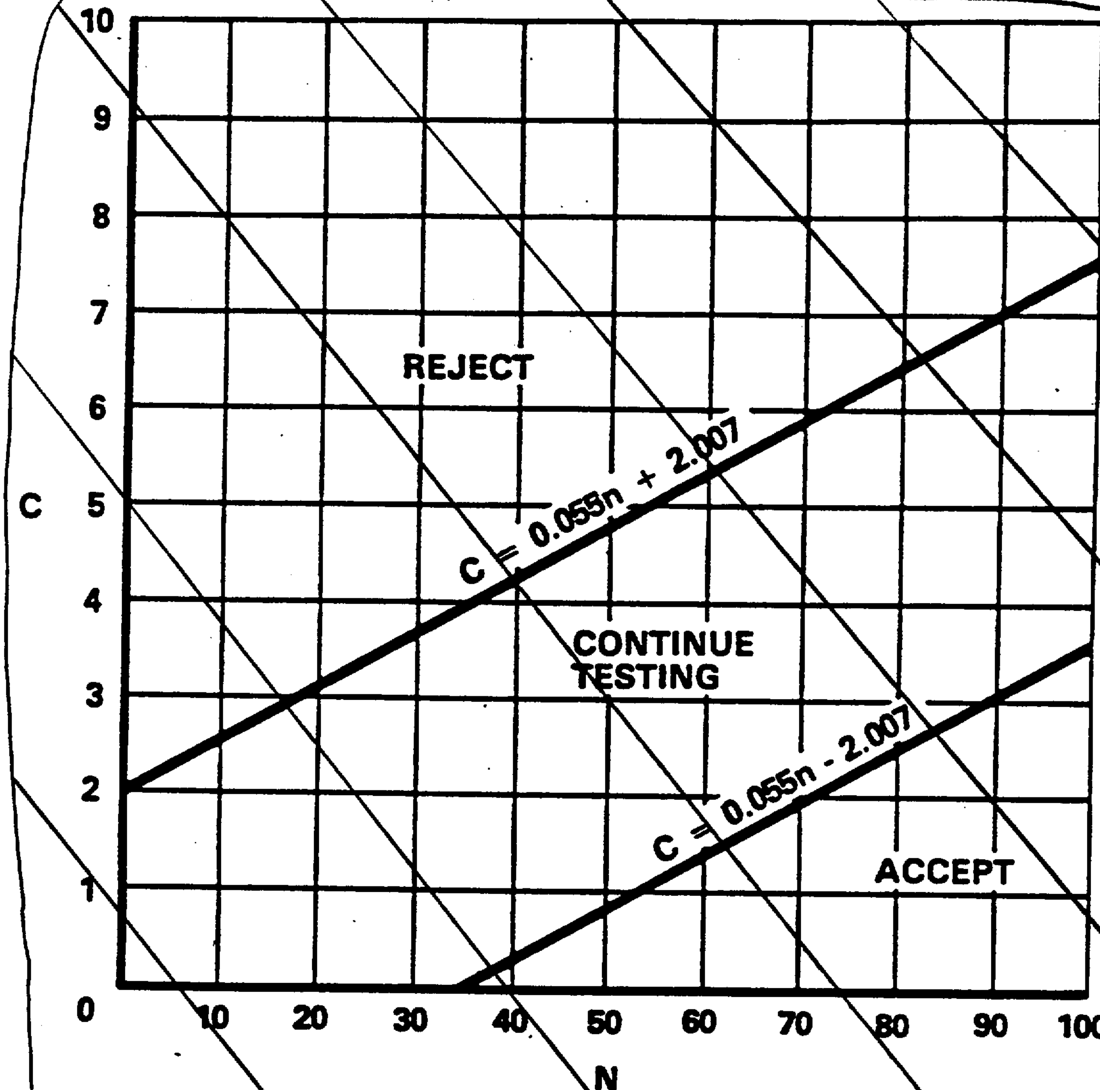

FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

LA.1

TABLE 4.7.9-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use the next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

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TABLE 4.7.9-1
SNUBBER VISUAL INSPECTION INTERVAL
(Continued)

~~Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.~~

~~Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.~~

~~Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.~~

~~Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.~~

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3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable, on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7.9-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7.9-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment 75

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and

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SURVEILLANCE REQUIREMENTS (Continued)

(3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.9f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspection

An inspection shall be performed of all hydralic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

At least once per 18 months during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested, in accordance with Figure 4.7-1. "C" is the

e. Functional Tests (Continued)

total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type may be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

SURVEILLANCE REQUIREMENTS (Continued)

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test

CTS 3/4.7.9

LA.1

SURVEILLANCE REQUIREMENTS (Continued)

h. Functional Testing of Repaired and Replaced Snubbers (Continued)

criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.5B.

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LA.1

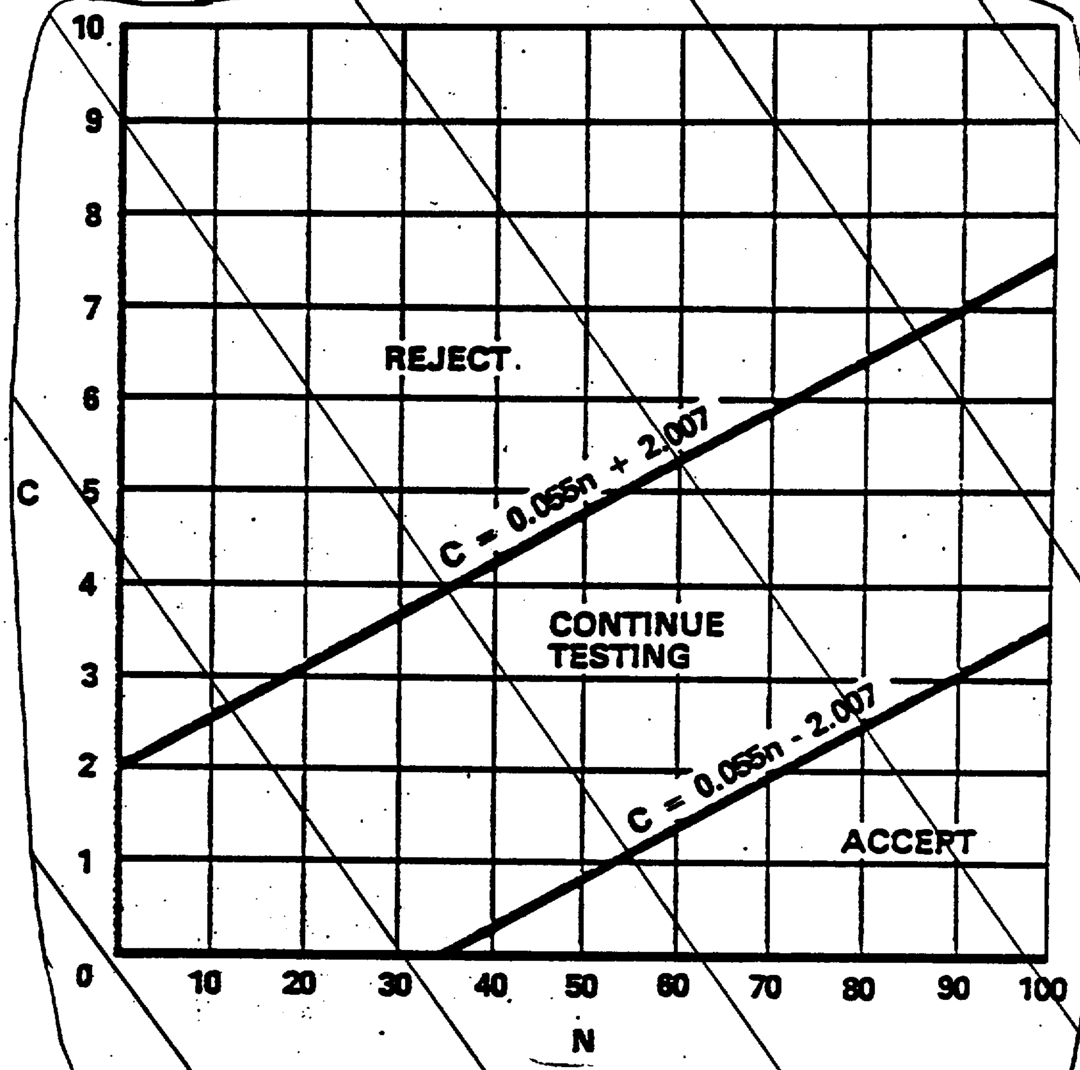

FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

CTS 3/4.7.9

LA, 1

TABLE 4.7.9-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use the next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

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TABLE 4.7.9-1
SNUBBER VISUAL INSPECTION INTERVAL
(Continued)

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

LA.1

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 The CTS 3/4.7.9 Snubber inspection and testing requirements will be part of the LaSalle 1 and 2 Snubber Program and are being relocated from the TS to the Technical Requirements Manual (TRM). The requirement to perform snubber inspections is specified in 10 CFR 50.55a and the requirement to perform snubber inspections and testing is specified in ASME Section XI. Therefore, both LaSalle 1 and 2 commitments and NRC Regulations or generic guidance will contain the necessary programmatic requirements for the Snubber Program without repeating them in the ITS. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference into the LaSalle 1 and 2 UFSAR at ITS Implementation. Snubber inspections and testing will continue to be performed in accordance with the CTS 3/4.7.9 requirements. Changes to the TRM will be controlled by 10 CFR 50.59. With the removal of operability requirements from the TS, snubber operability requirements will be determined in accordance with TS system operability requirements.

"Specific"

None

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS: SECTION 3.7 - PLANT SYSTEM BASES

The Bases of the current Technical Specifications for this section (pages B 3/4 7-1 through B 3/4 7-5 and B 3/4 11-1) have been completely replaced by revised Bases that reflect the format and applicable content of the LaSalle 1 and 2 ITS Section 3.7, consistent with the BWR ISTS, NUREG-1434, Rev. 1. The revised Bases are as shown in the LaSalle 1 and 2 ITS Bases.

Insert BWR/4 ISTS 3.7.1 *

1

<CTS>

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

<LCO 3.7.1.1> LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

<Appl 3.7.7.1> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One RHRSW pump inoperable.~~	~~A.1 Restore RHRSW pump to OPERABLE status.~~	~~30 days~~ 2
~~B. One RHRSW pump in each subsystem inoperable.~~	~~B.1 Restore one RHRSW pump to OPERABLE status.~~	~~7 days~~
<3.7.1.1 Act a.1 3.7.1.1 Act b> ~~C.~~ A 2 One RHRSW subsystem inoperable ~~for reasons other than Condition A~~.	~~C.~~ A.1 ---------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.~~8~~ 9 "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for ~~RHR shutdown cooling~~ subsystem made inoperable by RHRSW System. -------------------- Restore RHRSW subsystem to OPERABLE status.	3 4 7 days 5

(continued)

* This BWR/4 Specification was used because it best represented the LaSalle 1 and 2 design

Insert BWR/4 ISTS 3.7.1 [1]

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.1.1 Act a.2> <3.7.1.1 Act b> B [2] ~~D~~. Both RHRSW subsystems inoperable ~~for reasons other than Condition B~~.	B ~~D~~.1 ---------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.9 ~~8~~ for [RHR shutdown cooling] subsystems made inoperable by RHRSW System. [3] [4] ---------------------- Restore one RHRSW subsystem to OPERABLE status.	[8] hours [4]
<3.7.1.1 Act a.1> <3.7.1.1 Act a.2> C [2] ~~E~~. Required Action and associated Completion Time not met.	C ~~E~~.1 Be in MODE 3. AND C ~~E~~.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.7.1.1> SR 3.7.1.1	Verify each RHRSW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.7.1 - RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

1. A new Specification has been added, ITS 3.7.1, for the RHRSW System. This Specification is from the BWR/4 ISTS (NUREG-1433, ISTS 3.7.1), because the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the RHRSW System. Therefore, the BWR/4 ISTS is used and any deviations from the BWR/4 ISTS are discussed below.

2. The Actions of ISTS 3.7.1 have been revised to be consistent with the actions provided in the LaSalle 1 and 2 CTS. Subsequent Actions are renumbered, as required.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Typographical error corrected.

1

3.7 PLANT SYSTEMS

3.7.2 High Pressure Core Spray (HPCS) Service Water System (SWS)

LCO 3.7.2 The HPCS SWS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. HPCS SWS inoperable.	A.1 Declare HPCS System inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify water level of the [a standby service water] cooling tower basin is ≥ [7.25] ft.	24 hours
SR 3.7.2.2	-------------------NOTE------------------- Isolation of flow to individual components does not render [HPCS SWS] System inoperable. -- Verify each HPCS SWS manual, power operated, and automatic valve in the flow path [servicing safety related systems or components], that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.2.3	Verify the HPCS SWS actuates on an actual or simulated initiation signal.	[18] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.7.2 - HIGH PRESSURE CORE SPRAY (HPCS)
SERVICE WATER SYSTEM (SWS)

1. This Specification has been deleted since LaSalle 1 and 2 do not have a HPCS SWS.

Insert BWR/4 ISTS 3.7.3 * [1]

DGCW [3]

~~DG [1B] SSW~~ System
3.7.~~8~~ (2) [1] [3]

<CTS>

3.7 PLANT SYSTEMS

3.7.~~8~~ (2) [1] Diesel Generator ~~(DG) [1B] Standby Service~~ Cooling Water (~~SSW~~ DGCW) System

<LCO 3.7.1.2> LCO 3.7.~~8~~ (2) [1] Insert LCO ~~The DG [1B] SSW System shall be OPERABLE~~ [2]

<Appl 3.7.1.2> APPLICABILITY: MODES 1, 2, and 3. ~~When DG [1B] is required to be OPERABLE~~ [4]

<DOC A.2> ACTIONS

-------------------- NOTE --------------------
Separate Condition entry is allowed for each DGCW subsystem. [5]

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.1.2 Act> A. ~~DG [1B] SSW System~~ One or more DGCW subsystems inoperable.	~~-------------NOTE------------- LCO 3.0.4 is not applicable. -----------------------------~~	
	~~A.1 Align cooling water to DG [1B] from a Unit [1] plant service water (PSW) subsystem.~~	~~8 hours~~
	~~AND~~	
	~~A.2 Verify cooling water is aligned to DG [1B] from a Unit [1] PSW subsystem.~~	~~Once per 31 days~~
	~~AND~~	
	~~A.3 Restore DG [1B] SSW System to OPERABLE status.~~	~~60 days~~
~~B. Required Action and Associated Completion Time not met.~~	A.1 Declare ~~DG [1B]~~ supported component(s) inoperable.	Immediately

[6]

* This BWR/4 Specification was used because it best represented the LaSalle 1 and 2 design

<CTS>

<LCO 3.7.1.2>

Insert LCO

The following DGCW subsystems shall be OPERABLE:

a. Three unit DGCW subsystems; and

b. The opposite unit Division 2 DGCW subsystem.

Insert BWR/4 ISTS 3.7.3 (continued)

1

DGCW 3

~~DG [1B] SSW~~ System
3.7.~~3~~ 2 1

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
<4.7.1.2.a>	SR 3.7.~~3~~.1 (2) 1	Verify each ~~DG [1B] SSW System~~ required DGCW subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	2
<4.7.1.2.b>	SR 3.7.~~3~~.2 (2) 1	Verify ~~the DG [1B] SSW System~~ each required DGCW pump starts automatically ~~when DG [1B] starts and energizes the respective bus~~ on each required actual or simulated initiation signal.	~~[18]~~ 24 months	2, 3

2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434 REVISION 1
ITS: 3.7.2 - DIESEL GENERATOR COOLING WATER (DGCW) SYSTEM

1. A new Specification has been added, ITS 3.7.2, for the DGCW System. This Specification is from the BWR/4 ISTS (NUREG-1433, ISTS 3.7.3), because the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the DGCW System. Therefore, the BWR/4 ISTS is used and any deviations from the BWR/4 ISTS are discussed below.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. As a result of changes made to reflect the plant design (i.e., the DGCW System supports OPERABILITY of diesel generators and ECCS components), the Applicability has been revised to be consistent with the diesel generator Applicability of ITS 3.8.1, "AC Sources — Operating," and ITS 3.5.1, "Emergency Core Cooling Systems (ECCS) — Operating."

5. An ACTIONS Note has been added to ISTS 3.7.3 (ITS 3.7.2) to allow separate Condition entry for each inoperable DGCW subsystem consistent with the intent of the existing CTS 3.7.1.2 Action for one or more inoperable DGCW subsystems. The CTS 3.7.1.2 Action requires the associated diesel generator to be declared inoperable and the applicable Actions of CTS 3.8.1.1, "A.C. Sources - Operating," or CTS 3.8.1.2, "A.C. Sources - Shutdown," to be taken. This change is intended to ensure that each occurrence of an inoperable DGCW subsystem be assessed in accordance with the applicable Conditions and Required Actions of LCO 3.8.1 for its impact on the DG System's capability to function as an AC power source. This is consistent with Current Licensing Basis.

6. The ISTS 3.7.3 (ITS 3.7.2) Required Action Note and Required Actions A.1, A.2, and A.3 and their associated Completion Times have been deleted since they are not applicable to LaSalle 1 and 2. Required Action A.1 requires an alternative cooling water supply to be aligned to a DG with its normal cooling water supply inoperable. Required Actions A.2 and A.3 require periodic verification of the alternative cooling water supply alignment and restoration of the normal cooling water supply within 60 days. The Required Action Note provides an exception to LCO 3.0.4 such that MODE changes are allowed with the alternate cooling water supply aligned to a DG. The LaSalle 1 and 2 design does not afford the capability of aligning a qualified alternative cooling water source to the DGs in the event one or more DGCW subsystems are inoperable. For LaSalle 1 and 2, when one or more DGCW subsystems are inoperable, CTS 3.7.1.2 requires the associated DG to be declared inoperable and the applicable

6. (continued)

Actions of Specifications 3.8.1, "A.C. Sources – Operating," to be taken. Thus, since the current design and Technical Specification requirements do not provide for an alternative cooling water source to a DGCW subsystem, the ISTS 3.7.3 (ITS 3.7.2) requirements relative to the alternative cooling water source have been deleted. In addition, ISTS 3.7.3 (ITS 3.7.2) Condition B has been deleted and Required Action B.1 and the associated Completion Time have been moved and renumbered as A.1 in order to provide appropriate direction within the ITS format for declaring supported equipment inoperable when one or more DGCW subsystems are inoperable consistent with the existing requirements as modified by Discussion of Change M.1 for ITS 3.7.2.

<CTS>

3.7 PLANT SYSTEMS

1 3.7.1 3 ~~[Standby Service Water (SSW)] System and [Ultimate Heat Sink (UHS)]~~ 1 2

<LCO 3.7.1.3> LCO 3.7.1 3 ~~Division 1 and 2 [SSW] subsystems and [UHS]~~ The Core Standby Cooling System (CSCS) pond shall be OPERABLE.

<Appl 3.7.1.3> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One or more cooling towers with one cooling tower fan inoperable.~~	~~A.1 Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days~~
A. CSCS pond inoperable due to sediment deposition or bottom elevation not within limit.	A.1 Restore CSCS pond to OPERABLE Status.	90 days

2

(continued)

<3.7.1.3 Act> 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One [SSW] subsystem inoperable [for reasons other than Condition A].	B.1	-----NOTES----- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for diesel generator made inoperable by [SSW]. 2. Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for [RHR shutdown cooling] made inoperable by [SSW]. ----- Restore [SSW] subsystem to OPERABLE status.	72 hours

2

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
2 B C. Required Action and associated Completion Time of Condition A ~~or B~~ not met. 2 <3.7.1.3 Act a>	C.1 Be in MODE 3.	12 hours
	AND B 2	
	C.2 Be in MODE 4.	36 hours
2 ~~OR Both [SSW] subsystems inoperable [for reasons other than Condition A].~~		
3 OR [UHS] CSCS pond inoperable for reasons other than Condition A.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
~~SR 3.7.1.1~~	~~Verify the water level of each [UHS] cooling tower basin is ≥ [7.25] ft.~~	~~24 hours~~	2
~~SR 3.7.1.2~~	~~Verify the water level [in each SSW pump well of the intake structure] is ≥ [ ] ft.~~	~~24 hours~~	2
<DOC M.1> SR 3.7.1.3 3.1	Verify the average water temperature of [UHS] CSCS pond is ≤ [ ] 97°F.	24 hours	3
<4.7.1.3.a> SR 3.7.3.2	Verify sediment level is ≤ 1.5 ft in the intake flume and the CSCS pond.	24 months	2
<4.7.1.3.b> SR 3.7.3.3	Verify CSCS pond bottom elevation is ≤ 686.5 ft.	24 months	

(continued)

③ 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.1.4	Operate each [SSW] cooling tower fan for ≥ [15] minutes.	31 days
SR 3.7.1.5	-----NOTE----- Isolation of flow to individual components does not render [SSW] System inoperable. ----- Verify each [SSW] subsystem manual, power operated, and automatic valve in the flow path servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.1.6	Verify each [SSW] subsystem actuates on an actual or simulated initiation signal.	[18] months

2
2
2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.7.3 - ULTIMATE HEAT SINK (UHS)

1. NUREG-1434, Rev. 1, ISTS 3.7.1 and ISTS 3.7.2 have been divided and revised into three different Specifications to reflect the LaSalle design. The "Service Water System" at LaSalle consists of two completely separate systems - the Residual Heat Removal Service Water (RHRSW) System and the Diesel Generator Cooling Water (DGCW) System - both of which draw cooling water from the Ultimate Heat Sink (UHS). Therefore, proposed ITS 3.7.1 (proposed using ISTS 3.7.1 from NUREG-1433, Rev.1 (BWR/4 ISTS)) now represents the requirements for the RHRSW System; proposed ITS 3.7.2 (proposed using ISTS 3.7.3 from the BWR/4 ISTS) now represents the requirements for the DGCW System; and a new ITS 3.7.3, "Ultimate Heat Sink (UHS)," is proposed (using ISTS 3.7.1 from NUREG-1434, Rev. 1) to address the requirements for the cooling water source for the systems covered by proposed ITS 3.7.1 and ITS 3.7.2. Subsequent Specifications and requirements have been renumbered to reflect the addition of the new Specification.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect plant specific nomenclature, number, references, system description, analysis description, or licensing basis description.

3. The brackets have been removed and the proper plant specific information/value has been provided.

<CTS>

3.7 PLANT SYSTEMS Area Filtration (CRAF) 2

3.7.5 Control Room ~~Fresh Air (CRFA)~~ System 4 1

<LCO 3.7.2> LCO 3.7.5 Two ~~CRFA~~ CRAF 2 subsystems shall be OPERABLE.

<Appl 3.7.2> APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the 2 ~~primary or~~ secondary containment~~s~~, 2
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.2 Act a>	A. One ~~CRFA~~ CRAF 2 subsystem inoperable.	A.1 Restore ~~CRFA~~ CRAF 2 subsystem to OPERABLE status.	7 days
<3.7.2 Act a.1>	B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.2 Act a.2> C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the 2 primary or secondary containments, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------	
	C.1 ~~-------NOTE------- Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable. ------------------~~ 3	
	Place OPERABLE CRFA (CRAF 2) subsystem in isolation (pressurization 2) mode.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies in the primary and secondary containments. 2	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
<Doc A.3> D. Two CRFA (CRAF 2) subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.2 Act b 3.7.2 Act c> E. Two CRAF 2 ~~CRFA~~ subsystems inoperable during movement of irradiated fuel assemblies in the 2 ~~[primary or]~~ secondary containment~~s~~, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the ~~[primary and]~~ secondary containment~~s~~. 2	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.7.2.a.1> SR 3.7.~~3~~4.1 1	Operate each CRAF ~~CRFA~~ subsystem for ~~[~~≥ 10 continuous hours with the heaters operating. ~~or (for systems without heaters) ≥ 15 minutes]~~. 2	31 days
<4.7.2.b> SR 3.7.~~3.2~~4.3 4 1	Perform required CRAF ~~CRFA~~ filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. 2	In accordance with the ~~[~~VFTP~~]~~ 2
<4.7.2.a.2> SR 3.7.4.2	Manually initiate flow through the CRAF recirculation filters for ≥ 10 hours.	31 days 4

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.7.2.d.2>	SR 3.7.3.5 4 1 4 Verify each CRAF 2 [CRFA] subsystem actuates on an actual or simulated initiation signal.	24 2 [18] months
<4.7.2.d.2> 2	SR 3.7.3.6 4 5 1 4 Verify each 2 CRAF [CRFA] subsystem can maintain a positive pressure of ≥ 2 0.125 [ ] inches water gauge relative to adjacent areas [buildings] during the pressurization [isolation] mode of operation at a flow rate of ≤ 4000 [ ] cfm.	24 2 [18] months [on a STAGGERED TEST BASIS] 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.7.4 - CONTROL ROOM AREA FILTRATION (CRAF) SYSTEM

1. The Specification has been renumbered to accommodate additional plant specific changes to ISTS Section 3.7.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. The bracketed requirement has been deleted because is it not applicable to LaSalle 1 and 2.

4. An additional Surveillance Requirement, to operate the CRAF subsystems with flow through the recirculation filters once per 31 days, has been added consistent with the current licensing basis. Subsequent, SRs have been renumbered.

<CTS>

3.7 PLANT SYSTEMS

Area Ventilation

3.7.4 {Control Room Air Conditioning (AC)} System 2

1 5

<DOC M.1> LCO 3.7.4 Two {control room AC} subsystems shall be OPERABLE.

area ventilation 2

<DOC M.1> APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the [primary or] secondary containment}, 2
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.1> 2 A. One {control room AC} subsystem inoperable.	area ventilation 2 A.1 Restore {control room AC} subsystem to OPERABLE status.	30 days
<DOC M.1> B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

⟨CTS⟩

ACTIONS (continued) [2]

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨DOC M.1⟩ C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the [2] ~~[primary or]~~ secondary containment~~]~~, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE ~~[control room AC]~~ area ventilation [2] subsystem in operation.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies in the ~~[primary and]~~ secondary containment~~]~~. [2]	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
⟨DOC M.1⟩ D. Two ~~[control room AC]~~ area ventilation [2] subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two [control room AC] subsystems inoperable during movement of irradiated fuel assemblies in the [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable.	
	E.1 Suspend movement of irradiated fuel assemblies in the [primary and secondary containment].	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

Area ventilation 2

<DOC M.1>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Verify each [control room AC] subsystem has the capability to remove the assumed heat load.	[18] months

INSERT 3

<CTS>

INSERT

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1 <DOC M.1>	Monitor control room and auxiliary electric equipment room temperatures.	12 hours
SR 3.7.5.2 <DOC M.1>	Verify correct breaker alignment and indicated power are available to the control room area ventilation AC subsystems.	7 days

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.7.5 - CONTROL ROOM AREA VENTILATION
AIR CONDITIONING (AC) SYSTEM

1. This Specification has been renumbered to accommodate other plant specific changes to ISTS Section 3.7.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. ISTS SR 3.7.4.1, which requires verification that each control room AC subsystem has the capability to remove the assumed heat load, is not adopted. The LaSalle 1 and 2 control room and AEER AC subsystems have air cooled condensers and refrigerant compressors. While an appropriate testing methodology has been developed for systems with water cooled chillers, the Nuclear HVAC Utilities Group (NHUG) has not yet developed a capacity verification test methodology for systems with air cooled condensers. Therefore, alternate testing is proposed similar to testing previously approved for ComEd's Zion Nuclear Power Station.

<CTS>

3.7 PLANT SYSTEMS

3.7.~~5~~ 6 Main Condenser Offgas [1]

<LCO 3.11.2.2> LCO 3.7.~~5~~ 6 The gross gamma activity rate of the noble gases measured ~~at the offgas recombiner effluent~~ Prior to the holdup line shall be ≤ ~~380~~ 340,000 μCi/second ~~after decay of 30 minutes~~. [2]

<Appl 3.11.2.2> APPLICABILITY: MODE 1,
MODES 2 and 3 with any ~~main steam line not isolated and~~ steam jet air ejector (SJAE) in operation. [2]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.11.2.2 Act> A. Gross gamma activity rate of the noble gases not within limit.	A.1 Restore gross gamma activity rate of the noble gases to within limit.	72 hours
<3.11.2.2 Act> B. Required Action and associated Completion Time not met.	~~B.1 Isolate all main steam lines.~~	~~12 hours~~ [2]
	~~OR~~	
	B.2 Isolate SJAE.	12 hours
	OR	
	B.3.1 Be in MODE 3.	12 hours
	AND	
	B.3.2 Be in MODE 4.	36 hours

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.11.2.2.2> SR 3.7.5.1 1 6	-------------------NOTE------------------- Not required to be performed until 31 days after any main steam line not isolated and SJAE in operation. --	2
2	Verify the gross gamma activity rate of the noble gases is ≤ [380] mCi/second after decay of 30 minutes. 340,000 μ	31 days AND Once within 4 hours after a ≥ 50% increase in the nominal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.7.6 - MAIN CONDENSER OFFGAS

1. The Specification has been renumbered to accommodate additional plant specific changes to ISTS Section 3.7.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3.7 PLANT SYSTEMS

3.7.6 7 Main Turbine Bypass System 1

<LCO 3.7.10>

<Doc A.2>

<3.7.10 Act 1.a)2)
3.7.10 Act 2.a)>

LCO 3.7.6 7 The Main Turbine Bypass System shall be OPERABLE.

~~[OR~~

~~LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the [COLR], are made applicable.]~~ 2

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

TSTF-319 changes not adopted 4

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
<3.7.10 Act 1.a) 3.7.10 Act 2.a)>	A. [Requirements of the LCO not met ~~or Main Turbine Bypass System inoperable.]~~	A.1 [Satisfy the requirements of the LCO ~~or restore Main Turbine Bypass System to OPERABLE status]~~	2 hours	2
<3.7.10 Act 1.b) 3.7.10 Act 2.c)>	B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours	

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
<4.7.10> 1	SR 3.7.6 7 .1 Verify one complete cycle of each main turbine bypass valve.	31 7 days	3

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
<4.7.10.b.1>	SR 3.7.~~6~~ 7 .2 1	Perform a system functional test.	~~18~~ 24 months	2
<4.7.10.b.2>	SR 3.7.~~6~~ 7 .3 1	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	~~18~~ 24 months	2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.7.7 - MAIN TURBINE BYPASS SYSTEM

1. The Specification has been renumbered to accommodate additional plant specific changes to ISTS Section 3.7.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Plant specific design and configuration requires that these valves be tested more often than the monthly ISTS requirement to ensure the reliability of these valves. The CTS 4.7.10.a frequency of weekly is retained.

4. TSTF-319 revised the Main Turbine Bypass System LCO (ISTS LCO 3.7.6) to require adjusting APLHGR limits, in addition to the ISTS LCO 3.7.6 requirement to adjust MCPR limits, when the Main Turbine Bypass System is inoperable. The plant-specific turbine bypass out-of-service analysis does not require adjustment of APLHGR or LHGR limits when the Main Turbine Bypass System is inoperable. Therefore, the change from TSTF-319 is not adopted.

<CTS>

3.7 PLANT SYSTEMS

3.7.~~8~~ 8 1 Fuel Pool Water Level
Spent Storage 2

<LCO 3.9.9>

LCO 3.7.~~8~~ 2 The Spent fuel Storage pool water level shall be ≥ ~~23 ft~~ 21 ft 4 inches 3 over the top of irradiated fuel assemblies seated in the spent fuel storage pool ~~and upper containment fuel storage pool~~ 2 racks.

<Appl 3.9.9>

APPLICABILITY: During movement spent 2 of irradiated fuel assemblies in the ~~associated~~ 2 fuel storage pool~~s~~ 5,
During movement of new fuel assemblies in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool. 4

ACTIONS

<3.9.9 Act>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent 2 fuel Storage 2 pool water level not within limit.	A.1 ---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of ~~irradiated~~ 4 fuel assemblies in the ~~associated~~ spent 2 fuel storage pool~~s~~ 2.	Immediately

SURVEILLANCE REQUIREMENTS

<4.9.9>

SURVEILLANCE	FREQUENCY
SR 3.7.~~8~~ 8 1 .1 Verify the Spent 2 fuel Storage 2 pool water level is ≥ ~~23 ft~~ 21 ft 4 inches 3 over the top of irradiated fuel assemblies seated in the spent fuel 2 storage pool 2 racks.	7 days

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.7.8 - SPENT FUEL STORAGE POOL WATER LEVEL

1. The Specification has been renumbered to accommodate additional plant specific changes to ISTS Section 3.7.

2. The proper LaSalle Units 1 and 2 plant specific nomenclature/value has been provided.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The Applicability has been changed to be consistent with current licensing bases, as it relates to fuel handling. In addition, it is consistent with the Applicability of ISTS 3.9.7, which specifies a water level requirement when moving new fuel over irradiated fuel. Also, the word "irradiated" has been deleted from Required Action A.1. This change was necessary because the proposed Applicability includes movement of both irradiated and new fuel assemblies and suspension of movement of both types of fuel assemblies is required to put the plant in a condition that is outside the Applicability.

Insert BWR/4 ISTS B 3.7.1 *

1

B 3.7 PLANT SYSTEMS

B 3.7.1 Residual Heat Removal Service Water (RHRSW) System

BASES

BACKGROUND

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System.

2 INSERT B 3.7.1 BKGD - A

The RHRSW System consists of two independent and redundant subsystems. Each subsystem is made up of a header, two [4000] gpm pumps, a suction source, valves, piping, heat exchanger, and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity with one pump operating to maintain safe shutdown conditions. The two subsystems are separated from each other ~~by normally closed motor operated cross tie valves~~, so that failure of one subsystem will not affect the OPERABILITY of the other subsystem. The RHRSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RHRSW System is described in the FSAR, Section [9.2.7], Reference 1.

(together capable of providing a nominal flow of 7400gpm) 2

2 both

s 2

2

u 2

1 3

~~Cooling water is pumped by the RHRSW pumps from the [Altamaha River] through the tube side of the RHR heat exchangers, and discharges to the [circulating water flume]. A minimum flow line from the pump discharge to the intake structure prevents the pump from overheating when pumping against a closed discharge valve.~~

2 INSERT B 3.7.1 BKGD-B

2

offsite power 2

The system is initiated manually from the control room. If operating during a loss of coolant accident (LOCA), the system is automatically tripped to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time [10 minutes] after the LOCA, ~~or manually started any time the LOCA signal is manually overridden or clears~~.

INSERT B 3.7.1 BKGD - C

2 load shed

2

(continued)

*This BWR/4 Bases was used to match the BWR/4 Specification inserted in the LCO Section.

Insert B 3.7.1 BKGD-A

The RHRSW System also provides cooling water to the RHR pump seal coolers which are required for RHR pump operation during the shutdown cooling mode in MODE 3.

Insert B 3.7.1 BKGD-B

The RHRSW and the Diesel Generator Cooling Water subsystems are subsystems to the Core Standby Cooling System (CSCS) – Equipment Cooling Water System (ECWS). The CSCS – ECWS consists of three independent piping subsystems corresponding to essential electrical power supply Divisions 1, 2, and 3. The CSCS – ECWS subsystems take suction from the service water tunnel located in the Lake Screen House. The RHRSW subsystems are manually initiated. Cooling water is then pumped from the service water tunnel by the RHRSW pumps to the supported system and components (RHR heat exchangers and RHR pump seal coolers). After removing heat from its supported systems and components, the water from the RHRSW subsystem is discharged to the CSCS Pond (i.e., the Ultimate Heat Sink) through a discharge line that is common to the corresponding divisional discharge from the other unit. The discharge line terminates in the discharge structure at an elevation above the normal CSCS Pond level.

Insert B 3.7.1 BKGD-C

In addition, the Division 2 RHRSW subsystem may be initiated manually from the remote shutdown panel in the auxiliary electric equipment room.

Insert BWR/4 ISTS B 3.7.1

(continued)

1

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RHRSW System to support long term cooling of the reactor or primary 3 ~~containment is discussed in the~~ FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

2 U

The safety analyses for long term cooling were performed for various combinations of RHR System failures. The worst case single failure that would affect the performance of the RHRSW System is any failure that would disable one subsystem of the RHRSW System. As discussed in the FSAR, U 2 Section [6.2.1.4.3] (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystem and the 2 associated RHR System is assumed to occur [10] minutes after a DBA. The RHRSW flow assumed in the analyses is [4000] gpm ~~per pump~~ with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are [806.41]°F and [56.50] psig, respectively, well below the design temperature of [340]°F and maximum ~~allowable~~ pressure of [62] psig.

3 6.2.2.3.1

3 7400

3 200

design

30.6 3 275 3

45 3

The RHRSW System satisfies Criterion 3 of ~~the NRC Policy Statement.~~

2 10CFR 50.36(c)(2)(ii).

LCO

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

An RHRSW subsystem is considered OPERABLE when:

a. Two pumps are OPERABLE; and

(continued)

Insert BWR/4 ISTS B 3.7.1

(continued)

1

BASES

CSCS service water tunnel 2

LCO (continued)

b. An OPERABLE flow path is capable of taking suction from the ~~intake structure~~ and transferring the water to the RHR heat exchangers at the assumed flow rate. ~~Additionally, the RHRSW cross tie valves (which allow the two RHRSW loops to be connected) must be closed so that failure of one subsystem will not affect the OPERABILITY of the other subsystems.~~

associated 2

2

An adequate suction source is not addressed in this LCO since the minimum net positive suction head ~~([59] ft mean sea level in the pump well) is bounded by the plant service water pump requirements~~ (LCO 3.7.8, "~~Plant Service Water (PSW) System and~~ Ultimate Heat Sink (UHS)"). 3

and the Maximum suction source temperature are covered by the requirements specified in

2 3

APPLICABILITY

In MODES 1, 2, and 3, the RHRSW System is required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling (LCO 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray") and decay heat removal (LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown"). The Applicability is therefore consistent with the requirements of these systems.

5 9

In MODES 4 and 5, the OPERABILITY requirements of the RHRSW System are determined by the systems it supports. 4

INSERT B 3.7.1 App

ACTIONS

5

~~A.1~~

~~With one RHRSW pump inoperable, the inoperable pump must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW pumps are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced RHRSW capability. The 30 day Completion Time is based on the remaining RHRSW heat removal capability, including enhanced reliability afforded by manual cross connect capability, and the low probability of a DBA with concurrent worst case single failure.~~

(continued)

Insert B 3.7.1-APP

and therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, the LCOs of the RHR Shutdown Cooling System (LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown," LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level," and LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level"), which require portions of the RHRSW System to be OPERABLE, will govern RHRSW System operation in MODES 4 and 5

Insert BWR/4 ISTS B 3.7.1 1
(continued)

BASES

ACTIONS (continued)

~~B.1~~

~~With one RHRSW pump inoperable in each subsystem, if no additional failures occur in the RHRSW System, and the two OPERABLE pumps are aligned by opening the normally closed cross tie valves, then the remaining OPERABLE pumps and flow paths provide adequate heat removal capacity following a design basis LOCA. However, capability for this alignment is not assumed in long term containment response analysis and an additional single failure in the RHRSW System could reduce the system capacity below that assumed in the safety analysis. Therefore, continued operation is permitted only for a limited time. One inoperable pump is required to be restored to OPERABLE status within 7 days. The 7 day Completion Time for restoring one inoperable RHRSW pump to OPERABLE status is based on engineering judgment, considering the level of redundancy provided.~~ 5

A ~~C~~.1

Required Action A ~~C~~.1 is intended to handle the inoperability of one RHRSW subsystem ~~for reasons other than Condition A~~. The Completion Time of 7 days is allowed to restore the RHRSW subsystem to OPERABLE status. With the unit in this condition, the remaining OPERABLE RHRSW subsystem is adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE RHRSW subsystem could result in loss of RHRSW function. The Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE subsystem and the low probability of an event occurring requiring RHRSW during this period. 5

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4 9 ~~.8,~~ be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable ~~[~~RHR shutdown cooling~~]~~. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. 5 3

B ~~D~~.1

With both RHRSW subsystems inoperable ~~for reasons other than Condition B~~ (e.g., both subsystems with inoperable pump(s) or flow 5 2

(continued)

Insert BWR/4 ISTS B 3.7.1 1

(continued)

BASES

ACTIONS B.1 (continued) 5

paths, or one subsystem with an inoperable pump and one subsystem with an inoperable flow path), the RHRSW System is not capable of performing its intended function. At least one subsystem must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time for restoring one RHRSW subsystem to OPERABLE status, is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.9 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. 2 3

C.1 and C.2 5

If any Required Action and associated Completion Time of Condition A or B are not met

~~If the RHRSW subsystems cannot be not restored to OPERABLE status within the associated Completion Times~~, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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. 16

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual, power operated, and automatic valve in each RHRSW subsystem flow path provides assurance that the proper flow paths will exist for RHRSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be realigned to its accident position. This is acceptable because the RHRSW System is a manually initiated system.

(continued)

Insert BWR/4 ISTS B.3.7.1 1
(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

REFERENCES

2 U

1. FSAR, Section [9.2.7]. 1 3
2. FSAR, Chapter [6]. 3
3. FSAR, Chapter [15]. 3
4. FSAR, Section ~~[6.2.1.4.3]~~. 6.2.2.3.1 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.7.1 - RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

1. A new Specification has been added, ITS 3.7.1 for the RHRSW System. This system is similar to, but not identical to, the RHRSW discussed in ISTS 3.7.1 of NUREG-1433, Revision 1. Thus, the Bases for proposed ITS 3.7.1 are based on ISTS 3.7.1 of NUREG-1433, Revision 1. The deviations from the BWR/4 ISTS are discussed below.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis design.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The Applicability Section of the Bases has been revised to add clarification regarding OPERABILITY requirements regarding the RHRSW System during MODES 4 and 5, since the proposed TS does not have an LCO for the RHRSW System in these MODES.

5. Changes have been made to reflect changes made to the Specification.

6. Editorial change made to enhance clarity or to be consistent with similar statements in other places in the Bases.

1

B 3.7 PLANT SYSTEMS

B 3.7.2 High Pressure Core Spray (HPCS) Service Water System (SWS)

BASES

BACKGROUND	The HPCS SWS is designed to provide cooling water for the removal of heat from components of the Division 3 HPCS System. The HPCS SWS consists of the Ultimate Heat Sink (UHS) Basin A, one cooling water header (subsystem C of the Standby Service Water (SSW) System), and the associated pumps, piping, and valves. The UHS is also considered part of the SSW System (LCO 3.7.1, "[Standby Service Water (SSW)] System and [Ultimate Heat Sink (UHS)]"). Cooling water is pumped from a UHS water source by the HPCS service water pump to the essential components through the HPCS service water supply header. After removing heat from the components, the water is discharged to the cooling towers, where the heat is rejected through direct contact with ambient air. The HPCS SWS specifically supplies cooling water to the Division 3 HPCS diesel generator jacket water coolers and HPCS pump room cooler. The HPCS SWS pump is sized such that it will provide adequate cooling water to the equipment required for safe shutdown. Following a Design Basis Accident or transient, the HPCS SWS will operate automatically and without operator action as described in the FSAR, Section [9.2.1] (Ref. 1).
APPLICABLE SAFETY ANALYSES	The ability of the HPCS SWS to provide adequate cooling to the HPCS System is an implicit assumption for safety analyses evaluated in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). The HPCS SWS satisfies Criterion 3 of the NRC Policy Statement.
LCO	The HPCS SWS is required to be OPERABLE to ensure that the HPCS System will operate as required. An OPERABLE HPCS SWS

(continued)

1

BASES

LCO (continued)

consists of an OPERABLE UHS; an OPERABLE pump; and an OPERABLE UHS flow path, capable of taking suction from the associated SSW source and transferring the water to the appropriate unit equipment.

The OPERABILITY of the UHS is specified in LCO 3.7.1. However, the OPERABILITY of the basin cooling tower fans does not affect the OPERABILITY of the HPCS SWS, due to the limited heat removal during its operation.

APPLICABILITY

In MODES 1, 2, and 3, the HPCS SWS is required to be OPERABLE to support OPERABILITY of the HPCS System since it is required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the HPCS SWS and the UHS are determined by the HPCS System.

ACTIONS

A.1

When the HPCS SWS is inoperable, the capability of the HPCS System to perform its intended function cannot be ensured. Therefore, if the HPCS SWS is inoperable, the HPCS System must be declared inoperable immediately and Condition C of LCO 3.5.1, "ECCS—Operating," entered.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR ensures that adequate cooling can be maintained. With the UHS water source below the minimum level, the HPCS SWS must be declared inoperable. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.2.2

Verifying the correct alignment for each manual, power operated, and automatic valve in the HPCS service water flow path provides assurance that the proper flow paths will exist for HPCS service water operation. This SR does not

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.2 (continued)

apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in correct position prior to locking, sealing, or securing.

A valve is also allowed to be in the nonaccident position and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR is modified by a Note indicating that isolation of the [HPCS SWS] System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the [HPCS SWS] System. As such, when all [HPCS SWS] pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the [HPCS SWS] System is still OPERABLE.

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

SR 3.7.2.3

This SR verifies that the automatic valves of the HPCS SWS will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This is demonstrated by use of an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.5.1.5 overlaps this SR to provide complete testing of the safety function.

Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

(continued)

1

BASES (continued)

REFERENCES

1. FSAR, Section [9.2.1].
2. FSAR, Chapter [6].
3. FSAR, Chapter [15].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.7.2 - HIGH PRESSURE CORE SPRAY (HPCS)
SERVICE WATER SYSTEM (SWS)

1. The Bases has been deleted since the Specification has been deleted.

Insert BWR/4 ISTS B3.7.3 * [1]

All changes are [2] unless otherwise indicated

DGCW [4]

~~DG [1B] SSW~~ System

B 3.7.~~3~~ 2 [1]

B 3.7 PLANT SYSTEMS

B 3.7.~~3~~ 2 [1] Diesel Generator ~~(DG) [1B] Standby Service~~ Cooling Water (~~SSW~~ DGCW) System [4]

BASES

BACKGROUND

The ~~DG [1B] SSW~~ DGCW [4] System is designed to provide cooling water for the removal of heat from the ~~DG [1B]. DG [1B] is the only component served by the DG [1B] SSW System.~~ standby diesel generators, Low Pressure Core Spray (LPCS) pump motor cooling coils, and Emergency Core Cooling System (ECCS) cubicle area cooling coils that support equipment required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient.

Insert B 3.7.2 BKGD

~~The DG [1B] SSW pump autostarts upon receipt of a diesel generator (DG) start signal when power is available to the pump's electrical bus. Cooling water is pumped from the [Altamaha River] by the DG [1B] SSW pump to the essential DG components through the SSW supply header. After removing heat from the components, the water is discharged to the unit service water (PSW) discharge header. The capability exists to manually cross connect the PSW System to supply cooling to the DG [1B] during times when the SSW pump is inoperable.~~ A complete description of the ~~DG [1B] SSW~~ DGCW [4] System is presented in the U FSAR, Section ~~[9.5.5]~~ 9.2.1 [4] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ability of the ~~DG [1B] SSW~~ DGCW [4] System to provide adequate cooling to the DGs ~~[1B]~~, the LPCS pump motor cooling coils, and ECCS cubicle area cooling coils is an implicit assumption for the ~~safety analyses presented~~ in ~~the~~ U FSAR, Chapters ~~[~~6~~]~~ and ~~[~~15~~]~~ [4] (Refs. 2 and 3, respectively). The ability to provide onsite emergency AC power is dependent on the ability of the ~~DG [1B] SSW~~ DGCW [4] System to cool the DGs ~~[1B]~~. U

The ~~DG [1B] SSW~~ DGCW [4] System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

~~The OPERABILITY of the DG [1B] SSW System is required to provide a coolant source to ensure effective operation of the DG [1B] in the event of an accident or transient.~~ Insert B 3.7.2 LCO

The OPERABILITY of ~~the DG [1B] SSW System~~ each DGCW subsystem is based on having an OPERABLE pump and an OPERABLE flow path ~~.~~ capable of taking suction from the CSCS service water tunnel and transferring cooling water to the associated diesel generator, LPCS pump motor cooling coils, and ECCS cubicle area cooling coils, as required.

An adequate suction source is not addressed in this LCO since the minimum net positive suction head of the ~~DG [1B] SSW~~ DGCW [4] pump ~~is bounded by the PSW requirements~~ and the maximum suction source temperature are covered by the requirements specified in LCO 3.7.~~2~~ 3 ~~"[Unit Service Water (PSW)] System and [Ultimate~~ Heat Sink (UHS)~~]."~~

(continued)

* This BWR/4 Bases is used to match the BWR/4 Specification inserted in the LCO section.

Insert B 3.7.2 BKGD

The DGCW System consists of three independent cooling water headers (Divisions 1, 2, and 3), and their associated pumps, valves, and instrumentation. The pump and header for the Division 1 DGCW subsystem is common to both units (and supplies cooling to equipment on both units). The other divisions have independent pumps and suction headers.

The following combinations of DGCW pumps are sized to provide sufficient cooling capacity to support the required safety related systems during safe shutdown of the unit following a loss of coolant accident (LOCA):

a. The unit Division 1 and 2 DGCW pumps;

b. The unit Division 1 and 3 DGCW pumps and opposite unit's Division 2 DGCW pump; or

c. The unit Division 2 and 3 DGCW pumps.

The unit Division 1 DGCW subsystem services its associated Diesel Generator (DG) and ECCS cubicle area coolers, and the LPCS pump motor cooler. The unit Division 2 DGCW subsystem services its associated DG and ECCS cubicle area cooler. The unit Division 3 DGCW subsystem services the High Pressure Core Spray (HPCS) DG and its associated ECCS cubicle area cooler. The opposite unit Division 2 DGCW subsystem services its associated DG for support of systems required by both units.

The DGCW and the Residual Heat Removal Service Water (RHRSW) subsystems are subsystems to the Core Standby Cooling System (CSCS) – Equipment Cooling Water System (ECWS). The CSCS – ECWS consists of three independent piping subsystems corresponding to essential electrical power supply Divisions 1, 2, and 3. The CSCS – ECWS subsystems take a suction from the service water tunnel located in the Lake Screen House. Each DGCW pump auto-starts upon receipt of a diesel generator (DG) start signal when power is available to the pump's electrical bus or on start of ECCS cubicle area coolers. The Division 1 DGCW pump also auto-starts upon receipt of a start signal for the LPCS pump. Cooling water is then pumped from the service water tunnel by the DGCW pumps to the supported systems and components (i.e., the DGs, LPCS pump motor cooler, and the ECCS cubicle area coolers). After removing heat from these systems and components, the water from the DGCW subsystem is discharged to the CSCS pond (i.e., the Ultimate Heat Sink) through a discharge line that is common to the corresponding divisional discharge from the other unit. The discharge line terminates in the discharge structure at an elevation above the normal CSCS Pond level.

Insert B 3.7.2 LCO

The unit's Division 1, 2, and 3, and the opposite unit's Division 2 DGCW subsystems are required to be OPERABLE to ensure the effective operation of the DGs, the LPCS pump motor, and the ECCS equipment supported by the ECCS cubicle area coolers during a DBA or transient.

Insert BWR/4 ISTS B 3.7.3 (continued)

1

BASES (continued)

APPLICABILITY

~~The requirements for OPERABILITY of the DG [1B] SSW System are governed by the required OPERABILITY of the DG [1B] (LCO 3.8.1, "AC Sources—Operating," and LCO 3.8.2, "AC Sources—Shutdown").~~

Insert APPL 5

ACTIONS

Insert ACTIONS 5

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

~~The Required Actions are modified by a Note indicating that the LCO 3.0.4 does not apply. As a result, a MODE change is allowed when the DG [1B] SSW System is inoperable, provided the DG [1B] has an adequate cooling water supply from the Unit [1] PSW.~~

~~If the DG [1B] SSW System is inoperable, the OPERABILITY of the DG [1B] is affected due to loss of its cooling source; however, the capability exists to provide cooling to DG [1B] from the PSW System of Unit [1]. Continued operation is allowed for 60 days if the OPERABILITY of a Unit 1 PSW System, with respect to its capability to provide cooling to the DG [1B], can be verified. This is accomplished by aligning cooling water to DG [1B] from the Unit 1 PSW System within 8 hours and verifying this lineup once every 31 days. The 8 hour Completion Time is based on the time required to reasonably complete the Required Action, and the low probability of an event occurring requiring DG [1B] during this period. The 31 day verification of the Unit [1] PSW lineup to the DG [1B] is consistent with the PSW valve lineup SRs. The 60 day Completion Time to restore the DG [1B] SSW System to OPERABLE status allows sufficient time to repair the system, yet prevents indefinite operation with cooling water provided from the Unit [1] PSW System.~~

5

A ~~B~~.1

one or more DGCW subsystems are inoperable,

~~If cooling water cannot be made available to the DG [1B] within the 8 hour Completion Time, or if cooling water cannot be verified to be aligned to DG [1B] from a Unit [1] PSW subsystem as required by the 31 day verification Required Action,~~ the associated DG~~, [1B]~~(s) and ECCS components, supported by the affected DGCW loop, including LPCS pump motor cooling coils or ECCS cubicle area cooling coils, as applicable, cannot perform ~~its~~ their intended function and must be immediately declared inoperable. In accordance with LCO 3.0.6, this also requires entering into the Applicable Conditions and Required Actions for LCO 3.8.1 "AC Sources - Operating," and LCO 3.5.1, "Emergency Core Cooling Systems (ECCS) - Operating." ~~or LCO 3.8.2. Additionally, if the DG [1B] SSW System is~~

5

(continued)

3 2

Insert APPL

In MODES 1, 2, and 3, the DGCW subsystems are required to support the OPERABILITY of equipment serviced by the DGCW subsystems and required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the DGCW subsystems are determined by the systems they support. Therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, the LCOs of the systems supported by the DGCW subsystems will govern DGCW System OPERABILITY requirements in MODES 4 and 5.

Insert ACTIONS

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

Insert BWR/4 ISTS B 3.7.3 1

(continued)

BASES

ACTIONS

~~B.1 (continued)~~

~~not restored to OPERABLE status within 60 days, DG [1B] must be immediately declared inoperable.~~ 5

SURVEILLANCE REQUIREMENTS

SR 3.7.~~3~~ 2.1 1

Verifying the correct alignment for manual, power operated, and automatic valves in ~~the DG [1B] SSW System~~ each required DGCW subsystem 4 flow path provides assurance that the proper flow paths will exist for ~~DG [1B] SSW System~~ DGCW subsystem 4 operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet be considered in the correct position provided it can be automatically realigned to its accident position, within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

SR 3.7.~~3~~ 2.2 1

This SR ensures that ~~the DG [1B] SSW System~~ each required DGCW subsystem 4 pump will automatically start to provide required cooling to the associated 2 DG ~~[1B]~~, LPCS pump motor cooling coils, and ECCS cubicle area cooling coils, as applicable, 2 when the associated 2 DG ~~[1B]~~ starts and the respective bus is energized, or on start of the applicable ECCS cubicle area cooler. For the Division 1 DGCW subsystem, this SR also ensures the DGCW pump automatically starts on receipt of a start signal for the unit LPCS pump. These starts may be performed by using actual or simulated initiation signals. 2

Operating experience has shown that these components usually pass the SR when performed at the ~~[18]~~ 24 4 month Frequency, which is based on the refueling cycle. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

Insert BWR/4 ISTS B3.7.3 $\boxed{1}$
(continued)

DGCW $\boxed{4}$
~~DG [1B] SSW~~ System
B 3.7.~~3~~ 2 $\boxed{1}$

BASES (continued)

REFERENCES $\boxed{2}$
1. UFSAR, Section ~~[9.8.5]~~ 9.2.1.
2. UFSAR, Chapter ~~[6]~~.
3. UFSAR, Chapter ~~[15]~~.

$\boxed{4}$

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.7.2 - DIESEL GENERATOR COOLING WATER (DGCW) SYSTEM

1. A new Specification has been added, ITS 3.7.2, for the DGCW System. This system is similar to, but not identical to, the Diesel Generator Standby Service Water System discussed in ISTS 3.7.3 of NUREG-1433, Revision 1. Thus, the Bases for proposed ITS 3.7.2 are based on ISTS 3.7.3 of NUREG-1433, Revision 1. The deviations from the BWR/4 ISTS are discussed blow.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Editorial changes made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Changes have been made to reflect changes made to the Specification.

B 3.7 PLANT SYSTEMS

B 3.7.1 3 [Standby Service Water (SSW)] System and [Ultimate Heat Sink (UHS)] 4 1

BASES

BACKGROUND

2

The [SSW] System is designed to provide cooling water for the removal of heat from unit auxiliaries, such as Residual Heat Removal (RHR) System heat exchangers, standby diesel generators (DGs), and room coolers for Emergency Core Cooling System equipment required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The [SSW] System also provides cooling to unit components, as required, during normal shutdown and reactor isolation modes. During a DBA, the equipment required for normal operation only is isolated from the [SSW] System, and cooling is directed only to safety related equipment.

The [SSW] System consists of the [UHS], two independent cooling water headers (subsystems A and B), and their associated pumps, piping, valves, and instrumentation. The two [SSW] pumps, or one [SSW] pump and the high pressure core spray service water pump, are sized to provide sufficient cooling capacity to support the required safety related systems during safe shutdown of the unit following a loss of coolant accident (LOCA). Subsystems A and B are redundant and service equipment in [SSW] Divisions 1 and 2, respectively.

2

2 INSERT B 3.7.3-BKGD

The [UHS] consists of two concrete makeup water basins, each containing one cooling tower with two fan cells per basin. The combined basin volume is sized such that sufficient water inventory is available for all [SSW] System post LOCA cooling requirements for a 30 day period with no external makeup water source available (Regulatory Guide 1.27, Ref. 1). Normal makeup for each basin is provided automatically by the Plant Service Water System.

Cooling water is pumped from the cooling tower basins by the two [SSW] pumps to the essential components through the two main redundant supply headers (subsystems A and B). After removing heat from the components, the water is discharged to the cooling towers where the heat is rejected through direct contact with ambient air.

Subsystems A and B supply cooling water to redundant equipment required for a safe reactor shutdown. Additional

(continued)

Insert B 3.7.3 BKGD

The UHS (i.e., the Core Standby Cooling System (CSCS) Pond) consists of the volume of water remaining in the cooling lake following the failure of the main dike. This water has a depth of approximately 5 feet and a top water elevation established at 690 feet. The volume of the remaining water in the cooling lake is sufficient to permit a safe shutdown and cooldown of the station for 30 days with no water makeup for both accident and normal conditions (Regulatory Guide 1.27, Ref. 1).

The CSCS Pond provides a source of water to the service water tunnel from which the Residual Heat Removal Service Water (RHRSW) and Diesel Generator Cooling Water (DGCW) pumps take suction. The service water tunnel is filled from the CSCS Pond by six inlet lines which connect to the circulating water pump forebays. Prior to entering the service water tunnel inlet pipes, the water is strained by the Lake Screen House traveling screens to prevent large pieces of debris from entering the system and blocking flow or damaging equipment. However, because the traveling screens are not safety related, a 54-inch bypass line around the screens, isolated by a normally closed manual valve, is provided to assure a continuous supply of CSCS Pond water to the service water tunnel.

Additional information on the design and operation of the CSCS Pond is provided in UFSAR, Sections 9.2.1 and 9.2.6 (Refs. 2 and 3). The excavation slopes of the CSCS Pond and flume are designed to be stable under all conditions of emergency operation while providing the capability to supply adequate cooling water to equipment required for safe reactor shutdown.

4 3 1

BASES

BACKGROUND (continued)

Information on the design and operation of the [SSW] System and [UHS] along with the specific equipment for which the [SSW] System supplies cooling water is provided in the FSAR, Section [9.2.1] and the FSAR, Table [9.2-3] (Refs. 2 and 3, respectively). The [SSW] System is designed to withstand a single active or passive failure, coincident with a loss of offsite power, without losing the capability to supply adequate cooling water to equipment required for safe reactor shutdown. 2

Following a DBA or transient, the [SSW] System will operate automatically without operator action. Manual initiation of supported systems (e.g., suppression pool cooling) is, however, performed for long term cooling operations.

APPLICABLE SAFETY ANALYSES

the CSCS pond

The volume of each water source incorporated in a [UHS] complex is sized so that sufficient water inventory is available for all [SSW] System post LOCA cooling requirements for a 30 day period with no additional makeup water source available (Ref. 1). The ability of the [SSW] System to support long term cooling of the reactor or containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the FSAR, Sections [9.2.1], [6.2.1.1.3.3.1.6] and Chapter [15], (Refs. 2, 4, and 5, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA. The [SSW] System provides cooling water for the RHR suppression pool cooling mode to limit suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its intended function of limiting the release of radioactive materials to the environment following a LOCA. The [SSW] System also provides cooling to other components assumed to function during a LOCA (e.g., RHR and Low Pressure Core Spray systems). Also, the ability to provide onsite emergency AC power is dependent on the ability of the [SSW] System to cool the DGs. 2 2 2

to permit the safe shutdown and cooldown of the units

for both normal and accident conditions

The safety analyses for long term containment cooling were performed, as discussed in the FSAR, Sections [6.2.1.1.3.3.1.6] and [6.2.2.3] (Refs. 4 and 6, respectively), for a LOCA, concurrent with a loss of offsite power, and minimum available DG power. The worst case single failure affecting the performance of the [SSW] System

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~is the failure of one of the two standby DGs, which would in turn affect one [SSW] subsystem. The [SSW] flow assumed in the analyses is [7900] gpm per pump to the heat exchanger (FSAR, Table [6.2-2], Ref. 7). Reference 2 discusses [SSW] System performance during these conditions.~~ 2

The ~~[SSW] System, together with the~~ [UHS], 4 CSCS satisfy ies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 2

LCO

~~The OPERABILITY of subsystem A (Division 1) and subsystem B (Division 2) of the [SSW] System is required to ensure the effective operation of the RHR System in removing heat from the reactor, and the effective operation of other safety related equipment during a DBA or transient. Requiring both subsystems to be OPERABLE ensures that either subsystem A or B will be available to provide adequate capability to meet cooling requirements of the equipment required for safe shutdown in the event of a single failure.~~ 2

~~A subsystem is considered OPERABLE when:~~

~~a. The associated pump is OPERABLE;~~

~~b. The associated [UHS] is OPERABLE; and~~

~~c. The associated piping, valves, instrumentation, and controls required to perform the safety related function are OPERABLE.~~

pond

OPERABILITY of the [UHS] 1 is based on a maximum water temperature of ~~[95]~~ 4 97 °F ~~with OPERABILITY of each subsystem requiring~~ 2 and a minimum ~~basin~~ water level at or above elevation ~~[130 ft 3 inches]~~ 4 690 ft mean sea level ~~(equivalent to an indicated level of ≥ [7 ft 3 inches]) and four OPERABLE cooling tower fans~~. 2

~~The isolation of the [SSW] System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the [SSW] System.~~ 2

~~OPERABILITY of the High Pressure Core Spray (HPCS) Service Water System (SWS) is addressed by LCO 3.7.2, "HPCS SWS."~~

(continued)

In addition, to ensure the volume of water available in the CSCS pond is sufficient to maintain adequate long term cooling, sediment deposition (in the intake flume and in the pond) must be ≤ 1.5 ft and CSCS pond bottom elevation must be ≤ 686.5 ft. 2

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the [SSW] System and [UHS] ~~are~~ is required to be OPERABLE to support OPERABILITY of the equipment serviced by the [SSW] System and [UHS], and ~~are~~ required to be OPERABLE in these MODES. 4 4 1

In MODES 4 and 5, the OPERABILITY requirements of the [SSW] System and [UHS] ~~are~~ is determined by the systems ~~they~~ it support. 4 5 1

3 INSERT B3.7.3 APPL.

ACTIONS

A.1

1 INSERT B3.7.3 ACTION A

~~If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days.~~ 1

~~The 7 day Completion Time is reasonable, based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable in one or more cooling towers, the number of available systems, and the time required to complete the Required Action.~~

~~B.1~~

~~If one [SSW] subsystem is inoperable [for reasons other than Condition A], it must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE [SSW] subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE [SSW] subsystem could result in loss of [SSW] function. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.~~

~~The Required Action is modified by two Notes indicating that the applicable Conditions of LCO 3.8.1, "AC Sources—Operating," and LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," be entered and the Required Actions taken if the inoperable [SSW] subsystem results in an inoperable DG or RHR shutdown cooling, respectively. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for these components.~~

(continued)

Insert B 3.7.3 APP

Therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. The LCOs of the systems supported by the UHS will govern UHS OPERABILITY requirements in MODES 4 and 5.

Insert B 3.7.3 Action A

If the CSCS pond is inoperable, due to sediment deposition > 1.5 ft (in the intake flume, CSCS pond, or both) or the pond bottom elevation > 686.5 ft, action must be taken to restore the inoperable UHS to an OPERABLE status within 90 days. The 90 day Completion Time is reasonable based on the low probability of an accident occurring during that time, historical data corroborating the low probability of continued degradation (i.e., further excessive sediment deposition or pond bottom elevation changes) of the CSCS pond during that time, and the time required to complete the Required Action.

BASES

ACTIONS (continued)

1 B.1 and B.2 CSCS pond 1

If the ~~[SSW] subsystem~~ cannot be restored to OPERABLE status within the associated Completion Time, ~~or both [SSW] subsystems are inoperable [for reasons other than Condition A]~~, or the ~~[UHS]~~ is determined inoperable for reasons other than Condition A, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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. 1

2 (e.g., inoperable due to CSCS pond average water temperature > 97°F)

CSCS pond 4

SURVEILLANCE REQUIREMENTS

~~SR 3.7.1.1~~ 1

~~This SR ensures adequate long term (30 days) cooling can be maintained. With the [UHS] water source below the minimum level, the affected [SSW] subsystem must be declared inoperable. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~

~~SR 3.7.1.2~~

~~This SR verifies the water level [in each [SSW] pump well of the intake structure] to be sufficient for the proper operation of the [SSW] pumps (net positive suction head and pump vortexing are considered in determining this limit). The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~

SR 3.7.1.3 3.1

CSCS pond

Verification of the [UHS] temperature ensures that the heat removal capability of the [SSW] System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. 4

2 RHRSW System and DGCW System are

ies

(continued)

Insert SRs 3.7.3.2 and 3.7.3.3 1

Insert SRs 3.7.3.2 and 3.7.3.3

SR 3.7.3.2

This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the sediment level in the intake flume and the CSCS pond is $\leq$ 1.5 feet. Sediment level is determined by a series of sounding cross-sections compared to as-built soundings. The 24 month Frequency is based on historical data and engineering judgement regarding sediment deposition rate.

SR 3.7.3.3

This SR ensures adequate long term (30 days) cooling can be maintained, by verifying the CSCS pond bottom elevation is $\leq$ 686.5 feet. The 24 month Frequency is based on historical data and engineering judgement regarding pond bottom elevation changes.

BASES

1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.4

Operating each cooling tower fan for ≥ 15 minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the cooling tower fans occurring between Surveillances.

SR 3.7.1.5

Verifying the correct alignment for each manual, power operated, and automatic valve in each [SSW] subsystem flow path provides assurance that the proper flow paths will exist for [SSW] operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position and yet considered in the correct position, provided it can be automatically realigned to its accident position. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR is modified by a Note indicating that isolation of the [SSW] System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the [SSW] System. As such, when all [SSW] pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the [SSW] System is still OPERABLE.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

1

~~SR 3.7.1.6~~

~~This SR verifies that the automatic isolation valves of the [SSW] System will automatically switch to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This is demonstrated by use of an actual or simulated initiation signal. This SR also verifies the automatic start capability of the [SSW] pump and cooling tower fans in each subsystem. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.5.1.6 overlaps this SR to provide complete testing of the safety function.~~

~~Operating experience has shown that these components usually pass the SR when performed on the [18] month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.~~

REFERENCES

1. Regulatory Guide 1.27, Revision 2, January 1976.
2. 2. UFSAR, Section ~~[~~9.2.1~~]~~. 4
3. ~~FSAR, Table [9.2-3].~~ 2
4. ~~FSAR, Section [6.2.1.1.3.3.1.6].~~
5. ~~FSAR, Chapter [15].~~
6. ~~FSAR, Section [6.2.2.3].~~
7. ~~FSAR, Table [6.2-2].~~

3. UFSAR, Section 9.2.6. 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.7.3 - ULTIMATE HEAT SINK (UHS)

1. Changes have been made to reflect changes made to the Specifications.

2. Changes have been made (additions, deletions, and/or change to the NUREG) to reflect plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. The Applicability Section of the Bases has been revised to add clarification regarding OPERABILITY requirements for the UHS during MODES 4 and 5, since the ITS does not have an LCO for the UHS in these MODES.

4. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.7 PLANT SYSTEMS

Area Filtration (CRAF) 2

1 B 3.7.~~5~~ 4 [Control Room ~~Fresh Air (CRFA)~~ System

BASES

3 (i.e., the emergency make up air filter units (EMUs) for treatment of outside supply air. Recirculation filters are also provided for treatment of recirculated air.

BACKGROUND

2 CRAF

3 (control room and auxiliary electric equipment room)

The ~~[CRFA]~~ System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA). INSERT B 3.7.4 BKGD-A 3

2 CRAF

The safety related function of the ~~[CRFA]~~ System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems ~~for treatment of recirculated air or outside supply air.~~ Each EMU subsystem consists of a demister, an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork, ~~and~~ dampers, and instrumentation and controls 3. Demisters remove water droplets from the airstream. The electric heater reduces the relative humidity of the air entering the EMUs. 3 Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay. 3 INSERT B 3.7.4 BKGD - B

2 CRAF

In addition to the safety related standby emergency filtration function, parts of the ~~[CRFA]~~ System that are shared with the Control Room Area HVAC System 3 are operated ~~to maintain the control room environment during normal operation.~~ Upon receipt of ~~the initiation~~ a high radiation 3 signal~~(s)~~ from the outside air intake 3 (indicative of conditions that could result in radiation exposure to control room personnel), the 3 area 2 CRAF ~~[CRFA]~~ System automatically ~~switches to the isolation mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and control room air flow is recirculated and processed through either of the two filter subsystems.~~ 3 Insert B 3.7.4 BKGD-C

2 CRAF

The ~~[CRFA]~~ System is designed to maintain the control room area 3 environment for a 30 day continuous occupancy after a DBA, without exceeding a 5 rem whole body dose or its equivalent to any part of the body. ~~[CRFA]~~ CRAF 2 System operation in maintaining the control room area 3 habitability is discussed in the FSAR, Sections [6.5.1] 6.4, 3 and [9.4.1] 1 (Refs. 1, ~~and~~ 2, and 3, 3 respectively). 2 a 3

(continued)

Insert B 3.7.4 BKGD-A

The Control Room Area Heating Ventilation and Air Conditioning (HVAC) System is comprised of the Control Room HVAC System and the Auxiliary Electric Equipment Room (AEER) HVAC System. The Control Room HVAC System is common to both units and serves the control room, main security control center, and the control room habitability storage room (toilet room). The AEER HVAC System is common to both units and services the auxiliary electrical equipment rooms. The control room area is comprised of the areas covered by the Control Room and AEER HVAC Systems.

Insert B 3.7.4 BKGD-B

Each Control Room and AEER Ventilation System has a charcoal recirculation filter in the supply of the system that is normally bypassed. In addition, the OPERABILITY of the CRAF System is dependent upon portions of the Control Room Area HVAC System, including the control room and auxiliary electric equipment room outside air intakes, supply fans, ducts, dampers, etc.

Insert B 3.7.4 BKGD-C

isolates the normal outside air supply to the Control Room Area HVAC System, and diverts the minimum outside air requirement through the EMUs before delivering it to the control room area. The recirculation filters for the control room and AEER must be manually placed in service within 4 hours of receipt of any control room high radiation alarm.

BASES (continued)

area 3

APPLICABLE SAFETY ANALYSES	The ability of the CRFA CRAF System to maintain the habitability of the control room area is an explicit assumption for the safety analyses presented in the USAR, Chapters ~~6~~ 2 and ~~15~~ (Refs. ~~3 and~~ 4 and 5 respectively). The ~~isolation~~ pressurization mode of the ~~CRFA~~ CRAF System is assumed to operate following a loss of coolant accident, main steam line break, fuel handling accident, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference ~~4~~ 5. No single active ~~or passive~~ failure will cause the loss of outside or recirculated air from the control room area. 3 2 3 3 3
	The ~~CRFA~~ CRAF System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 2 3 3
LCO	Two redundant subsystems of the ~~CRFA~~ CRAF System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA. 2
	The ~~CRFA~~ CRAF System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated: EMU is OPERABLE and the associated charcoal recirculation filters for the Control Room and AEER are OPERABLE. An EMU is considered OPERABLE when its associated 2 3
	a. Fan is OPERABLE;
	b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and
	c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation through the EMU can be maintained. 3
	3 INSERT B 3.7.4 LCO-A
	In addition, the control room area boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors. 3
	4 INSERT B 3.7.4 LCO-B
APPLICABILITY	In MODES 1, 2, and 3, the ~~CRFA~~ CRAF System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release. 2

(continued)

Insert B 3.7.4 LCO-A

Additionally, the portions of the Control Room Area HVAC System that supply the outside air to the EMUs are required to be OPERABLE. This includes the outside air intakes, associated dampers and ductwork.

Insert B 3.7.4 LCO-B

, such that the pressurization limit of SR 3.7.4.5 can be met. However, it is acceptable for access doors to be open for normal control room area entry and exit and not consider it to be a failure to meet the LCO.

BASES

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRFA System CRAF 2 OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

c a. During operations with a potential for draining the reactor vessel (OPDRVs);.

b. During CORE ALTERATIONS; and

a c. During movement of irradiated fuel assemblies in the 2 primary or secondary containment; 2

5

ACTIONS

A.1

CRAF 2 CRAF 2

With one CRFA subsystem inoperable, the inoperable CRFA subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CRFA subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of CRFA System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

2 CRAF

B.1 and B.2

CRAF 2

In MODE 1, 2, or 3, if the inoperable CRFA subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1, C.2.2, and C.2.3

5 LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. ~~Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~ 5 Insert ACTION C

During movement of irradiated fuel assemblies in the 2 ~~primary or~~ secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CRFA CRAF 2 subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRFA subsystem may be placed in the isolation 3 Pressurization mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

6 ~~Required Action C.1 is modified by a Note alerting the operator to [place the system in the toxic gas protection mode if the toxic gas, automatic transfer capability is inoperable].~~

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room 3 area. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the ~~primary and~~ secondary containment must be suspended immediately. 2 Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action 7 must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action 7 must continue until the OPDRVs are suspended.

(continued)

Insert ACTION C

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

BASES

ACTIONS (continued)

D.1

CRAF 2

If both CRFA subsystems are inoperable in MODE 1, 2, or 3, the CRFA System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

5 LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,

E.1, E.2, and E.3

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

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

CRAF 2

2 During movement of irradiated fuel assemblies in the ~~primary or~~ secondary containment, during CORE ALTERATIONS, ~~or during OPDRVs~~, with two CRFA subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the ~~primary and~~ secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended. 2 7 7

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 4 1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing

(continued)

Insert E.1, E.2, and E.3

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

3 for ≥ 10 continuous hours during system operation

each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. ~~[Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.]~~ 2 6 Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

6 INSERT SR 3.7.4.2

SR 3.7.3.2 4 1 3 6

5 Specification 5.5.8," CRAF 2

This SR verifies that the required CRFA testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CRFA filter tests are in accordance with ~~Regulatory Guide 1.52 (Ref. 5)~~. The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP]. 2 5 2

ANSI/ASME N 510-1989 (Ref. 6) 3 7

SR 3.7.3.3 1 4 4 6

CRAF 2 3 air intake radiation monitors

13 automatically switches to the pressurization mode of operation 3

This SR verifies that each CRFA subsystem ~~starts and~~ operates on an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.6 overlaps this SR to provide complete testing of the safety function. ~~The [18] month Frequency is specified in Reference 5.~~ 3 4 6 3

Operating experience has shown that these components normally pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was found to be acceptable from a reliability standpoint 3

SR 3.7.3.4 1 4 5 6

area 3 area 3

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CRFA System. During the emergency mode of operation, the CRFA System is designed to slightly pressurize the control room to [0.1] inches water gauge positive pressure with respect to adjacent areas to prevent unfiltered inleakage. The CRFA 2 CRAF

≥0.125 2

pressurization 3 3 area

(continued)

Insert SR 3.7.4.2

SR 3.7.4.2

This SR verifies that flow can be manually realigned through the CRAF System recirculation filters and maintained for $\geq$ 10 hours. Standby systems should be checked periodically to ensure that they function. Monthly operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and two subsystem redundancy available.

4 1 5 6

BASES

SURVEILLANCE REQUIREMENTS

3

SR 3.7.~~3~~.~~4~~ (continued)

≤4000 2 3 area 3

System is designed to maintain this positive pressure at a flow rate of ~~3500~~ cfm to the control room area in the ~~isolation~~ pressurization mode. This test also requires manual initiation of flow through the control room and AEER recirculation filters like when the CRAF System is in the pressurization mode of operation. The Frequency of ~~18~~ 24 months ~~on a STAGGERED TEST BASIS~~ is consistent with industry practice and other filtration system SRs.

6 2

REFERENCES

1. UFSAR, Section 6.4.
2. ~~1.~~ UFSAR, Section ~~[~~6.5.1~~]~~.
3. ~~2.~~ UFSAR, Section ~~[~~9.4.1~~]~~.
4. ~~3.~~ UFSAR, Chapter ~~[~~6~~]~~.
5. ~~4.~~ UFSAR, Chapter ~~[~~15~~]~~.
6. ~~5.~~ ~~Regulatory Guide 1.52, Revision 2, March 1978~~ ANSI/ASME N510-1989.

3 2 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.7.4 - CONTROL ROOM AREA FILTRATION (CRAF) SYSTEM

1. The Bases has been renumbered to accommodate additional plant specific changes to ISTS Section B 3.7.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. These words have been added to clarify that the boundary is not necessarily required to be leak-tight, but is required to meet the leak tightness requirements of SR 3.7.4.5 (i.e., leakage can occur as long as a 0.125 inch pressure is maintained in the control room area). Also, an allowance to open control room access doors for entry and exit has been added.

5. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

6. Changes have been made to reflect changes made to the Specification.

7. Changes have been made to more closely match the LCO requirements.

2 Area Ventilation

B 3.7 PLANT SYSTEMS

B 3.7 4 ~~Control Room Air Conditioning (AC)~~ System 5 1

BASES

BACKGROUND

INSERT BKGD 2 3 5

~~The [Control Room AC] System provides temperature control for the control room following isolation of the control room.~~

~~The [Control Room AC] System consists of two independent, redundant subsystems that provide cooling and heating of recirculated control room air. Each subsystem consists of heating coils, cooling coils, fans, chillers, compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.~~

~~The [Control Room AC] System is designed to provide a controlled environment under both normal and accident conditions. A single subsystem provides the required temperature control to maintain a suitable control room environment for a sustained occupancy of 12 persons. The design conditions for the control room environment are 72°F and 50% relative humidity. The [Control Room AC] System operation in maintaining the control room temperature is discussed in the FSAR, Sections [6.4] and [9.4.1] (Refs. 1 and 2, respectively).~~

APPLICABLE SAFETY ANALYSES

2 Area Ventilation

The design basis of the ~~[Control Room AC]~~ System is to maintain, 5 temperatures of the the control room ~~temperature~~ and AEERs 3 for a 30 day period after a Design Basis Accident (DBA). 3 ~~continuous occupancy~~.

The ~~[Control Room AC]~~ 2 Area Ventilation System components are arranged in redundant safety related subsystems. During emergency operation, the ~~[Control Room AC]~~ System maintains a habitable environment and ensures the OPERABILITY of components in the control room and AEERs 3. A single active failure of a component of the ~~[Control Room AC]~~ System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The ~~[Control Room AC]~~ System is designed in accordance with Seismic Category I requirements, with exceptions described in UFSAR Section 9.4.1.1.1 (Ref. 3) 3. The ~~[Control Room AC]~~ 12 Area Ventilation System is capable of removing sensible and latent heat loads from the control room, and AEERs 3 including consideration of equipment

(continued)

Insert BKGD

The Control Room Area Ventilation AC System provides temperature control for the control room area. The control room area is comprised of the control room and the Auxiliary Electric Equipment Rooms (AEERs).

The Control Room Area Ventilation AC System is comprised of two independent, redundant subsystems that provide cooling and heating of control room air and the auxiliary electric equipment rooms air. Each Control Room Area Ventilation AC subsystem consists of a Control Room AC subsystem and an AEER AC subsystem. The associated Control Room AC and AEER AC subsystems share a common outside air intake with a common emergency makeup air filter unit. The Control Room AC System is common to both units and serves the control room, main security control center, and the control room habitability storage room (toilet room). The AEER AC System is common to both units and services the AEERs.

Each Control Room Area Ventilation AC subsystem is powered from a Division 2 power source. One subsystem is powered from Unit 1 Division 2 and the other subsystem is powered from Unit 2 Division 2.

Each control room AC and AEER AC subsystem consists of a supply air filter, supply and return air fans, direct expansion cooling coils, an air-cooled condenser, a refrigerant compressor and receiver, heating coils, ductwork, dampers, and instrumentation and controls to provide temperature control for their respective areas. However, the heating coils are not safety related.

The Control Room Area Ventilation AC System is designed to provide a controlled environment under both normal and accident conditions. A single control room area ventilation AC subsystem provides the required temperature control to maintain a suitable control room and AEER environment for a sustained occupancy of at least the required normal and emergency shift crew complements. The design conditions for habitability of the control room and AEER environment are 65°F to 85°F and a maximum of 50% relative humidity. The Control Room Area Ventilation AC System operation in maintaining the temperatures of the control room and AEERs is discussed in the UFSAR, Sections 6.4 and 9.4.1 (Refs. 1 and 2, respectively).

BASES

APPLICABLE SAFETY ANALYSES (continued)

heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

Area Ventilation 2

The [Control Room AC] System satisfies Criterion 3 of the ~~NRC Policy Statement~~. 3 10 CFR 50.36(c)(2)(ii) 3

LCO

Two independent and redundant subsystems of the [Control Room AC] System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits. 2 Area Ventilation 3 Subsystem

Area Ventilation 2

The [Control Room AC] System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the ~~cooling coils, fans, chillers, compressors, ductwork, dampers, and associated instrumentation and controls~~. 5 5 and AEERs 3

3 Insert 3.7.5 LCO-A

APPLICABILITY

2 Area Ventilation 5 5

In MODE 1, 2, or 3, the [Control Room AC] System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following ~~control room isolation~~. 3 and AEERs

3 during operation of the Control Room Area Filtration (CRAF) System in the pressurization mode

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the [Control Room AC] System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

2 Area Ventilation

5

a. (c) During operations with a potential for draining the reactor vessel (OPDRVs); .

b. During CORE ALTERATIONS; and

c. (a) During movement of irradiated fuel assemblies in the ~~primary or~~ secondary containment. ; 2

(continued)

Insert B 3.7.5 LCO-A

supply and return air fans, direct expansion cooling coils, an air-cooled condenser, a refrigerant compressor and receiver, ductwork, dampers, and instrumentation and controls.

BASES (continued)

ACTIONS

A.1

area ventilation 2 2 2

With one {control room AC} subsystem inoperable, the inoperable {control room AC} subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE {control room AC} subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring ~~control room isolation~~, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate cooling methods. 3

operation of the CRAF System in the pressurization mode and

B.1 and B.2

area ventilation 2

In MODE 1, 2, or 3, if the inoperable {control room AC} subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

C.1, C.2.1, C.2.2, and C.2.3

5 Insert B3.7.5 Action C.1 -A

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. 5

If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. ~~Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

5 Insust B 3.7.5 Action C.1 - B

2 During movement of irradiated fuel assemblies in the ~~{primary or~~ secondary containment}, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE {control room AC} subsystem may be placed immediately in operation. 2

(continued)

Insert B 3.7.5 Action C.1-A

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,

Insert B 3.7.5 Action C.1-B

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the [primary and secondary containment] [2] must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions [5] must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions [5] must continue until the OPDRVs are suspended.

D.1

area ventilation [2]

If both [control room AC] subsystems are inoperable in MODE 1, 2, or 3, the [Control Room AC] System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately. Area Ventilation [2]

E.1, E.2, and E.3

The Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

area ventilation [2]

During movement of irradiated fuel assemblies in the [2] [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs with two [control room AC] subsystems [2] inoperable, action must be taken to immediately suspend activities that present a potential for releasing

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

2 If applicable, CORE ALTERATIONS and handling of irradiated fuel in the ~~[primary or~~ secondary containment~~]~~ must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action~~s~~ must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action~~s~~ must continue until the OPDRVs are suspended. 5 5

SURVEILLANCE REQUIREMENTS

INSERT 4

~~SR 3.7.4.1~~

~~This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the [safety analyses]. The SR consists of a combination of testing and calculation. The [18] month Frequency is appropriate since significant degradation of the [Control Room AC] System is not expected over this time period.~~

REFERENCES 3 U

1. UFSAR, Section ~~[6.4]~~. 2
2. UFSAR, Section ~~[9.4.1]~~. 2
3. UFSAR, Section 9.4.1.1.1.1. 3

INSERT

SR 3.7.5.1

This SR monitors the control room and AEER temperatures for indication of Control Room Area Ventilation AC System performance. Trending of control room area temperature will provide a qualitative assessment of refrigeration unit OPERABILITY. Limiting the average temperature of the Control Room and AEER to less than or equal to 85°F provides a threshold beyond which the operating control room area ventilation AC subsystem is no longer demonstrating capability to perform its function. This threshold provides margin to temperature limits at which equipment qualification requirements could be challenged. Subsystem operation is routinely alternated to support planned maintenance and to ensure each subsystem provides reliable service. The 12 hour Frequency is adequate considering the continuous manning of the control room by the operating staff.

SR 3.7.5.2

Verifying proper breaker alignment and power available to the control room area ventilation AC subsystems provides assurance of the availability of the system function. The 7 day Frequency is appropriate in view of other administrative controls that assure system availability.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.7.5 - CONTROL ROOM AREA VENTILATION AIR CONDITIONING (AC) SYSTEM

1. The Bases has been renumbered to accommodate other plant specific changes to ISTS Section B 3.7.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. Changes have been made to reflect changes made to the Specification.

5. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

B 3.7 PLANT SYSTEMS

B 3.7.~~5~~ 6 [1] Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases. [2]

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and ~~moisture~~ water separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the ~~moisture~~ water separator prior to entering the holdup line. [3]

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event as discussed in the U FSAR, Section ~~[15.7.1]~~ .1 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits ~~(NUREG-0800, Ref. 2)~~ of 10 CFR 100 (Ref. ~~3~~ 2) ~~, or the NRC staff approved licensing basis~~. [3] [4]

The main condenser offgas limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). [3]

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/Mwt-second after decay of 30 minutes. The LCO is conservatively established ~~consistent with~~ based on the safety analysis discussed in Reference 1. [3]

(continued)

BASES

LCO (continued)

this requirement ([3833] Mwt x 100 µCi/Mwt-second = [380] mCi/second). 3

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, main steam is not being exhausted to the main condenser and the requirements are not applicable. 2

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment considering the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of a Main Condenser Offgas System rupture occurring.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits within the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from the sources of the significant radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in each drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems. 4 2

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 6 † 3 taken prior to the holdup line 3

representative

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable based on operating experience.

3 Xe-135m

m 3

3 (as indicated by the offgas pre-treatment noble gas activity monitor)

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates. 4

REFERENCES

3 1. ~~FSAR~~ U, Section 15.7.1 .1 4

3 ~~2. NUREG-0800.~~

3 ~~3.~~ 2 10 CFR 100.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.7.6 - MAIN CONDENSER OFFGAS

1. The Bases has been renumbered to accommodate additional plant specific changes to ISTS Section B 3.7.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Turbine Bypass System 7 1

TSTF-319 changes not adopted 6

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is ~~[35]~~ approximately 25 2 % of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of a ~~two valve chest~~ five valves mounted on a valve manifold 3 connected to the main steam lines between the main steam isolation valves and the main 3 turbine stop valves. Each of these valves is sequentially operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the U 3 FSAR, Section ~~[7.7.1.5]~~ 7.7.5.2 2 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves, directing all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the ~~bypass chest~~ bypass valve outlet manifold 3, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser (Ref. 2) 3.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the ~~design basis~~ turbine trip, turbine generator load rejection, and 3 feedwater controller failure ~~, maximum demand event~~ transients 3, described in the U 3 FSAR, Sections ~~[15.1.2]~~ 15.2.3, 15.2.2A, and 15.1.2A 3 (Refs. ~~2~~ 3, 4, and 5, respectively 3). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty.

The Main Turbine Bypass System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 3

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, such that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow continued operation. 2

An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Refs. 2). The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. 3, 4, and 5 3

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit ~~and the cladding 1% plastic strain limit are~~ is not violated during the turbine trip, feedwater controller failure, maximum demand event, and turbine generator load rejection transients. 3 ~~As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and~~ LCO 3.2.2, sufficient margin to these limits exists < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level. 3 4

ACTIONS

~~A.1~~ and 5

~~If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.~~ 2

(continued)

BASES

ACTIONS (continued)

B.1

and 5

If the Main Turbine Bypass System cannot be restored to OPERABLE status ~~or~~ the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine trip, turbine generator load rejection, and feedwater controller failure, maximum demand ~~event~~ 3 transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~5~~.1 7 1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 6 7 ~~31~~ day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.~~5~~.2 7 1

Simulated 3

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 2 24 ~~[18]~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 2 24 ~~[18]~~ month Frequency, which is based on the refueling cycle. 3 3 Therefore the frequency was concluded to be ~~is~~ acceptable from a reliability standpoint.

(continued)

BASES

3 as defined in the transient analysis inputs for the cycle,

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.~~6~~.3 7 1

es 3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analys~~is~~. The response time limits are specified in ~~[unit specific documentation]~~. The ~~[18]~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the ~~[18]~~ month Frequency, which is based on the refueling cycle, ~~is~~ acceptable from a reliability standpoint.

2 the Technical Requirements Manual (TRM)

24 2

3 that these components usually pass the SR when performed at

24 2

3 Therefore, the Frequency was concluded to be

REFERENCES

1. U FSAR, Section ~~[7.7.1.5]~~. 7.7.5.2
~~2~~. 5 U FSAR, Section ~~[15.1.2]~~. 15.1.2A 2

2. UFSAR, Section 10.4.4. 3

3. UFSAR, Section 15.2.3.
4. UFSAR, Section 15.2.2A. 3

6. Technical Requirements Manual. 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.7.7 - MAIN TURBINE BYPASS SYSTEM

1. The Bases has been renumbered to accommodate additional plant specific changes to ISTS Section B 3.7.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. This LCO is needed to ensure the MCPR limit is not exceeded. The cladding 1% plastic strain limit is an LHGR concern, not an MCPR concern. Therefore, this statement has been deleted. In addition, the statement that refers to the APHLGR Bases has also been deleted, because this LCO is only concerned with MCPR.

5. Typographical/grammatical error corrected.

6. Changes have been made to reflect changes made to the Specification.

B 3.7 PLANT SYSTEMS

B 3.7.8 Fuel Pool Water Level [1] Storage [2] Spent

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool ~~and upper containment fuel storage pool~~ meets the assumptions of iodine decontamination factors following a fuel handling accident. [2]

A general description of the spent fuel storage pool ~~and upper containment fuel storage pool~~ design is found in the [2] U FSAR, Section ~~9.1.2~~ 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Sections ~~15.7.4~~ 15.7.4 and ~~15.7.6~~ (Refs. ~~2~~ 1 and ~~3~~ 2, respectively). [2] [3]

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% ([2] 4 NUREG-0800, Section 15.7.4, Ref. ~~4~~ 3) [2] of the 10 CFR 100 (Ref. ~~5~~ 4) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. ~~6~~ 5). [2] [2]

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto ~~stored fuel bundles~~ the reactor core. The consequences of a fuel handling accident ~~inside the auxiliary building and inside containment are documented in References 2 and 3, respectively.~~ over the spent fuel storage pool are less severe than those of the fuel handling accident over the reactor core (Ref. 2). [2] The water level~~s~~ in the spent fuel storage pool ~~and upper containment fuel storage pool~~ provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident. [2] [6]

The Spent fuel Storage pool water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). [2] TSTF-139 Change not adopted

(continued)

BASES (continued)

LCO

The specified water level preserves the assumption of the fuel handling accident analysis (Refs. 2 and 3). As such, it is the minimum required for fuel movement within the spent fuel storage pool and upper containment fuel storage pool. 2 2

APPLICABILITY

This LCO applies whenever spent 2 movement of irradiated fuel assemblies occurs in the associated fuel storage racks pool 2 since the potential for a release of fission products exists. or whenever movement of new fuel assemblies occurs in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool, 5

ACTIONS

A.1

4 LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since fuel assembly movement can occur in MODES 1, 2, or 3,

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated 5 fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. ~~Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.~~ INSERT 3.7.8 A.1 4

When the initial the spent 2 conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. With ~~either~~ the spent 2 fuel Storage pool level less than required, the movement of irradiated 5 fuel assemblies in the associated 2 storage pool is suspended immediately. 2 Spent fuel Suspension of this activity shall not preclude completion of movement of ~~an~~ 5 irradiated 5 fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 8 1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool and upper containment fuel storage racks 2 2 must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable and water level changes are controlled by unit procedures.

(continued)

Insert 3.7.8 A.1

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of fuel assembly movement are not postponed due to entry into LCO 3.0.3.

BASES (continued)

REFERENCES 2 U

1. FSAR, Section [9.1.2]. 3

2. FSAR, Section [15.7.4]. 3

~~3. FSAR, Section [15.7.6].~~ 2

2

3 ~~4.~~ NUREG-0800, Section 15.7.4, Revision 1, July 1981.

4 ~~5.~~ 10 CFR 100.

5 ~~6.~~ Regulatory Guide 1.25, March 1972.

1. The Bases has been renumbered to accommodate additional plant specific changes to ISTS Section B 3.7.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

5. Changes have been made to be consistent with changes made to the Specification.

6. TSTF-139 changed the Applicable Safety Analyses section to also state that spent fuel pool water level meets Criterion 3 (in addition to meeting Criterion 2, which is stated in Rev. 1 of the ISTS Bases). 10 CFR 50.36(c)(2)(ii) describes Criterion 3 as a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The justification for TSTF-139 states that fuel pool water level is a process variable which satisfies Criteria 2 and 3. A process variable is not a structure, system, or component. The Interim and Final Policy Statements, as well as the statement of considerations for the change to 10 CFR 50.36 (that added the four criteria to 10 CFR 50.36(c)(2)(ii)) state that Criterion 3 is for equipment only. Criterion 2 was specifically developed for process variables. The ISTS Bases currently states that spent fuel pool water level meets Criterion 2 only, which is correct. Therefore, this TSTF has not been adopted. In addition, other Technical Specification Bases for water level requirements (e.g., ISTS 3.9.6 and ISTS 3.9.7, RPV Water Level requirements, which are in Technical Specifications for the same reason as the spent fuel pool water level requirements, and ISTS 3.6.2.2, Suppression Pool Water Level) state that the water level requirements only meet Criterion 2.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: SECTION 3.7 - PLANT SYSTEMS

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the LaSalle 1 and 2 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. 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 involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be permitted.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions

3. (continued)

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1434, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. 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 imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. 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 involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1434, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase 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 involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions) (continued)

3. Does this change involve a significant reduction in a margin of safety?

   Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relaxes the existing allowed outage time for a single RHRSW subsystem from 72 hours to 7 days. The proposed change does not increase the probability of an accident because it will not involve any physical changes to plant systems, structures, or components, or the manner in which these systems, structures, or components are operated, maintained, modified, tested, or inspected. The RHRSW System is not assumed to be an initiator of any analyzed event. The RHRSW System's function is to mitigate the consequences of analyzed events by supplying cooling water to the RHR heat exchangers during an accident. The change will not allow continuous operation when the RHRSW subsystem is inoperable. The proposed allowed outage time provides a reasonable amount of time to perform required maintenance and Surveillances, and restore the RHRSW subsystem to OPERABLE status in order to ensure its continued reliability. Furthermore, the probability of an event requiring the RHRSW subsystem to function during the 7 day period is low. The consequences of an event occurring during the proposed allowed outage time are the same as the consequences of an event occurring during the current 72 hour allowed outage time. Therefore, this change will 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

L.1 CHANGE (continued)

3. Does this change involve a significant reduction in a margin of safety?

The increased time allowed to restore an inoperable RHRSW subsystem is acceptable based on the low probability of an event requiring the RHRSW subsystem to function, the capabilities of the remaining OPERABLE RHRSW subsystem, and the desire to minimize plant shutdown transients. The proposed 96 hour extension will provide sufficient time to restore a RHRSW subsystem to OPERABLE status and thus, avoid an undesired plant shutdown transient. In addition, the RHRSW System is a support system to other systems that currently have 7 day out of service times when one subsystem is inoperable. Therefore, this change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change provides additional time to restore one RHRSW subsystem, when both subsystems are inoperable, prior to requiring a plant shutdown. The proposed change will not affect the probability of an accident. The RHRSW System is not assumed to be an initiator of any analyzed event. Allowing 8 additional hours to comply with the LCO will not affect the consequences of an accident. The chance of an event occurring while in this condition is remote. The consequences of an event occurring during the proposed 8 hour period are the same as those associated with an event occurring with the current action. The 8 hour allowed outage time provided to restore one RHRSW subsystem to OPERABLE status is consistent with the allowed outage time provided for restoration of both subsystems of RHR suppression pool cooling and both RHR suppression pool spray subsystems (systems supported by the RHRSW System in MODES 1, 2, and 3).

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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change provides additional time to restore one RHRSW subsystem, when both subsystems are inoperable, prior to requiring a plant shutdown. The Completion Time is also acceptable due to the low probability of a DBA occurring within this 8 hour period when both RHRSW subsystems are inoperable. While the OPERABILITY of the RHRSW System is implicitly assumed in the analysis assumptions, allowing 8 hours to restore one RHRSW subsystem to OPERABLE status does not significantly decrease the margin of safety. In addition, the added 8 hour time period provides the benefit of restoring compliance with the LCO instead of having to shut down the plant, potentially challenging plant systems. Therefore, this change does not involve a significant reduction in a margin of safety.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The phrase "actual or simulated," in reference to the start signal specified in CTS 4.7.1.2.b.1 and b.2, has been added to the system functional test Surveillance test description. This change does not impose a requirement to create an "actual or simulated" start signal, nor does it eliminate any restriction on producing an "actual or simulated" start signal. While creating an "actual" signal could increase the probability of an event, existing procedures (and the 10 CFR 50.59 control of revisions to them) dictate the acceptability of generating this signal. In addition, the use of a simulated signal to initiate the DGCW System yields the desired result in demonstrating DGCW System OPERABILITY. The proposed change does not affect the procedures governing plant operations or the acceptability of creating or simulating these start signals; it simply would allow such signals to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of initiation will not affect the acceptance criteria of the system functional test, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   Use of an actual or simulated start signal will not affect the performance or acceptance criteria of the Surveillance test. Operability is adequately demonstrated in either case since the system itself cannot discriminate between "actual" or "simulated" start signals. Therefore, the change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.3 - ULTIMATE HEAT SINK (UHS)

There were no plant specific less restrictive changes identified for this Specification.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The Control Room Area Filtration (CRAF) System is used to mitigate the consequences of an accident; however, the CRAF System is not considered in the initiation of any previously analyzed accident. As such, the proposed revision to the Applicability for the CRAF System during shutdown conditions will not increase the probability of any accident previously evaluated. In MODE 4 or 5, activities are conducted for which significant releases of radioactivity are postulated which require the CRAF System for mitigation of potential consequences. Therefore, the CRAF System is required to be OPERABLE in MODE 4 or 5, when activities are in progress which could, if an event occurs, result in significant releases of radioactivity (during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs). This change alters the current Applicability requirements to only include these activities. This is considered acceptable since the Technical Specification requires the CRAF System to be OPERABLE when it is required to mitigate postulated events in MODE 4 or 5. This change maintains situations for which significant releases of radioactivity are postulated while the plant is in MODE 4 or 5. In addition, the change to Applicability is consistent with the intent of current Technical Specification ACTIONS (in Mode 4 and 5 with two CRAF subsystems inoperable, the CTS ACTIONS require suspension of those activities for which significant releases of radioactivity are postulated). The proposed change still ensures the CRAF System is OPERABLE during conditions when radioactive releases are postulated. Therefore, the proposed change does not affect 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 involve a physical modification to the plant or a change in parameters governing normal plant operation. The proposed change still requires the CRAF System to be OPERABLE when it is required to perform its safety function. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

L.1 CHANGE (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change alters MODE 4 and 5 Applicability requirements for the CRAF System to include only those activities which could, if an event occurs, result in significant releases of radioactivity (i.e., during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs). This is considered acceptable since the Technical Specifications still require the CRAF System to be OPERABLE when it is required to mitigate postulated events in MODE 4 or 5. The ITS 3.7.4 Applicability maintains situations for which significant releases of radioactivity are postulated while the plant is in MODE 4 or 5. In addition, the change is consistent with the intent of CTS 3.7.2 ACTIONS (in Mode 4 and 5 with two CRAF subsystems inoperable, the CTS ACTIONS require suspension of those activities for which significant releases of radioactivity are postulated). The proposed change still ensures the CRAF System is OPERABLE during conditions when radioactive releases are postulated. In addition, this change provides additional scheduling flexibility during plant refueling outages by not requiring the CRAF System to be OPERABLE during operations that do not have a potential for a significant radioactive release. The proposed change does not impact any accident analysis assumptions. Thus, no question of safety is involved. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The Control Room Area Filtration (CRAF) System is used to mitigate the consequences of an accident, but is not considered as the initiator of any previously analyzed accident. As such, the inoperability of the system will not increase the probability of any accident previously evaluated. The proposed deletion of the current use of STAGGERED TEST BASIS for this system will not impact the system response to an accident. Therefore, this change does not involve any 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it 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 CRAF System continues to be determined in the same manner. Staggered testing does not have a significant effect on reliability, and does not impact the capability of the CRAF System to perform its safety function. Since the CRAF subsystems are independent and common cause failure is evaluated, the proposed change provides an equivalent assurance of the capability of the CRAF System to perform its safety function. The conduct of the test and the frequency of testing remain the same, but the schedule for conducting the test is no longer regulated.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The phrase "actual or," in reference to the actuation test signal has been added to the system functional test Surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures (and the 10 CFR 50.59 control of revisions to them) dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations nor the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of initiation will not affect the acceptance criteria of the system functional test, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   Use of an actual signal instead of the existing requirement, which limits use to a test signal, will not affect the performance or acceptance criteria of the Surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between "actual" or "test" signals. Therefore, the change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.5 - CONTROL ROOM AREA VENTILATION
AIR CONDITIONING (AC) SYSTEM

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.6 - MAIN CONDENSER OFFGAS

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The applicability of this specification has been changed to reflect the plant conditions for which the offgas activity has a potential of exceeding the values assumed in the analysis. In addition, alternative ACTIONS have been provided to leave the new applicability so that main steam is not contributing to the offgas activity. The main condenser offgas gross gamma activity rate limit is not assumed to be an initiator of any accident previously analyzed. The main condenser offgas gross gamma activity rate limit is an initial condition of the main condenser offgas system failure event; as such, it mitigates the consequences of an accident. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits of 10 CFR 100. With the main steam lines isolated or the SJAE not in operation, the offgas system is not being used to process the gross gamma activity; it is essentially maintained within the reactor coolant. Therefore, the event cannot occur. 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 changed does not introduce a new mode of offgas system operation and does not involve physical modification to the plant. Therefore, it 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 LCO continues to be required to be met when there is a potential of the event occurring and exceeding the offgas activity limits assumed in the analysis.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change will provide additional time to isolate the main steam lines or main condenser SJAE. The amount of time operating with the offgas activity release rate exceeding the limit with the main steam isolation valves open or SJAE operating is not considered as an initiator for any accidents previously analyzed. The additional 6 hours to isolated the MSIVs or SJAE provides a reasonable amount of time to perform an orderly closure of the required valves (which requires entry into MODE 2). The consequences of an event occurring while the unit is reducing power in order to isolate the MSIVs or SJAE during the extra 6 hours will be similar to the consequences of an event occurring at power. However, since offgas activity is expected to be reduced as power is lowered, a reduction in power will tend to minimize the consequences. 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The increased time allowed for isolating the main steam lines or SJAE with the offgas activity release rate exceeding the limit is acceptable based on the small probability of an event requiring the activity to be within the limit, the ability the isolate the main steam lines or SJAE manually if an event occurs, and the minimization of unit transients. The proposed 6 hour extension will allow the MSIVs or SJAE to be isolated in an orderly manner. As a result, the potential for human error and the risk associated with challenging unit systems will be reduced. Any reduction in a margin of safety will be insignificant and offset by the benefit gained from avoiding potential unit transients. Therefore, this change does not involve a significant reduction in a margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change provides and alternative method to place the plant in a condition outside the Applicability of the Specification. ITS 3.7.6 Required Actions B.3.1 and B.3.2 will require the plant to be in MODE 3 in 12 hours and MODE 4 in 36 hours instead of requiring the main steam isolation valves to be closed within 6 hours. The method of placing the plant outside the Applicability of the Specification and the Completion Times do not impact the initiation of any previously analyzed accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated. This Specification is not required in MODE 4 since main steam is not being exhausted to the main condenser, therefore the assumptions of a Main Condenser Offgas System failure event will still be bounded by the current analyses when MODE 4 is achieved. The consequences of an event occurring while the unit is reducing power will be similar to the consequences of an event occurring at power. However, since offgas activity is expected to be reduced as power is lowered, a reduction in power will tend to minimize the consequences. The Completion Times are acceptable, based on operating experience, to reach the required plant conditions from full power conditions in a orderly manner and without challenging plant systems. Therefore, this change to the Required Actions and Completion Times 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 introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it 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 provides an alternative method to place the plant in a condition outside the Applicability of the Specification. ITS 3.7.6 Required Actions B.3.1 and B.3.2 will require the plant to be in MODE 3 in 12 hours and MODE 4 in 36 hours instead of requiring the main steam isolation valves to be closed within 6 hours. This Specification is not required in MODE 4 since main steam is not being exhausted to the

L.3 CHANGE

3. (continued)

main condenser, therefore the assumptions of a Main Condenser Offgas System failure event will still be bounded by the current analyses. The proposed alternative action may help avoid a plant transient caused by isolating the main steam isolation valves in the 6 hour period. The Completion Times are acceptable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. As such these changes do not involve a significant reduction in a margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change would allow 31 days to perform the Surveillance after placing the SJAE in operation with one or main steam lines not isolated. The Frequency of performing this Surveillance does not impact the initiation of any previously analyzed accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated. Additionally, the proposed Surveillance Requirement is still considered to be adequate to ensure the main condenser offgas release rate is maintained within limits. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The change will not result in a reduction in a margin of safety since the main condenser offgas release rate is still required to be within limits. The change would allow 31 days to perform the Surveillance, determination of main condenser offgas release rate, after placing the SJAE in operation with one or main steam lines not isolated. This determination is only meaningful with one or more main steam lines not isolated and the SJAE in operation. Only in this condition can radioactive gases be in the Main Condenser Offgas System at significant rates. The 31 day period is an acceptable time to establish conditions appropriate for data collection and evaluation and is considered acceptable given the availability of instrumentation to monitor the offgas activity release rate. Therefore, the proposed requirements will continue to provide the necessary assurance that the main condenser offgas release rate is within limits.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.7 - MAIN TURBINE BYPASS SYSTEM

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change eliminates the requirement to restore the Main Turbine Bypass System to an OPERABLE status following the implementation of the applicable MINIMUM CRITICAL POWER RATIO (MCPR) penalty. The Main Turbine Bypass System's role is in mitigating the design basis transients, thereby limiting the consequences of violating the MCPR Safety Limit. The Main Turbine Bypass System and the MCPR are not assumed to be initiators of any analyzed event. Maintaining the MCPR within the established limit will also ensure that the consequences of design basis transients are mitigated. Analyses have been performed assuming the Main Turbine Bypass System is out of service (i.e., all five bypass valves are inoperable). These analyses confirmed that continued plant operation with the Main Turbine Bypass System out of service was acceptable with the application of a specific cycle-dependent MCPR value for the inoperable Main Turbine Bypass System. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any design changes, plant modifications, or changes in plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change eliminates the requirement to restore the Main Turbine Bypass System to an OPERABLE status following the implementation of the applicable MINIMUM CRITICAL POWER RATIO (MCPR) penalty. Analyses have been performed assuming the Main Turbine Bypass System is out of service (i.e., all five bypass valves are inoperable). These analyses confirmed that continued plant operation with the Main Turbine Bypass System out of service was acceptable with the application of a specific cycle-dependent MCPR value for the inoperable Main Turbine Bypass System. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.7.4 - SEALED SOURCE CONTAMINATION

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.7.7 - AREA TEMPERATURE MONITORING

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.7.8 - STRUCTURAL INTEGRITY OF CLASS 1 STRUCTURES

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.7.9 - SNUBBERS

There were no plant specific less restrictive changes identified for this Specification.

ENVIRONMENTAL ASSESSMENT
ITS: SECTION 3.7 - PLANT SYSTEMS

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

1. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

2. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.