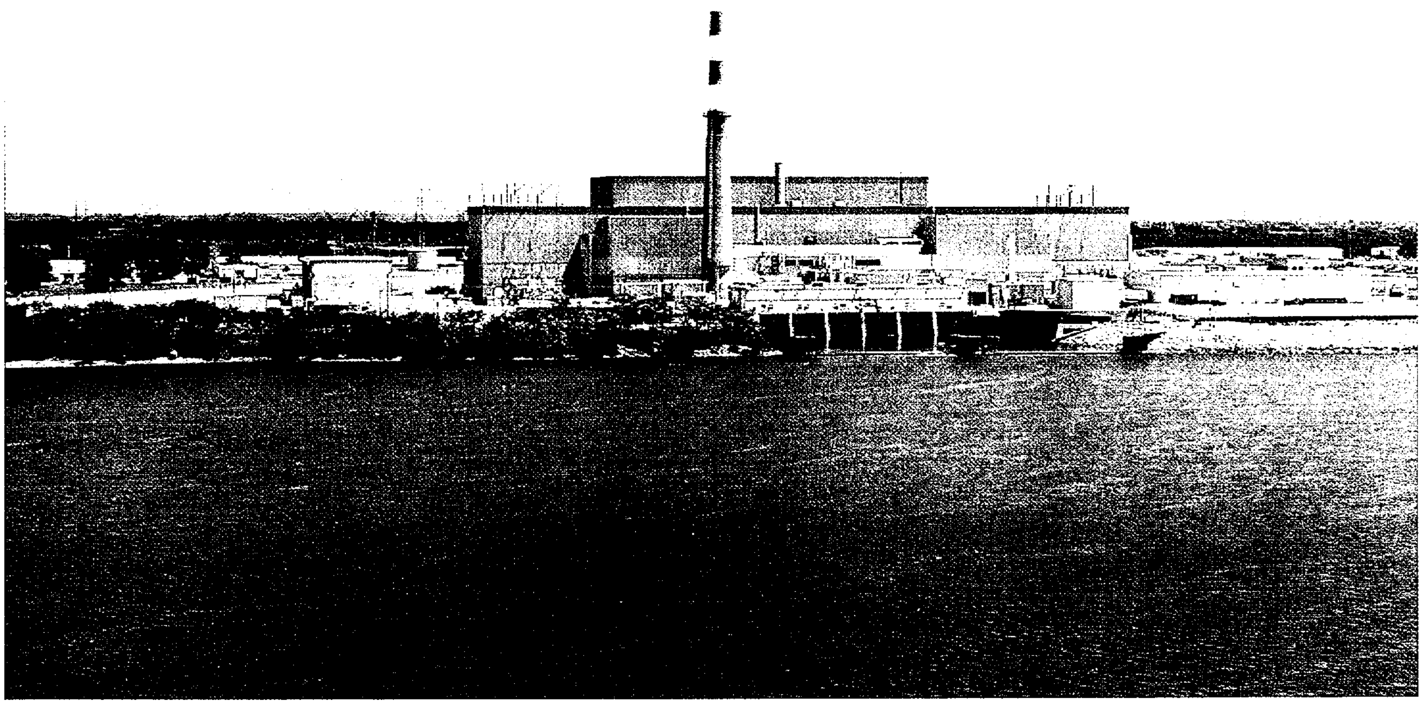
Improved Technical Specifications



Quad Cities 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 pump inoperable.	A.1	Restore RHRSW pump to OPERABLE status.	30 days
B. One RHRSW pump in each subsystem inoperable.	B.1	Restore one RHRSW pump to OPERABLE status.	7 days
C. One RHRSW subsystem inoperable for reasons other than Condition A.	C.1	--------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "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
D. Both RHRSW subsystems inoperable for reasons other than Condition B.	D.1	--------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling subsystems made inoperable by RHRSW System. --------------------- Restore one RHRSW subsystem to OPERABLE status.	8 hours
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	12 hours
	AND		
	E.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual and power operated 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. Two unit DGCW subsystems; and

b. The opposite unit DGCW subsystem capable of supporting its associated diesel generator (DG).

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 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 when its associated DG starts.	24 months

3.7 PLANT SYSTEMS

3.7.3 Ultimate Heat Sink (UHS)

LCO 3.7.3 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the water level in the intake bay is $\geq$ 568 ft mean sea level.	24 hours
SR 3.7.3.2	Verify the average water temperature of UHS is $\leq$ 95°F.	24 hours

3.7 PLANT SYSTEMS

3.7.4 Control Room Emergency Ventilation (CREV) System

LCO 3.7.4 The CREV System 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. CREV System inoperable in MODE 1, 2, or 3.	A.1 Restore CREV System 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. AND B.2 Be in MODE 4.	12 hours 36 hours
C. CREV System 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. ---------------------------- C.1 Suspend movement of irradiated fuel assemblies in the secondary containment. AND	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Suspend CORE ALTERATIONS. AND C.3 Initiate action to suspend OPDRVs.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Operate the CREV System for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.4.2	Perform required CREV filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.4.3	Verify the CREV System isolation dampers close on an actual or simulated initiation signal.	24 months.
SR 3.7.4.4	Verify the CREV System can maintain a positive pressure of ≥ 0.125 inches water gauge relative to the adjacent areas during the pressurization mode of operation at a flow rate of ≤ 2000 scfm.	24 months

3.7 PLANT SYSTEMS

3.7.5 Control Room Emergency Ventilation Air Conditioning (AC) System

LCO 3.7.5 The Control Room Emergency Ventilation AC System 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. Control Room Emergency Ventilation AC System inoperable in MODE 1, 2, or 3.	A.1 Restore Control Room Emergency Ventilation AC System 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. Control Room Emergency Ventilation AC System 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. ----------------------------	
	C.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify the Control Room Emergency Ventilation AC System has the capability to remove the assumed heat load.	24 months

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 offgas holdup line shall be $\leq$ 251,100 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 ≤ 251,100 μCi/second after decay of 30 minutes.	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

3.7 PLANT SYSTEMS

3.7.7 The 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 ≥ 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.	92 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 $\geq$ 19 ft 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 $\geq$ 19 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days

3.7 PLANT SYSTEMS

3.7.9 Safe Shutdown Makeup Pump (SSMP) System

LCO 3.7.9 The SSMP System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SSMP System inoperable.	A.1 Restore SSMP System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify each SSMP System 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 3.7.9.2	Verify SSMP System pump develops a flow rate $\geq$ 400 gpm against a system head corresponding to reactor pressure > 1120 psig.	92 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 consists of two independent and redundant subsystems. Each subsystem is made up of a header, two pumps, a suction source, valves, piping, heat exchanger, and associated instrumentation. Each pump can provide sufficient flow to the heat exchanger (3500 gpm) and all auxiliary loads. 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 valves located on the RHR heat exchanger discharge header, 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 Ultimate Heat Sink (UHS) consists of the Mississippi River, the intake flume, the crib house, and the discharge structure and flume. The UHS is described in the UFSAR Section 9.2.5, Reference 2. Cooling water is pumped by the RHRSW pumps via the suction pipes which begin in the crib house through the tube side of the RHR heat exchangers, and discharges to the discharge flume.

The system is initiated manually from the control room. If operating and a loss of coolant accident (LOCA) occurs, 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 the LOCA signal is manually overridden or clears and adequate electrical power is available.

(continued)

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 containment is discussed in UFSAR, Section 6.2 (Ref. 3). 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 6.2.1.3.3 (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 3500 gpm to the associated heat exchanger with one pump operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are 177°F and 28 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

The RHRSW System satisfies Criterion 3 of 10 CFR 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)

BASES

LCO (continued)

b. An OPERABLE flow path is capable of taking suction from the UHS and transferring the water to the RHR heat exchanger and separately to the associated safety related equipment at the assumed flow rate. Additionally, the RHRSW discharge header motor operated valves must be closed so that failure of one subsystem will not affect the OPERABILITY of the other subsystem.

An adequate suction source is not addressed in this LCO since the minimum net positive suction head and 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 3.4.7, "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 3.4.8, "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.

ACTIONS

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

(continued)

BASES

ACTIONS

A.1 (continued)

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 and the low probability of a DBA with concurrent worst case single failure.

B.1

With one RHRSW pump inoperable in each subsystem, the remaining OPERABLE pump in each subsystem can provide adequate heat removal capacity following a design basis LOCA with concurrent worst case single failure. 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 and low probability of an event occurring requiring RHRSW during this time period.

C.1

Required Action 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.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7, be entered and Required Actions taken if the inoperable RHRSW subsystem results in an inoperable RHR shutdown cooling subsystem. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

D.1

With both RHRSW subsystems inoperable for reasons other than Condition B (e.g., both subsystems with inoperable 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 3.4.7, be entered and Required Actions taken if an inoperable RHRSW subsystem results in an inoperable RHR shutdown cooling subsystem. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

E.1 and E.2

If any Required Action and associated Completion Time of Conditions A, B, C, or D 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 and power operated 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.1.1 (continued) 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, Section 9.2.5. 3. UFSAR, Section 6.2. 4. UFSAR, Section 6.2.1.3.3.

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 diesel generator (DG) heat exchangers and the Emergency Core Cooling System (ECCS) room emergency coolers. Each unit DGCW subsystem provides cooling water to its associated DG and the unit ECCS room emergency coolers. The DG 1/2 DGCW subsystem may be manually aligned to provide cooling to either unit's ECCS room emergency coolers.

The DGCW pump autostarts upon receipt of a DG start signal when power is available to the pump's electrical bus. Cooling water is pumped from the suction header of Residual Heat Removal Service Water (RHRSW) System by the DGCW pump to the associated DG heat exchangers. After removing heat from the heat exchangers, the water is discharged to the discharge flume. The DGCW subsystem associated with DG 1 (DG 2) is also normally aligned to provide cooling water to the unit ECCS room emergency coolers. However, the DGCW subsystem associated with DG 1/2 can be aligned as an alternate source of cooling water to the Unit 1 or Unit 2 ECCS room emergency coolers. The DGCW subsystem associated with DG 1 can be aligned as an alternate source of cooling water to the DG 1/2 heat exchanger. A complete description of the DGCW System is presented in the UFSAR, Section 9.5.5 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ability of the DGCW System to provide adequate cooling to the DG heat exchangers and ECCS room emergency coolers is an implicit assumption for the safety analyses presented in the 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).

(continued)

BASES (continued)

LCO

The OPERABILITY of the unit DGCW System is required to provide a coolant source to ensure effective operation of the DGs and ECCS in the event of an accident or transient. The OPERABILITY of the DGCW System is based on having an OPERABLE pump and an OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring cooling water to the associated DG heat exchangers and ECCS room emergency coolers. The OPERABILITY of the opposite unit's DGCW subsystem is required to provide adequate cooling to ensure effective operation of the required opposite unit's DG heat exchanger in the event of an accident in order to support operation of the shared systems such as the Standby Gas Treatment System and Control Room Emergency Ventilation System.

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 be OPERABLE 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.

(continued)

BASES

ACTIONS (continued)

A.1

If one or more DGCW subsystems are inoperable, the associated DG(s) and ECCS components, supported by the affected ECCS room emergency coolers, cannot perform 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 System (ECCS)—Operating," as applicable.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

Verifying the correct alignment for manual valves in the DGCW subsystem flow paths 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. 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 heat exchangers and ECCS room emergency coolers when the DG starts.

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.

(continued)

BASES (continued)

REFERENCES	1.	UFSAR, Section 9.5.5.
	2.	UFSAR, Chapter 6.
	3.	UFSAR, Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.3 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The Residual Heat Removal Service Water (RHRSW) and the Diesel Generator Cooling Water (DGCW) Systems are designed to provide cooling water to components required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is described in UFSAR, Section 9.2.1 (Ref. 1) while the DGCW System is described in UFSAR, Section 9.5.5 (Ref. 2). These systems are also described in the Bases for LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System," and LCO 3.7.2, "Diesel Generator Cooling Water (DGCW) System." The UHS provides a suction source and discharge pathway for the cooling water associated with these systems. The UHS is described in UFSAR, Section 9.2.5 (Ref. 3).

The Mississippi River provides an UHS with sufficient cooling capacity to either provide normal cooldown of the units, or mitigate the effects of accident conditions within acceptable limits for one unit while conducting a normal cooldown of the other unit. The water flows under a floating boom to the intake flume and into the intake bay of the crib house, where it is directed to various plant systems. There are seven bays within the crib house, one associated with each of the six circulating water pumps and another which houses the 1/2 B diesel-driven fire pump. The bay housing the 1/2 B diesel-driven fire pump receives its water from two of the bays associated with the circulating water pumps (bays 1A and 2C). This bay also supplies water to a suction header for each RHRSW subsystem (2 for each unit). Each DGCW subsystem also obtains a suction from one of these headers. The DGCW subsystem associated with DG 1/2 obtains a suction from one of the Unit 1 RHRSW suction headers. The RHRSW and DGCW Systems for both units can receive a sufficient amount of water from either bay 1A or 2C. The UHS also contains a discharge flume where the water is returned to the Mississippi River. A weir wall in the discharge flume maintains a minimum level in the discharge bay to ensure flow is directed to the river.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Sufficient water inventory is available for the RHRSW and the DGCW Systems post LOCA cooling requirements. This water source is provided by the UHS. The ability of the UHS to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the UFSAR, Section 6.2 (Ref. 4). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The ability of the UHS to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in Reference 4. The ability to provide onsite emergency AC power is dependent on the ability of the UHS to cool the DGs. The long term cooling capability of the RHR, core spray, DGCW, and RHR service water pumps is also dependent on the cooling provided by the UHS System.

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

LCO

The OPERABILITY of the UHS is based on having a minimum water level in the intake bay of 568 ft mean sea level and a maximum water temperature of 95°F.

APPLICABILITY

In MODES 1, 2, and 3, the UHS is required to be OPERABLE to support OPERABILITY of the equipment serviced by the RHRSW and DGCW Systems. Therefore, the UHS 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.

ACTIONS

<u>A.1 and A.2</u>

If the UHS is determined inoperable the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 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 3.7.3.1

This SR verifies the water level in the intake bay to be sufficient for the proper operation of the RHRSW and DGCW 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 3.7.3.2

Verification of the UHS temperature ensures that the heat removal capabilities of the RHRSW and DGCW Systems 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.

REFERENCES

1. UFSAR, Section 9.2.1.

2. UFSAR, Section 9.5.5.

3. UFSAR, Section 9.2.5.

4. UFSAR, Section 6.2.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Emergency Ventilation (CREV) System

BASES

BACKGROUND

The CREV System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA). The control room emergency zone served by the CREV System consists of the main control room, cable spreading room, auxiliary electric equipment room, computer room, and the Train B Heating Ventilation and Air Conditioning (HVAC) equipment enclosure.

The safety related function of the CREV System consists of a single high efficiency air filtration train for emergency treatment of outside supply air. The filter train consists of an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, two 100% capacity booster fans in parallel, the Train B air handling unit (excluding the refrigeration condensing unit), and the associated ductwork and dampers. The electric heater is used to limit the relative humidity of the air entering the filter train. Prefilters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The CREV System is a standby system, parts of which also operate during normal unit operations to maintain the control room emergency zone environment. Upon receipt of an isolation signal (indicative of conditions that could result in radiation exposure to control room emergency zone personnel), the control room emergency zone is automatically isolated to minimize infiltration of contaminated air into the control room emergency zone. A system of dampers isolates the control room emergency zone, and the air is recirculated. Operator action is required within one hour after an accident to verify isolation and activate the air filtration unit (AFU) of the CREV System to pressurize the control room emergency zone. Outside air is taken in at the outside air ventilation intake through the AFU for removal of airborne radioactive particles and is mixed with the recirculated air.

(continued)

BASES

BACKGROUND (continued)

The CREV System is designed to maintain the control room emergency zone environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. The CREV System will pressurize the control room emergency zone to about 0.125 inches water gauge to minimize infiltration of air from adjacent zones. CREV System operation in maintaining control room habitability is discussed in the UFSAR, Sections 6.4, 9.4, and 15.6.5 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CREV System to maintain the habitability of the control room emergency zone is an explicit assumption for the safety analyses presented in the UFSAR, Sections 6.4 and 15.6.5 (Refs. 1 and 3, respectively). The isolation of the control room emergency zone is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the UFSAR, Section 6.4 (Ref. 1). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 3.

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

LCO

The CREV System is required to be OPERABLE. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

The CREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE. The system is considered OPERABLE when its associated:

a. AFU is OPERABLE,

b. Train B air handling unit (fan portion only) is OPERABLE, including the ductwork, to maintain air circulation to and from the control room emergency zone; and

c. Outside air ventilation intake is OPERABLE.

(continued)

BASES

LCO (continued)

The AFU is considered OPERABLE when a booster fan is OPERABLE; HEPA filter and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and heater, ductwork, valves, and dampers are OPERABLE, and air circulation through the filter train can be maintained.

In addition, the control room emergency zone boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that the pressurization limit of SR 3.7.4.4 can be met. However, it is acceptable for access doors to be open for normal control room emergency zone entry and exit and not consider it to be a failure to meet the LCO.

APPLICABILITY

In MODES 1, 2, and 3, the CREV 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 because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREV 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 potential for draining the reactor vessel (OPDRVs).

ACTIONS

A.1

With the CREV System inoperable in MODE 1, 2, or 3, the inoperable CREV System must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CREV System 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, and C.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel 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.

With the CREV System inoperable, during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, 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. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the CREV System in a standby mode starts from the control room and continues to operate. This SR includes initiating flow through the HEPA filters and charcoal adsorbers. 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 the system once every month provides an adequate check on this system. Monthly heater operation for $\geq$ 10 continuous hours, during system operation dries out any moisture that has accumulated in the charcoal as a result of humidity in the ambient air. Furthermore, the 31 day Frequency is based on the known reliability of the equipment.

SR 3.7.4.2

This SR verifies that the required CREV testing is performed in accordance with Specification 5.5.7, "Ventilation Filter Testing Program (VFTP)." The CREV filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). 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 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, the CREV System isolation dampers close. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.6 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)	<u>SR 3.7.4.4</u> This SR verifies the integrity of the control room emergency zone and the assumed inleakage rates of potentially contaminated air. The control room emergency zone positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CREV System. During the emergency pressurization mode of operation, the CREV System is designed to slightly pressurize the control room emergency zone $\geq$ 0.125 inches water gauge positive pressure with respect to the adjacent areas to minimize unfiltered inleakage. The CREV System is designed to maintain this positive pressure at a flow rate of $\leq$ 2000 scfm to the control room emergency zone in the pressurization mode. The Frequency of 24 months is consistent with industry practice and other filtration systems SRs.
REFERENCES	1. UFSAR, Section 6.4. 2. UFSAR, Section 9.4. 3. UFSAR, Section 15.6.5. 4. Regulatory Guide 1.52, Revision 2, March 1978.

B 3.7 PLANT SYSTEMS

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

BASES

BACKGROUND

The Control Room Emergency Ventilation AC portion of the control room area Heating, Ventilation, and Air Conditioning (HVAC) System (hereafter referred to as the Control Room Emergency Ventilation AC System) provides temperature control for the control room emergency zone following isolation of the control room emergency zone.

The Control Room Emergency Ventilation AC System is a single zone system that services only those rooms that are a part of the control room emergency zone. The system provides cooling of the recirculated and outside air makeup for the control room emergency zone. The Control Room Emergency Ventilation AC System, addressed by this Specification, consists of the Train B air handling unit (AHU), ductwork, dampers, refrigeration condensing unit, and instrumentation and controls to provide for control room emergency zone temperature control.

The Control Room Emergency Ventilation AC System is designed to provide a controlled environment under both normal and accident conditions. The system provides the required temperature control to maintain a suitable control room emergency zone environment for a sustained occupancy of 10 persons. The design conditions for the habitability of the control room emergency zone environment are 70°F to 80°F and 50% relative humidity. The Control Room Emergency Ventilation AC System operation in maintaining the control room emergency zone temperature is discussed in the UFSAR, Section 6.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room Emergency Ventilation AC System is to maintain the control room emergency zone temperature for a 30 day continuous occupancy following isolation of the control room emergency zone.

During emergency operation, the Control Room Emergency Ventilation AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room emergency zone. The safety related Control Room Emergency Ventilation AC System (Train B HVAC) is powered from diesel

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

generator supported switchgear. Train B Control Room HVAC is normally in the standby condition and is used for accident mitigation. Train A Control Room HVAC is nonsafety related and is in operation during normal conditions. The Train B refrigeration condensing unit, normally served by the Service Water System, can be provided with cooling water from either the Unit 1 or 2 Residual Heat Removal Service Water (RHRSW) System. The Control Room Emergency Ventilation AC System is designed in accordance with Seismic Category I requirements, except for a portion of the return ductwork. The Control Room Emergency Ventilation AC System is capable of removing sensible and latent heat loads from the control room emergency zone, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The Control Room Emergency Ventilation AC System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The Control Room Emergency Ventilation AC System is required to be OPERABLE. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room Emergency Ventilation AC System is considered OPERABLE when the individual components necessary to maintain the control room emergency zone temperature are OPERABLE. These components include the cooling coils, fans, chillers, compressors, ductwork, dampers, and associated instrumentation and controls. In addition, during conditions in MODES other than MODES 1, 2, and 3 when the Control Room Emergency Ventilation AC System is required to be OPERABLE (e.g., during CORE ALTERATIONS), the necessary portions of the RHRSW System and Ultimate Heat Sink capable of providing cooling to the refrigeration condensing unit are part of the OPERABILITY requirements covered by this LCO.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room Emergency Ventilation AC System must be OPERABLE to ensure that the control room emergency zone temperature will not exceed equipment OPERABILITY limits following control room emergency zone isolation.

(continued)

BASES

APPLICABILITY (continued)

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 Emergency 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

A.1

With the Control Room Emergency Ventilation AC System inoperable in MODE 1, 2, or 3, the system must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on the low probability of an event occurring requiring control room emergency zone isolation and the availability of alternate nonsafety cooling methods.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable Control Room Emergency Ventilation AC System 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, and C.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel 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.

With the Control Room Emergency Ventilation AC System inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, 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. 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 verifies that the heat removal capability of the system is sufficient to remove the control room emergency zone heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room Emergency Ventilation AC System is not expected over this time period.

(continued)

BASES (continued)

REFERENCES	1. UFSAR, Section 6.4.

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 moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture 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, discussed in Reference 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 established consistent with this requirement (2511 MWt x 100 μCi/MWt-second = 251,100 μCi/second).

(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, 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.

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

If the gross gamma activity rate is not restored to within the limits in 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 at the recombiner outlet or the SJAE outlet if the recombiner is bypassed) to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85M, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly as indicated by the radiation monitors located prior to the offgas holdup line (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.

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. Letter E-DAS-023-00 from D. A. Studley (Scientech-NUS) to R. Tsai (ComEd), dated January 24, 2000.

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 40% 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 nine valve manifold connected to the main steam lines between the main steam isolation valves and the main turbine stop valves. Each of the nine valves is 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 7.7.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct 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 sequentially. When the bypass valves open, the steam flows from the main steam equalizing header to the bypass manifold through the bypass valve, to its bypass line, where an orifice further reduces the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the turbine trip, turbine generator load rejection and feedwater controller failure transients, as discussed in the UFSAR, Sections 15.2.3.2, 15.2.2.2, and 15.1.2 (Refs. 2, 3, and 4, 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, so 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 this LCO to be met. The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. 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 analyses (Refs. 2, 3, and 4).

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 generator load rejection, turbine trip, and feedwater controller failure transients. As discussed in the Bases for LCO 3.2.2, sufficient margin to these limits exists at < 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 generator load rejection, turbine trip, and feedwater controller failure 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 92 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 92 day Frequency. 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 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. 5). 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 7.7.4. 2. UFSAR, Section 15.2.3.2. 3. UFSAR, Section 15.2.2.2. 4. UFSAR, Section 15.1.2. 5. 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 Reference 2.
APPLICABLE SAFETY ANALYSES	The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. 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% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800 (Refs. 4 and 5) and less than the 10 CFR 50, Appendix A, GDC 19 limits (Ref. 6). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Regulatory Guide 1.25 (Ref. 7). The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. 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).
LCO	The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies 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 MODE 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, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is 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.

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 all water level changes are controlled by unit procedures.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.
2. Letter E-DAS-00-048 from D.A. Studley (Scientech) to Robert Tsai (ComEd), "Submittal of Calculation in Support of Improved Tech. Spec. Program," dated February 17, 2000.
3. 10 CFR 100.
4. NUREG-0800, Section 15.7.4, Revision 1, July 1981.
5. NUREG-0800, Section 6.4, Revision 2, July 1981.
6. 10 CFR 50, Appendix A, GDC 19.
7. Regulatory Guide 1.25, March 1972.

B 3.7 PLANT SYSTEMS

B 3.7.9 Safe Shutdown Makeup Pump (SSMP) System

BASES

BACKGROUND

The SSMP System is designed to operate manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the Feedwater System to provide makeup water to the RPV. Under these conditions, the High Pressure Coolant Injection (HPCI), the Reactor Core Isolation Cooling (RCIC) and the SSMP Systems perform similar functions. The SSMP System design requirements ensure that the criteria of 10 CFR 50, Appendix R, Section III.G (Ref. 1) are satisfied.

The SSMP System (Ref. 2) consists of a motor driven pump unit, as well as piping and valves to transfer water from the suction source to the RPV through the Feedwater System line via the HPCI System line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the contaminated condensate storage tanks (CCSTs). An alternate source of makeup water is available from the Fire Protection System header in the turbine building.

The SSMP System is designed to provide makeup water for a wide range of reactor pressures, 150 psig to 1120 psig. The SSMP System injection valves are interlocked to allow injection into only one RPV at a time since the system is common to Units 1 and 2. Electric power for the system is normally fed from Division 2 of Unit 1, however an alternate source is available from Division 2 of Unit 2.

The SSMP System does not include a minimum flow line therefore a trip will occur if the flow control valve closes. This will prevent pump damage due to overheating in low flow conditions. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the SSMP System discharge piping is kept full of water. The SSMP System is normally aligned to the CCST. The height of water in the CCST is sufficient to maintain the piping full of water up to the unit injection valves. The feedwater header pressure ensures the remaining portion of the SSMP System discharge line is full of water. Therefore, the SSMP System does not require a "keep fill" system.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The function of the SSMP System is to respond to transient events by providing makeup coolant to the reactor. The SSMP System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for SSMP System operation. The system provides a backup to the Unit 1 and 2 RCIC Systems to satisfy the requirements of criteria of 10 CFR 50, Appendix R, Section III.G (Ref. 1). Based on its contribution to the reduction of overall plant risk, the system satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii) and is therefore included in the Technical Specifications.

LCO

The OPERABILITY of the SSMP System ensures sufficient reactor water makeup is provided in the event of RPV isolation accompanied by a loss of feedwater flow. The SSMP System has sufficient capacity for maintaining RPV inventory during an isolation event.

APPLICABILITY

The SSMP System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since the SSMP System provides a non-Emergency Core Cooling System water source for makeup when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, the SSMP System is not required to be OPERABLE since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the RPV and since the plant risk associated with fire is also reduced during these MODES.

ACTIONS

A.1 and A.2

If the SSMP System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, the SSMP System must be restored to OPERABLE status within 14 days. In this Condition, loss of the SSMP System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the RCIC and HPCI System are required to be OPERABLE. The 14 day Completion Time is consistent with the Completion Time for a RCIC System inoperability, because of similar functions of the RCIC and SSMP Systems. The same Completion Time for RCIC is also applied to the SSMP System since the SSMP System and the RCIC System have the same post-fire shutdown functionality goals to provide reactor water makeup (Ref. 3).

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the SSMP System cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to $\leq$ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SSMP System flow path provides assurance that the proper flow path will exist for SSMP System 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. 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. For the SSMP System, this SR also includes the flow controller position since it controls the pump discharge flow control valve position.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the SSMP System. This Frequency has been shown to be acceptable through operating experience.

SR 3.7.9.2

The SSMP System pump flow rate ensures that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow test is performed by utilizing the full flow test line to the CCST.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.9.2 (continued) The requirements include verifying that the pump discharge pressure is greater than or equal to a pressure that would produce the desired injection flow including allowances for the flow and elevation head losses of the injection line. This provides adequate assurance of SSMP System OPERABILITY based on performance at nominal conditions. A 92 day Frequency for SR 3.7.9.2 is consistent with the Inservice Testing Program requirements.
REFERENCES	1. 10 CFR 50, Appendix R, Section III.G. 2. UFSAR, Section 5.4.6.5. 3. Letter from J.A. Grobe (NRC) to O.D. Kingsley (ComEd), "NRC Inspection Report 50-254/98011 (DRS); 50-265/98011 (DRS)," dated July 2, 1998.

3.8 - LIMITING CONDITIONS FOR OPERATION

A. Residual Heat Removal Service Water System

LCO 3.7.1 At least the following independent residual heat removal service water (RHRSW) subsystems, ~~with each subsystem comprised of:~~

~~1. Two OPERABLE RHRSW pumps, and~~

~~2. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water:~~

~~a. Through one RHR heat exchanger, and separately,~~

~~b. To the associated safety related equipment,~~

LA.1

shall be OPERABLE:

1. In OPERATIONAL MODE(s) 1, 2 and 3, two subsystems.

~~2. In OPERATIONAL MODE(s) 4, 5 and * the subsystem(s) associated with subsystems/loops and components required OPERABLE by Specifications 3.6.O, 3.6.P, 3.8.D, 3.10.K and 3.10.L.~~

LA.2

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, ~~4, 5 and *.~~ LA.2

4.8 - SURVEILLANCE REQUIREMENTS

A. Residual Heat Removal Service Water System

SR 3.7.1.1 Each of the required RHRSW subsystems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve, manual or power operated, 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.2

LA.2

~~* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.~~

ACTION:

1. In OPERATIONAL MODE 1, 2 or 3:

ACTION A

a. With one RHRSW pump inoperable, restore the inoperable pump to OPERABLE status within 30 days

ACTION E

or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION B

b. With one RHRSW pump in each subsystem inoperable, restore at least one inoperable pump to OPERABLE status within 7 days

ACTION E

or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION C

c. With one RHRSW subsystem otherwise inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump within 7 days

ACTION E

or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION D

d. With both RHRSW subsystems otherwise inoperable, restore at least one subsystem to OPERABLE status within 8 hours

ACTION E

or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN[a] within the following 24 hours.

A.3

a 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.

3.8 - LIMITING CONDITIONS FOR OPERATION | 4.8 - SURVEILLANCE REQUIREMENTS

Note to Required Action C.1 and D.1

2. In OPERATIONAL MODE 3 or 4 with the RHRSW subsystem which is associated with an RHR subsystem required OPERABLE by Specification 3.6.O or 3.6.P inoperable, declare the associated RHR subsystem inoperable and take the ACTION required by Specification 3.6.O or 3.6.P, as applicable. LA.2

3. ~~In OPERATIONAL MODE 5 with the RHRSW subsystem which is associated with an RHR subsystem required OPERABLE by Specification 3.10.K or 3.10.L inoperable, declare the associated RHR subsystem inoperable and take the ACTION required by Specification 3.10.K or 3.10.L, as applicable.~~

4. ~~In OPERATIONAL MODE * with both unit RHRSW subsystem(s) inoperable, declare the control room emergency filtration system, Train B, inoperable and take the ACTION required by Specification 3.8.D.~~

LA.2

LA.2

* ~~When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.~~

DISCUSSION OF CHANGES
ITS: 3.7.1 - RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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 CTS 4.8.A 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 a 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 4.8.A (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.

A.3 The CTS 3.8.A Action1.d footnote (a) requirement that if unable to attain COLD SHUTDOWN when both RHRSW subsystems are inoperable, then maintain reactor coolant temperature as low as practical by use of alternate heat removal methods is deleted since it provides unnecessary duplication of the Actions, 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. The Action to be in MODE 4, which is modified by the footnote, adequately prescribes the requirement to make efforts to "maintain reactor coolant temperature as low as practical" (i.e., the duplicative requirement of the footnote). If conditions are such that MODE 4 cannot be attained, the Action remains in effect, essentially requiring efforts to reach MODE 4 to continue. Elimination of the footnote reflects an administrative presentation preference.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.8.A relating to system OPERABILITY, that the RHRSW subsystems shall have two RHRSW pumps capable of taking a suction from the ultimate heat sink and transferring the water through one RHR heat exchanger and separately to the associated safety related equipment, 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 3/4.8.A provides LCO requirements, Actions, and Surveillance Requirements for the RHRSW System when in MODES 4, 5, and when handling irradiated fuel in the secondary containment, during CORE ALTERATION(s) and operations with a potential for draining the reactor vessel. These requirements are proposed to be relocated to the Technical Requirements Manual (TRM). Since this system is a support system for other required 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 into the Quad Cities 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"

None

RELOCATED SPECIFICATIONS

None

3.8 - LIMITING CONDITIONS FOR OPERATION

B. Diesel Generator Cooling Water System

LCO 3.7.2 ~~A diesel generator cooling water (DGCW) subsystem shall be OPERABLE for each required diesel generator with each subsystem comprised of:~~

LA.1

1. ~~One OPERABLE DGCW pump, and~~
2. ~~An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring cooling water to the associated diesel generator.~~

APPLICABILITY:

LA.2 ~~When the diesel generator is required to be OPERABLE.~~

MODES 1, 2 and 3 M.1

ACTION:

ACTION A With one or more DGCW subsystems inoperable, declare the associated diesel generator inoperable and take the ACTION required by Specifications 3.9.A or 3.9.B, as applicable. M.1 A.3

4.8 - SURVEILLANCE REQUIREMENTS

B. Diesel Generator Cooling Water System

Each of the required DGCW subsystems shall be demonstrated OPERABLE:

A.4

SR 3.7.2.1 1. 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 positon.

SR 3.7.2.2 2. At least once per ~~18~~ 24 months by verifying that each pump starts automatically upon receipt of a start signal for the associated diesel generator. LD.1

M.1

The following DGCW subsytem shall be OPERABLE:

a. Two unit DGCW subsystems, and

b. The opposite unit DGCW subsystem capable of supporting its associated DG.

add proposed ACTIONS Note A.2

DISCUSSION OF CHANGES
ITS: 3.7.2 - DIESEL GENERATOR COOLING WATER (DGCW) SYSTEM

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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 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 3.8.B 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 LCO 3.8.1, "AC Sources - Operating." This is consistent with the intent of the CTS 3.8.B 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 3.8.B Action requires action to be taken per CTS 3.9.A or CTS 3.9.B 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 3.9.A (ITS 3.8.1) and CTS 3.9.B (ITS 3.8.2) adequately prescribe the necessary conditions for compliance without such references. Therefore, the existing reference to "take the ACTION required by Specifications 3.9.A or 3.9.B, as applicable" in the CTS 3.8.B Action serves no functional purpose, and its removal is purely an administrative difference in presentation.

A.4 CTS 4.8.B.1 requires a verification that each valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. This Surveillance is included as SR 3.7.2.1, however a clarification has been added to identify the types of valves in the system. Since all the valves in the flow path are manual valves the word "manual" has been added. Since this addition provides a clarification of the current requirement, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.8.B requires a DGCW subsystem to be OPERABLE for each required diesel generator. ITS 3.7.2 will require two unit DGCW subsystems and the opposite unit DGCW subsystem capable of supporting its associated diesel generator to be OPERABLE. The opposite unit requirements are necessary since safety related systems are shared between both units (e.g., Standby Gas Treatment System and Control Room Emergency Ventilation System) and powered from the opposite unit diesel generator. The opposite unit DGCW subsystem supports the OPERABILITY of these systems by cooling the associated diesel generator heat exchanger. The proposed change requiring two unit DGCW subsystems to be OPERABLE (in MODES 1, 2, and 3) is consistent with the current requirements (CTS 3.8.B in conjunction with CTS 3.9.A) and is considered administrative. However, the proposed change requiring the opposite unit's DGCW to also be OPERABLE represents an additional restriction on plant operation.

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 each of the ECCS room emergency unit coolers as well as the associated diesel generator. A commensurate change is also made to the CTS 3.8.B 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 3.8.B relating to system OPERABILITY (in this case that the DGCW subsystem shall have one OPERABLE DGCW pump, and an OPERABLE flow path capable of taking suction from the ultimate heat sink 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.

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) 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 3/4.8.B 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 Quad Cities 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.

LD.1 The Frequency for performing CTS 4.8.B.2 (proposed SR 3.7.2.2) has been extended from 18 months to 24 months. The DGCW System functional test, CTS 4.8.B.2 (proposed SR 3.7.2.2) ensures that a system start signal from the associated diesel generator will cause the system to operate as designed, by automatically starting the DGCW pump. The proposed change will allow the 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.B 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.B and proposed SR 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. Extending this Surveillance is acceptable in part because this requirement is also verified on a more frequent basis every 31 days when performing SR 3.8.1.2 during diesel generator start testing. 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 these tests normally pass their Surveillance at the current Frequency. An evaluation has been performed using this data, and

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. In addition, each of the DGCW pumps (required by this Specification) are tested according to the ASME Section XI inservice testing program to ensure that each subsystem can provide the proper flow against a specified test pressure. This test will detect significant failures of the DGCW subsystems to perform their safety function. Based on historical maintenance and surveillance data, the inherent system and component reliability, and the testing performed more frequently during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. 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"

None

RELOCATED SPECIFICATIONS

None

A.1

3.8 - LIMITING CONDITIONS FOR OPERATION

C. Ultimate Heat Sink

LCO 3.7.3 The ultimate heat sink shall be OPERABLE with:

SR 3.7.3.1 1. A minimum water level at or above elevation 568 ft Mean Sea Level, and

SR 3.7.3.2 2. An average water temperature of ≤95°F.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, ~~4, 5 and *~~. LA.1

ACTION:

With the requirements of the above specification not satisfied:

ACTION A 1. In OPERATIONAL MODE(s) 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

~~2. In OPERATIONAL MODE(s) 4 or 5 declare the RHRSW system and the diesel generator cooling water system inoperable and take the ACTION(s) required by Specifications 3.8.A and 3.8.B.~~

~~3. In OPERATIONAL MODE *, declare the diesel generator cooling water system inoperable and take the ACTION(s) required by Specification 3.8.B. The provisions of Specification 3.0.C are not applicable.~~

LA.1

4.8 - SURVEILLANCE REQUIREMENTS

C. Ultimate Heat Sink

SR 3.7.3.1 SR 3.7.3.2 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

LA.1

~~* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.~~

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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)).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 3/4.8.C provides LCO requirements, Actions, and Surveillance Requirements for the UHS when in MODES 4 and 5 and when handling irradiated fuel in the secondary containment, during CORE ALTERATIONS(s), and operations with a potential for draining the reactor vessel. These requirements are proposed to be relocated to the Technical Requirements Manual (TRM). Since the UHS supports the OPERABILITY of other required equipment with their own Specifications, the definition of OPERABILITY in ITS 1.1 will provide sufficient assurance the UHS can perform its required support function. In addition, the Bases for the supported systems will require the necessary portions of the UHS 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 into the Quad Cities 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"

None

RELOCATED SPECIFICATIONS

None

3.8 - LIMITING CONDITIONS FOR OPERATION

D. Control Room Emergency Ventilation System

LCO 3.7.4 The control room emergency ventilation system shall be OPERABLE~~, with the system comprised of an OPERABLE control room emergency filtration system and an OPERABLE refrigeration control unit (RCU).~~ LA.1

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, and *.

ACTION:

1. In OPERATIONAL MODE(s) 1, 2 or 3:

   ACTION A a. With the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or

   ACTION B be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With the refrigeration control unit (RCU) inoperable, restore the inoperable RCU to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

<See ITS 3.7.5>

4.8 - SURVEILLANCE REQUIREMENTS

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be demonstrated OPERABLE:

1. At least once per 18 months by verifying that the RCU has the capability to remove the required heat load. <See ITS 3.7.5>

SR 3.7.4.1 2. At least once per 31 days ~~by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and~~ verifying that the system operates for at least 10 hours with the heaters operating. LA.2

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

   a. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of <0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 scfm ±10%.

<See ITS 5.5>

A.2 Add proposed SR 3.7.4.2

Applicability * When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

ACTION C 2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

ACTION C NOTE 3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

<See ITS 3.7.5>

4.8 - SURVEILLANCE REQUIREMENTS

b. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

A.2

c. Verifying a system flow rate of 2000 scfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

4. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

LD.1

5. At least once per 18 months by: 24

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ± 10%.

A.2

<See ITS 5.5>

Applicability * When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION | 4.8 - SURVEILLANCE REQUIREMENTS

SR 3.7.4.3 b. Verifying that the isolation dampers close on ~~each of the following~~ signals: L.2

L.1 actuator

1) ~~Manual initiation from the control room, and~~

2) Simulated automatic isolation signal.

SR 3.7.4.4 c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

A.2

<See ITS 5.5>

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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 The filter testing requirements of CTS 4.8.D.3, 4.8.D.4, 4.8.D.5.a, 4.8.D.5.d, 4.8.D.6, and 4.8.D.7, are being moved to ITS 5.5.7 in accordance with the format of the BWR ISTS, NUREG-1433, Rev. 1. Any technical changes to this requirement will be addressed in the Discussion of Changes for ITS: 5.5. A Surveillance Requirement is added (proposed SR 3.7.4.2) to clarify that the tests of the Ventilation Filter Testing Program must also be completed and passed for determining OPERABILITY of the CREV System. Since this is a presentation preference that maintains current requirements, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 3.8.D specific details of what constitutes the Control Room Emergency Ventilation and Control Room Emergency Ventilation AC Systems are proposed to be relocated to the Bases. These are design details that are not necessary to be included in the Technical Specifications to ensure the OPERABILITY of the CREV System since the OPERABILITY requirements are adequately addressed in ITS 3.7.4 and ITS 3.7.5, "Control Room Emergency Ventilation Air Conditioning (AC) System." 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The CTS 4.8.D.2 details of the methods for performing the CREV System 31 day operating Surveillance (by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers) are proposed to be relocated to the Bases. These details are not necessary to ensure the OPERABILITY of the CREV System. The requirements of ITS 3.7.4 and proposed SR 3.7.4.1 are adequate to ensure the CREV 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.

LD.1 The Frequency for performing CTS 4.8.D.5.b and 4.8.D.5.c (proposed SR 3.7.4.3 and SR 3.7.4.4) has been extended from 18 months to 24 months. These SRs ensure that the CREV System is capable of automatic isolation and that the control room emergency zone boundary leakage is within the capacity of the CREV System by demonstrating that control room emergency zone can be maintained at a positive pressure with respect to adjacent areas when in the emergency pressurization mode of operation.

The proposed change will allow these Surveillances to extend the 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 CTS 4.0.B 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.B 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 these tests normally pass the Surveillances at the current Frequencies. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequencies will be minimal.

The CREV System will be tested every 31 days according to proposed SR 3.7.4.1, 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 CREV 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 the verification of control room emergency zone boundary integrity is acceptable because the control room emergency zone boundary is maintained at a positive pressure during normal operation. Therefore, any substantial degradation of the boundary that would prevent maintaining the control room emergency zone 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 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 The phrase "actual or," in reference to the simulated automatic isolation test signal in CTS 4.8.D.5.b.2), has been added to proposed SR 3.7.4.3, which verifies that the CREV System isolation dampers close on an actuation test signal. This allows satisfactory automatic CREV System damper isolation for other than surveillance purposes to be used to fulfill the Surveillance Requirement. Operability is adequately demonstrated in either case since the CREV System isolation dampers cannot discriminate between "actual" or "test" signals.

L.2 The requirement in CTS 4.8.D.5.b.1), the verification that the isolation dampers close on a manual initiation from the control room, has been deleted. The Control Room Manual Initiation Function is not credited in any design bases accident or transient analysis. The automatic isolation Functions are credited to

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) provide the appropriate isolation signal to the control room isolation dampers. The requirements to test the automatic portion of the control room isolation logic in CTS 4.8.D.5.b.2) and in CTS 3.2.A) are being retained in proposed SR 3.7.4.3 and proposed ITS 3.3.7.1, CREV Isolation Instrumentation, respectively. Since the Manual Initiation Function does not satisfy any criteria of 10 CFR 50. 36(c)(2)(ii), its removal from the Technical Specification is considered appropriate.

RELOCATED SPECIFICATIONS

None

AC A.2

3.8 - LIMITING CONDITIONS FOR OPERATION

D. Control Room Emergency Ventilation System

LCO 3.7.5 The control room emergency ventilation system shall be OPERABLE, with the system comprised of an OPERABLE control room emergency filtration system and an OPERABLE refrigeration control unit (RCU).

LA.1

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, and *.

ACTION:

1. In OPERATIONAL MODE(s) 1, 2 or 3:

   a. With the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

<See ITS 3.7.4>

   b. With the refrigeration control unit (RCU) inoperable, restore the inoperable RCU to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Action A

Action B

Control Room Emergency Ventilation AC System

A.2

4.8 - SURVEILLANCE REQUIREMENTS

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be demonstrated OPERABLE:

24 LD.1

SR 3.7.5.1 1. At least once per 18 months by verifying that the RCU has the capability to remove the required heat load.

<See ITS 3.7.4>

2. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters operating.

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

   a. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of $<0.05\%$ and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 scfm $\pm 10\%$.

<See ITS 5.5>

Applicability * When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

Action C 2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU Control Room Emergency Ventilation AC System inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

A.2

Action C Note 3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

<See ITS 3.7.4>

4.8 - SURVEILLANCE REQUIREMENTS

b. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

4. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

5. At least once per 18 months by:

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

<See ITS 5.5>

Applicability * When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

b. Verifying that the isolation dampers close on each of the following signals:

1) Manual initiation from the control room, and

2) Simulated automatic isolation signal.

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

<See ITS 3.7.4>

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

<See ITS 5.5>

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Quad Cities 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 CTS 3.8.D states that the control room emergency ventilation system includes the control room refrigeration control unit. In addition, the CTS 3.8.D Actions and the CTS 4.8.D Surveillance Requirements discuss both the control room emergency ventilation system and the refrigeration control unit. In the ITS, these two requirements have been split into separate Technical Specifications; ITS 3.7.4 for the Control Room Emergency Ventilation (CREV) System and ITS 3.7.5 for the Control Room Emergency Ventilation AC System. Therefore, in ITS 3.7.5, the LCO, Actions, and Surveillance Requirements all refer to the Control Room Emergency Ventilation AC System. Since the Control Room Emergency Ventilation AC System is included in ITS 3.7.5, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 3.8.D specific details of what constitutes the Control Room Emergency Ventilation and Control Room Emergency Ventilation AC Systems are proposed to be relocated to the Bases. These are design details that are not necessary to be included in the Technical Specifications to ensure the OPERABILITY of the Control Room Emergency Ventilation AC System since OPERABILITY requirements are adequately addressed in ITS 3.7.4 and ITS 3.7.5. 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LD.1 The Frequency for performing CTS 4.8.D.1 (proposed SR 3.7.5.1) has been extended from 18 months to 24 months. This SR ensures that the Control Room Emergency Ventilation AC System is capable of removing the required heat load from the control room emergency zone.

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.B 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.B 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 minimal.

The Control Room Emergency Ventilation AC System auto-starts on control room temperature when the Control Room Emergency Ventilation (CREV) System is operating. The CREV and Control Room Emergency Ventilation AC Systems are normally maintained in standby and are operated only for required Surveillances. The CREV System will be tested every 31 days according to proposed SR 3.7.4.1, therefore, any significant mechanical component failures will be detected and repaired during plant operation. Furthermore, the proposed 24 month Frequency for performing CTS 4.8.D.1 (proposed ITS SR 3.7.5.1) is deemed appropriate since significant degradation of the Control Room Emergency Ventilation AC System is not expected over this time period due to its normal standby status.

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"

None

RELOCATED SPECIFICATIONS

None

L

A.1

3.8 - LIMITING CONDITIONS FOR OPERATION

I. Main Condenser Offgas Activity

LCO 3.7.6 The release rate of the sum of the activities of the noble gases measured prior to the offgas holdup line shall be limited to ≤100 μCi/sec/MWt, after 30 minutes decay.

251,100 A.2

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

A.2 251,100

ACTION:

ACTION A With the release rate of the sum of the activities of the noble gases in the main condenser air ejector effluent (as measured prior to the offgas holdup line) >100 μCi/sec/MWt, after 30 minutes decay, restore the release rate to within its limit within 72 hours ~~or be in at least~~ STARTUP with the main steam isolation valves closed within the next 8 hours.

Required Action B.1

12 L.1

add proposed Required Action B.2 A.3

add proposed Required Action B.3.1 and B.3.1 L.2

4.8 - SURVEILLANCE REQUIREMENTS

I. Main Condenser Offgas Activity

LA.1

1. The release rate of noble gases from the main condenser air ejector shall be continuously monitored in accordance with the ODCM.

SR 3.7.6.1 2. The release rate of the sum of the activities from noble gases from the main condenser air ejector shall be determined to be within the limits of Specification 3.8.I at the following frequencies[b] by performing an isotopic analysis of a representative sample of gases taken at the recombiner outlet, or the air ejector outlet, if the recombiner is bypassed.

LA.2

a. At least once per 31 days, and

b. Within 4 hours following the determination of an increase of >50%.

or equal to M.1

after factoring out increases due to changes in THERMAL POWER level

L.3

With any main steam line not isolated and A.3

L.4

---

a When the main condenser air ejector is in operation.

SR 3.7.6.1 Note b The provisions of Specification 4.0.D are not applicable.

add proposed Note to SR 3.7.6.1

DISCUSSION OF CHANGES
ITS: 3.7.6 - MAIN CONDENSER OFFGAS

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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 CTS 3.8.I identifies the main condenser offgas release rate limit for noble gases as $\leq$ 100 microcuries/sec/MWt, after 30 minutes decay. The limit as specified in the CTS 3.8.I LCO and Action is proposed to be converted to units of μCi/sec. Since the unit THERMAL POWER licensing limit for Quad Cities 1 and 2 is 2511 MWt, the proposed limit in ITS LCO 3.7.6 and SR 3.7.6.1 is presented as 251,100 μCi/sec. Since there is no technical difference between the two limits, this change is considered a presentation preference only and, as such, is administrative.

A.3 CTS 3.8.I Applicability is modified by Note (a), "When the main condenser air ejector is in operation." An additional requirement has been added to the Applicability which includes the condition when any main steam line is not isolated. Since a main condenser air ejector cannot be placed in service without main steam pressure (i.e., any main steam line not isolated) the addition of this requirement is considered administrative. This proposed Applicability is consistent with the CTS 3.8.I default action to be in at least STARTUP with the main steam isolation valves closed. A new Required Action (ITS 3.7.6 Required Action B.2), which requires the isolation of the air ejector within 12 hours, has also been added consistent with the Applicability. Thus, these changes are considered administrative in nature only, since they are simply assuring the Actions and Applicability match up.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.8.I.2.b requires verification that the release rate of noble gases from the main condenser air ejector is within limits within 4 hours following an increase of > 50%. The amount of increase is changed from > 50% to include an increase equal to 50% in ITS SR 3.7.6.1. This is an inconsequential change that is considered more restrictive since 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 4.8.I.1 requirement to continuously monitor the radioactivity release rate of noble gases from the main condenser air ejector 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 release rate is within limits. Proposed SR 3.7.6.1 provides adequate assurance the main condenser offgas activity release 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.

LA.2 The CTS 4.8.I.2 detail defining the methods for performing this Surveillance is proposed to be relocated to the Bases. These details are not necessary to ensure the main condenser offgas activity release 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 release 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 default action of the CTS 3.8.I Action requires the plant to be in at least STARTUP with the main steam isolation valves to be closed in 8 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 of the Applicability of the Specification has been extended from 8 hours to 12 hours. The explicit requirement to be in STARTUP has been deleted since the closure of all main steam line isolation valves will require the mode switch to be placed in the startup/hot standby position to avoid a scram on Main Steam Line Isolation Valve— Closure. This proposed time is required to shutdown and cool down the unit from full power conditions and isolate the main steam isolation valves in an orderly manner and without challenging unit systems. This proposed time is considered reasonable based on operating

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) experience and is consistent with the BWR ISTS, NUREG-1433, Rev. 1. Allowing 12 hours to complete these 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 unit shutdown.

L.2 CTS 3.8.I Action requires the plant to be in at least STARTUP with the main steam isolation valves closed within 8 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 8 hours as currently required by the CTS 3.8.I Action (see Discussion of Change L.1 for further changes to the 8 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.3 CTS 4.8.I.2.b requires the main condenser offgas activity to be determined within 4 hours following the determination of an increase of 50%. Proposed ITS SR 3.7.6.1 requires the performance of this Surveillance at the same Frequency however it is proposed to allow factoring out increases in activity as a result of a THERMAL POWER increase. Main condenser offgas activity levels are expected to increase as a result of THERMAL POWER level increases. However, the increase is expected to stabilize. The intent of the Surveillance is to trend and determine the extent of fuel failure so that alternative plant operating strategies are taken. This change will therefore reduce the number of times the test must be performed when the main condenser offgas activity is expected to change (i.e., during THERMAL POWER increases) and only require it to be performed at this 4 hour Frequency when the increase in the activity level is not

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) expected and therefore fuel failure may exist. This change is acceptable since the offgas activity limit is not expected to be challenged during THERMAL POWER increases and offgas activity increases due to changes in THERMAL POWER would not necessarily be indicative of fuel failure.

L.4 CTS 4.8.I.2 requires the release rate of the sum of the activities from noble gases from the main condenser air ejector to be determined at a Frequency of at least every 31 days. However, footnote b (the provisions of Specification 4.0.D are not applicable) allows entry into the Applicability prior to performing the Surveillance. The proposed Note to ITS SR 3.7.6.1 retains this provision and clarifies when the Surveillance must be performed. The Note specifies that the Surveillance is not required to be met until 31 days after any main steam line is not isolated and SJAE in operation. This allowance is acceptable, since if the Surveillance were not met prior to unit shutdown, the anomaly would have been corrected prior to reactor startup. If the plant was shutdown for some other reason (scheduled refueling), the fission product release rate could be considered to be within limits since the activities during an outage are not expected to increase the fission product release rate. Furthermore, the allowance is acceptable in view of other instrumentation that will be continuously monitoring the offgas during plant operations. If this instrumentation indicated a $\geq$ 50% increase in the minimal steady state fission gas release after factoring out increases due to changes in THERMAL POWER level (refer to Discussion of Change L.3), the Surveillance must be performed.

RELOCATED SPECIFICATIONS

None

M.1

Insert New Specification 3.7.7

Insert proposed Specification 3.7.7, "Main Turbine Bypass System," as shown in the Quad Cities 1 and 2 Improved Technical Specifications.

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A new Specification requiring the Main Turbine Bypass System to be OPERABLE is proposed to be added. Proposed ITS 3.7.7, "Main Turbine Bypass System," will require the Main Turbine Bypass System to be OPERABLE or an MCPR penalty shall be applied. This proposed change is an additional restriction on plant operations since the CTS does not provide any explicit restrictions with the Main Turbine Bypass System inoperable and does not include any Surveillance Requirements associated with the system. This Specification will help ensure the safety analyses assumptions of certain events are maintained by limiting the resulting MCPR if the event were to occur. Appropriate ACTIONS and Surveillance Requirements have also been added.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

None

RELOCATED SPECIFICATIONS

None

3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

LCO 3.7.8 The pool water level shall be maintained at a level of ~~≥33 feet.~~

≥ 19 feet over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks M.1

APPLICABILITY: During movement of A.2

~~Whenever~~ 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.C are not applicable. LA.1 LA.2

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

SR 3.7.8.1 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.

DISCUSSION OF CHANGES
ITS: 3.7.8 - SPENT FUEL STORAGE POOL WATER LEVEL

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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 CTS 3.10.H, 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 3.10.H 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 3.10.H 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.10.H requires the spent fuel water level to be maintained at a level of $\geq$ 33 ft. ITS 3.7.8 requires the spent fuel storage pool water level to be $\geq$ 19 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. This change results in an increase in the water level of approximately 9 inches. This change is necessary to ensure the minimum water level in the spent fuel storage pool meets the assumptions of the fuel handling accident. Since this change increases the minimum required spent fuel storage pool water level, the change imposes more restrictive requirements on the movement of fuel assemblies.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 3.10.H 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, Section 9.1.4.3.2. 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 3.10.H 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

3.8 LIMITING CONDITIONS FOR OPERATION

J. Safe Shutdown Makeup Pump

LCO 3.7.9 The Safe Shutdown Makeup Pump (SSMP) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

ACTION A
1. With the SSMP system inoperable, restore the inoperable SSMP system to OPERABLE status within 14 days, ~~or~~

ACTION B
be in at least HOT SHUTDOWN within the next 12 hours and ~~in COLD SHUTDOWN~~ within the following 24 hours.

≤ 150 psig

A.2

4.8 - SURVEILLANCE REQUIREMENTS

J. Safe Shutdown Makeup Pump

The SSMP system shall be demonstrated OPERABLE:

1. At least once per 31 days by:

   a. SR 3.7.9.1 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.

   b. ~~Verifying that the pump flow controller is in the correct position.~~ LA.1

2. SR 3.7.9.2 At least once per 92 days by verifying that the SSMP develops a flow of greater than or equal to 400 gpm against a system head corresponding to reactor vessel pressure of greater than 1120 psig.

DISCUSSION OF CHANGES
ITS: 3.7.9 - SAFE SHUTDOWN MAKEUP PUMP SYSTEM

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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 CTS 3.8.J Action 1 requirement to be in COLD SHUTDOWN has been changed to reduce reactor steam dome pressure to $\leq$ 150 psig (ITS 3.7.9 Required Action B.2). Since this proposed Required Actions places the unit outside the Applicability of the current and proposed Specification, this change is considered to be administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail of CTS 4.8.J.1.b relating to the method for performing Surveillance (i.e., by verifying that the Safe Shutdown Makeup Pump System pump controller is in the correct position) is proposed to be relocated to the Bases. This detail is not necessary to ensure the OPERABILITY of the Safe Shutdown Makeup Pump System. The requirements of ITS 3.7.9, Safe Shutdown Makeup Pump System, and the associated Surveillance Requirements are adequate to ensure the Safe Shutdown Makeup Pump 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.

"Specific"

None

RELOCATED SPECIFICATIONS

None

CTS 3/4.8.E

R.1

3.8 - LIMITING CONDITIONS FOR OPERATION

E. Flood Protection

Flood protection shall be available for all required safe shutdown systems, components and structures.

APPLICABILITY:

At all times.

ACTION:

With the water level, as measured at the plant intake bay:

1. Above elevation 586 ft Mean Sea Level USGS datum, initiate the applicable flood protection measures.

2. Above, or predicted to exceed within 3 days, elevation 594 ft Mean Sea Level USGS datum, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.8 - SURVEILLANCE REQUIREMENTS

E. Flood Protection

The water level at the plant intake bay shall be determined to be within the limit by:

1. Measurement at least once per 24 hours when the water level is below elevation 585.5 ft Mean Sea Level USGS datum, and

2. Measurement at least once per 2 hours when the water level is equal to or above elevation 585.5 ft Mean Sea Level USGS datum.

Page 1 of 1

DISCUSSION OF CHANGES
CTS: 3/4.8.E - FLOOD PROTECTION

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 The purpose of flood protection requirements (CTS 3/4.8.E) is to protect the plant systems and equipment necessary for safe shutdown during high water conditions. Floods are not a design basis accident or transient, thus the flood protection equipment is not credited in any safety analysis. The Flood Protection Technical Specification requirements were put in place to ensure that timely action is taken when river level exceeds the specified limits. A high water level is a preliminary indication of flood conditions. Therefore, the requirements specified in CTS 3/4.8.E did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the Quad Cities 1 and 2 Technical Specifications and will be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the Quad Cities 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59

 LA.1

3.8 - LIMITING CONDITIONS FOR OPERATION

F. Snubbers

All required snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse impact on any safety-related system.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3. OPERATIONAL MODE(s) 4 and 5 for snubbers located on systems required OPERABLE in OPERATIONAL MODE(s) 4 and 5.

ACTION:

With one or more snubbers inoperable, on any system, within 72 hours:

1. Replace or restore the inoperable snubber(s) to OPERABLE status, and
2. Perform an engineering evaluation per Specification 4.8.F.7 on the attached component.

Otherwise, declare the attached system inoperable and follow the appropriate ACTION statement for that system.

4.8 - SURVEILLANCE REQUIREMENTS

F. Snubbers

Each snubber shall be demonstrated OPERABLE by the performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.E.

1. Inspection Types

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

2. 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 4.8.F-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.8.F-1[a].

3. 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 (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

a 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 nos 171 and 167.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

visual inspections shall be classified as unacceptable. 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.

Snubbers originally classified as unacceptable may be reclassified as 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 4.8.F.6.

4. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients, as determined from a review of operational data or a visual inspection of the systems, within 72 hours for accessible systems and 6 months for inaccessible systems following this determination. 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.

3.8 - LIMITING CONDITIONS FOR OPERATION	4.8 - SURVEILLANCE REQUIREMENTS

5. Functional Tests

At least once per 18 months, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. 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:

a. 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 4.8.F.6, 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

b. A representative sample of each type of snubber shall be functionally tested, in accordance with Figure 4.8.F-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.8.F.6. 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 4.8.F-1.

- CTS 3/4.8.F

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

If at any time the point plotted falls on or above the "Reject" line, all snubbers of that type shall be functionally tested. If at any time the point plotted falls on or below the "Accept" line, 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, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

c. An initial representative sample of 55 snubbers of each type 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

3.8 - LIMITING CONDITIONS FOR OPERATION | 4.8 - SURVEILLANCE REQUIREMENTS

"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 on or below the "Accept" line 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 practical 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, and failure of this functional test shall not be the sole cause for increasing the sample size under the sample plan. If during testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

6. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

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

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

b. The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

c. 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.

7. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause for 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.

3.8 - LIMITING CONDITIONS FOR OPERATION	4.8 - SURVEILLANCE REQUIREMENTS

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 4.8.F.5 for snubbers not meeting the functional test acceptance criteria.

8. <u>Functional Testing of Repaired and Replaced Snubbers</u>

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 result 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.

9. <u>Snubber Service Life Program</u>

The service life of all 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 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

3.8 - LIMITING CONDITIONS FOR OPERATION	4.8 - SURVEILLANCE REQUIREMENTS
	the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained.

TABLE 4.8.F-1

SNUBBER VISUAL INSPECTION CRITERIA

Population[a][b] or Category	NUMBER OF UNACCEPTABLE SNUBBERS Column A [c][f] Extend Interval	Column B [d][f] Repeat Interval	Column C [e][f] Reduce Interval
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	29	56	109

a 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 decision must be made and documented before any inspection and shall be used as the basis upon which to determine the next inspection interval for that category.

b Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use 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.

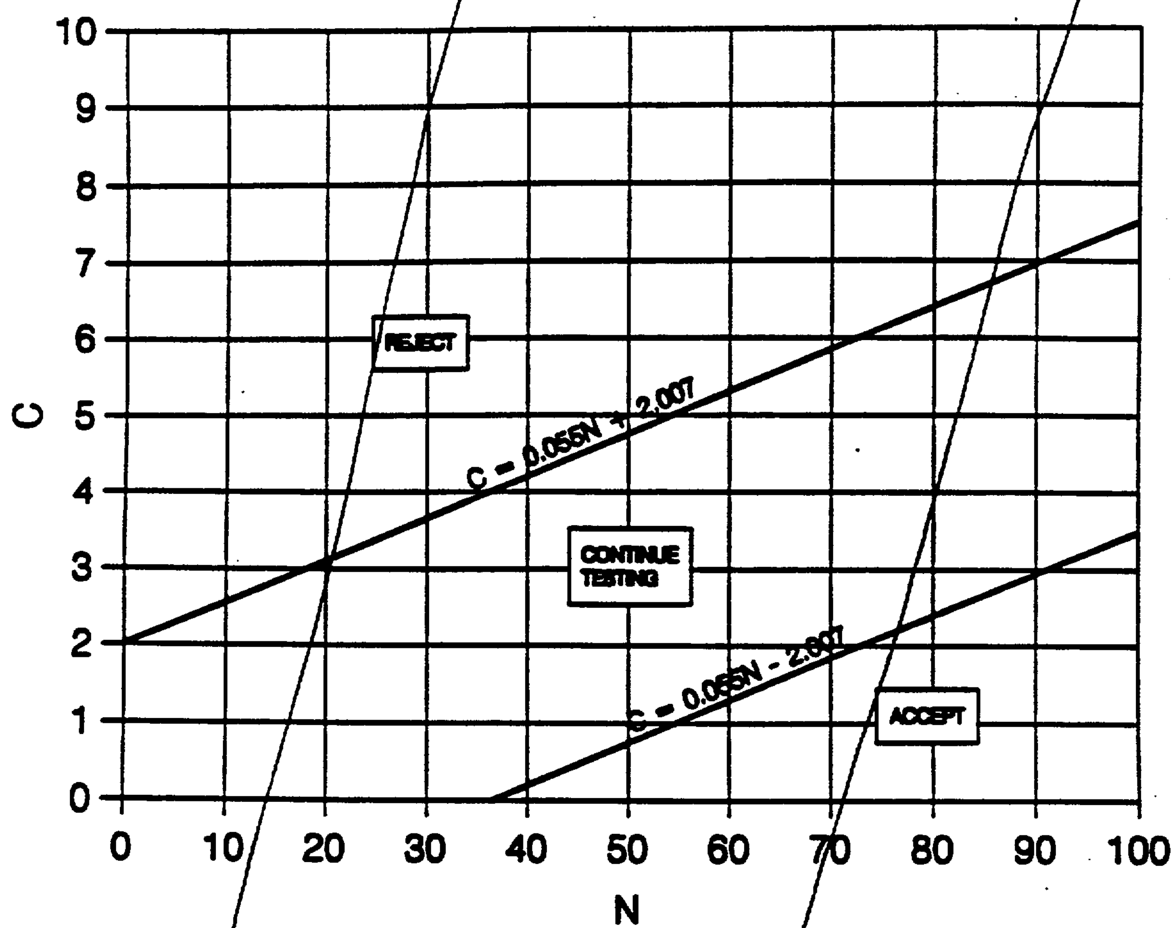
c If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval, but not greater than 48 months.

d 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.

e 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, but not less than 31 days. 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.

f The provisions of Specification 4.0.B are applicable for all inspection intervals up to and including 48 months.

FIGURE 4.8.F-1
SAMPLING PLAN FOR SNUBBER FUNCTIONAL TESTING



N = Cumulative number of snubbers of a type tested.

C = Total number of snubbers of a type not meeting acceptance requirements.

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS 3/4.8.F Snubber inspection and testing requirements will be part of the Quad Cities 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 Quad Cities 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 Quad Cities 1 and 2 UFSAR at ITS Implementation. Snubber inspections and testing will continue to be performed in accordance with the CTS 3/4.8.F 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

3.8 - LIMITING CONDITIONS FOR OPERATION

G. Sealed Source Contamination

Each sealed source containing radioactive material either in excess of 100 μCi of beta and/or gamma emitting material or 5 μCi of alpha emitting material shall be free of ≥0.005 μCi of removable contamination.

APPLICABILITY:

At all times.

ACTION:

1. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:
   a. Decontaminate and repair the sealed source, or
   b. Dispose of the sealed source in accordance with Commission Regulations.
2. With a sealed source leakage test revealing the presence of removable contamination in excess of the above limit, a report shall be prepared and submitted to the Commission on an annual basis.
3. The provisions of Specification 3.0.C are not applicable.

4.8 - SURVEILLANCE REQUIREMENTS

G. Sealed Source Contamination

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 μCi per test sample.
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 6 months for all sealed sources containing radioactive material:
      1) With a half-life >30 days, excluding Hydrogen 3, and
      2) In any form other than gas.
   b. Stored sources not in use - Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

CTS 3/4.8.G

R.1

3.8 - LIMITING CONDITIONS FOR OPERATION	4.8 - SURVEILLANCE REQUIREMENTS
	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.

DISCUSSION OF CHANGES
CTS: 3/4.8.G - SEALED SOURCE CONTAMINATION

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.8.G, which provides requirements for sealed source contamination, does not identify a parameter which is an initial condition assumption for a DBA or transient, 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 3/4.8.G did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the Quad Cities 1 and 2 Technical Specifications and will be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the Quad Cities 1 and 2 UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

The Bases of the current Technical Specifications for this section (pages B 3/4.8-1 through B 3/4.8-5, have been completely replaced by revised Bases that reflect the format and applicable content of the Quad Cities 1 and 2 ITS Section 3.7, consistent with the BWR Standard Technical Specification, NUREG-1433, Rev. 1. The revised Bases are as shown in the Quad Cities 1 and 2 ITS Bases.

<CTS>

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

<3.8.A> LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

<Appl/ 3.8.A> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.8.A Act 1.a> A. One RHRSW pump inoperable.	A.1 Restore RHRSW pump to OPERABLE status.	30 days
<3.8.A Act 1.b> B. One RHRSW pump in each subsystem inoperable.	B.1 Restore one RHRSW pump to OPERABLE status.	7 days
<3.8.A Act 1.c> <3.8.A Act 2> C. One RHRSW subsystem inoperable for reasons other than Condition A.	C.1 --------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.~~8~~ 7, "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.	1 2 7 days 3

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.8.A Act 1.d⟩ ⟨3.8.A Act 2⟩ D. Both RHRSW subsystems inoperable for reasons other than Condition B.	D.1 [1] ---------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.~~8~~ 7 for ~~{~~RHR shutdown cooling~~}~~ [2] subsystems made inoperable by RHRSW System. ------------------- Restore one RHRSW subsystem to OPERABLE status.	~~{~~8~~}~~ [2] hours
⟨3.8.A Act 1.c⟩ ⟨3.8.A Act 1.b⟩ ⟨3.8.A Act 1.c⟩ ⟨3.8.A Act 1.d⟩ E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. **AND** E.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
⟨4.8.A⟩ SR 3.7.1.1	Verify each RHRSW manual~~,~~ and 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 [4]

1. The correct LCO number has been included.

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

3. Typographical error corrected.

4. 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.

<CTS>

3.7 PLANT SYSTEMS

Cooling [1]

DG CW [3]

<3.8.B> 3.7.8 2 [2] Diesel Generator ~~(DG) [1B] Standby Service~~ Water (~~SSW~~) System

Insert LCO [4]

<M.1> LCO 3.7.8 ~~The DG [1B] SSW System shall be OPERABLE.~~

MODES 1, 2 and 3 [4]

<Appl 3.8.B> <M.1> APPLICABILITY: ~~When DG [1B] is required to be OPERABLE.~~

<A.2> ACTIONS

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

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

<3.8.B Act> [6] [6] [6]

4

<CTS>

Insert LCO

The following DGCW subsystems shall be OPERABLE:

<DOC M.1>

a. Two unit DGCW subsystems; and

b. The opposite unit DGCW subsystem capable of supporting its associated diesel generator (DG).

<CTS>

4

required DGCW subsystem

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.8.B.1> SR 3.7.~~8~~2.1 (2, 2)	Verify each ~~DG [1B] SSW System 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. 3	31 days
<4.8.B.2> SR 3.7.~~8~~2.2	Verify ~~the DG [1B] SSW System~~ each required DGCW 4 1 pump starts automatically when ~~DG [1B]~~ its associated DG 1 starts ~~and energizes the respective bus~~. 3 3	~~[18]~~ 24 1 months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.7.2 - DIESEL GENERATOR COOLING WATER (DGCW) SYSTEM

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

2. The proper LCO/SR number has been provided. This change was necessary since ISTS 3.7.2 (Plant Service Water System and Ultimate Heat Sink) has been numbered as ITS 3.7.3.

3. Changes have been made to reflect the plant specific design/nomenclature, or licensing basis requirements.

4. An additional requirement has been added to ISTS LCO 3.7.3 (ITS LCO 3.7.2) to address the shared systems (e.g., Standby Gas Treatment System and Control Room Emergency Ventilation System) between both units. The requirement was added as LCO 3.7.2.b to ensure that the opposite unit DGCW subsystem is OPERABLE and capable of supporting the opposite unit diesel generator (DG). This requirement was necessary to ensure that the opposite unit DG will be available to provide onsite AC electrical power to the shared systems in the event of a design basis accident (DBA) loss of coolant accident (LOCA). In addition, the Applicability has been revised to be consistent with the DG 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 3.8.B Action for one or more inoperable DGCW subsystems. The CTS 3.8.B Action requires the associated diesel generator to be declared inoperable and the applicable Actions of CTS 3.9.A, "A.C. Sources - Operating," or CTS 3.9.B, "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.

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 Quad Cities 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 Quad Cities 1 and 2 licensing basis does not provide the allowance of aligning a qualified alternative cooling water source to the DGs in the event one or more DGCW subsystems are inoperable. For Quad Cities 1 and 2, when one or more DGCW subsystems are inoperable, CTS 3.8.B requires the associated DG to be declared

6. (continued)

inoperable and the applicable Actions of Specifications 3.9.A, "A.C. Sources - Operating," or 3.9.B, "A.C. Sources - Shutdown," to be taken. Thus, since the current 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

<3.8.C> 3.7.2 3 [Plant Service Water (PSW)] ~~System and~~ [Ultimate Heat Sink (UHS)] 1 2

3

LCO 3.7.2 ~~Two [PSW] subsystems and~~ [UHS] shall be OPERABLE. 2

The 1

<Appl 3.8.C> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One [PSW] pump inoperable.~~	~~A.1 Restore [PSW] pump to OPERABLE status.~~	~~30 days~~
~~B. One [PSW] pump in each subsystem inoperable.~~	~~B.1 Restore one [PSW] pump to OPERABLE status.~~	~~7 days~~
~~[C. One or more cooling towers with one cooling tower fan inoperable.~~	~~C.1 Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days]~~

1 4

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One [PSW] subsystem inoperable for reasons other than Condition[s] A [and C]. 4	D.1 ---------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for diesel generator made inoperable by [PSW]. 2. Enter applicable Conditions and Required Actions of LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for [RHR shutdown cooling] made inoperable by [PSW]. -------------------- Restore the [PSW] subsystem to OPERABLE status.	72 hours

1

(continued)

< (TS)>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A D. ~~Required Action and associated Completion Time of Condition A or B not met.~~ ~~OR~~ ~~Both [PSW] subsystems inoperable for reasons other than Condition(s) B [and C].~~ 4 [OR [UHS] inoperable for reasons other than Condition C.] 2 4	D.1 Be in MODE 3. AND A 1 D.2 Be in MODE 4.	12 hours 36 hours

<3.8.C Act>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~[SR 3.7.2.1]~~	~~Verify the water level of each [PSW] cooling tower basin is ≥ [ ] ft.~~	~~24 hours~~ 4
[SR 3.7.2.2] 3 1 4	Verify the water level, [in ~~each PSW pump well of~~ the intake ~~structure~~] bay is ≥ [60.5] 568 ft [mean sea level]. 2	[24 hours] 2
[SR 3.7.2.3] 3 2	Verify the average water temperature of [UHS] is ≤ [ ] 95 °F. 2	[24 hours] 2

<4.8.C>

<4.8.C>

3 4 2

(continued)

<CTS>

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[SR 3.7.2.4	Operate each [PSW] cooling tower fan for ≥ [15] minutes.	31 days]
SR 3.7.2.5	-----NOTE----- Isolation of flow to individual components does not render [PSW] System inoperable. ----- Verify each [PSW] subsystem manual, power operated, and automatic valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.2.6	Verify each [PSW] subsystem actuates on an actual or simulated initiation signal.	[18] months

4

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.7.3 - ULTIMATE HEAT SINK (UHS)

1. The requirements related to the Plant Service Water System have been deleted since they are not applicable to Quad Cities 1 and 2. The Residual Heat Removal Service Water System and the Diesel Generator Cooling Water Systems, in conjunction with the UHS, are required for safe shutdown. The requirements of these systems are included in ITS 3.7.1 and 3.7.2, respectively. Therefore, the Plant Service Water System requirements, including the associated ACTIONS and Surveillance Requirements have been deleted. The subsequent requirements have been renumbered where applicable.

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

3. ISTS 3.7.2 has been renumbered as ITS 3.7.3, consistent with the sequence in the CTS.

4. ISTS 3.7.2 Action C and ISTS SR 3.7.2.1 and SR 3.7.2.4 have been deleted since the Quad Cities 1 and 2 design does not include cooling tower fans or a fan basin water level requirements.

3.7 PLANT SYSTEMS

Emergency Ventilation (CREV)

3.7.4 ~~[Main]~~ Control Room ~~Environmental Control (MCREC)~~ System

1

⟨3.8.D⟩ LCO 3.7.4 The CREV ~~Two [MCREC] sub~~system~~s~~ shall be OPERABLE.

2

⟨Appl 3.8 D⟩ 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
⟨3.8.D Act 1.a⟩	A. ~~One~~ CREV ~~[MCREC] sub~~system inoperable in MODE 1, 2 or 3. 2	A.1 Restore CREV ~~[MCREC] sub~~system to OPERABLE status. 2	7 days
⟨3.8.D Act 1.a⟩	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)

CREV System inoperable 2

ACTIONS (continued)

3.8.D Act 3

3.8.D Act 2

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. (1)	------------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. -------------------- Place OPERABLE [MCREC] subsystem in [pressurization] mode.~~ (3)	~~Immediately~~
	~~OR~~	
	C.2.1 Suspend movement of irradiated fuel assemblies in the ~~[secondary]~~ containment. (3)	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
~~D. Two [MCREC] subsystems inoperable in MODE 1, 2, or 3.~~	~~D.1 Enter LCO 3.0.3.~~	~~Immediately~~ (3)

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two [MCREC] 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

3

SURVEILLANCE REQUIREMENTS

2

	SURVEILLANCE	FREQUENCY
⟨4.8.D.2⟩ SR 3.7.4.1	Operate the each CREV [MCREC] subsystem for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days
⟨4.8.D.3⟩ ⟨A.2⟩ SR 3.7.4.2	Perform required CREV [MCREC] filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]

1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.8.D.5.b> SR 3.7.4.3	Verify ~~each~~ the CREV System ~~[MCREC] subsystem actuates~~ isolation dampers close on an actual or simulated initiation signal.	~~[18]~~ 24 months 4
<4.8.D.5.c> SR 3.7.4.4	2 Verify ~~each~~ the ~~[MCREC]~~ CREV ~~sub~~system can maintain a positive pressure of ≥ ~~[0.1]~~ 0.125 inches water gauge relative to the ~~[turbine building]~~ adjacent areas during the ~~{pressurization}~~ mode of operation at a flow rate of ≤ ~~[400]~~ 2000 cfm. 2	~~[18]~~ 24 months ~~on a STAGGERED TEST BASIS~~ 5 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.7.4 - CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM

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

2. Changes have been made to reflect the plant specific nomenclature/value and the current licensing basis requirements.

3. The Quad Cities 1 and 2 CREV System consists of a single train. Therefore, Required Action C.1 and the associated bracketed NOTE have been deleted. Subsequent Required Actions have been renumbered. In addition, Actions D and E have been deleted for the same reason. These changes are consistent with the current licensing basis.

4. Due to the design of the Quad Cities 1 and 2 CREV System, proposed ITS 3.7.4.3 has been revised to reflect the proper plant specific design. Isolation for the pressurization mode is the only automatic function associated with the CREV System.

5. Due to the design of the Quad Cities 1 and 2 CREV System, the control room emergency zone pressurization test (ISTS SR 3.7.4.4) must be verified every 24 months. Therefore, the SR has been revised to be consistent with the Quad Cities 1 and 2 current licensing basis.

<CTS>

3.7 PLANT SYSTEMS

3.7.5 {Control Room Air Conditioning (AC)} System 1

<3.8.D> LCO 3.7.5 The ~~Two~~ {control room AC} Emergency Ventilation ~~sub~~systems shall be OPERABLE. 1 2

<Appl 3.8.D> APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the {secondary} containment, 1
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.8.D Act 1.b> 2 A. ~~One~~ {control room AC} Emergency Ventilation ~~sub~~system inoperable in MODE 1, 2, or 3.	A.1 Restore {control room AC} Emergency Ventilation ~~sub~~system to OPERABLE status.	30 days 1
<3.8.D Act 1.b> 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)

2 Control Room Emergency Ventilation AC System inoperable

ACTIONS (continued)

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 AC] subsystem in operation.~~	~~Immediately~~
	~~OR~~	
	C.2.1 Suspend movement of irradiated fuel assemblies in the ~~secondary~~ containment.	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
~~D. Two [control room AC] subsystems inoperable in MODE 1, 2, or 3.~~	~~D.1 Enter LCO 3.0.3.~~	~~Immediately~~

⟨3.8.D Act. 3⟩

⟨3.8.D Act 2⟩

1 1 1 3 3 3

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two [control room 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
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate actions to suspend OPDRVs.	Immediately

3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify ~~each~~ the ~~control room AC subsystem~~ Emergency Ventilation system has the capability to remove the assumed heat load.	~~[18]~~ 24 months

⟨4.8.D.1⟩ 2 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.7.5 - CONTROL ROOM EMERGENCY VENTILATION
AIR CONDITIONING (AC) SYSTEM

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

2. Changes have been made to reflect the plant specific nomenclature/value and the current licensing basis requirements.

3. The Control Room Emergency Ventilation AC System consists of a single train. Therefore, Required Action C.1 has been deleted and the subsequent Required Actions renumbered. In addition, Actions D and E have been deleted for the same reason. These changes are consistent with the current licensing basis.

E <CTS>

3.7 PLANT SYSTEMS

3.7.6 Main Condenser Offgas

$\boxed{1}$ prior to the offgas holdup line

<3.8.I>

LCO 3.7.6 The gross gamma activity rate of the noble gases measured ~~at~~ ~~[the main condenser evacuation system pretreatment monitor station]~~ shall be ≤ ~~[240]~~ mCi/second [after decay of 30 minutes]. $\boxed{1}$ 251,100μ

<Appl 3.8.I>
<3.8.I footnote a>

APPLICABILITY: MODE 1,
MODES 2 and 3 with any [main steam line not isolated and] steam jet air ejector (SJAE) in operation. $\boxed{1}$

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.8.I 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
<3.8.I Act> <DOC A.3> B. Required Action and associated Completion Time not met.	[B.1	Isolate all main steam lines.	12 hours] $\boxed{1}$
	OR]		
	B.2	Isolate SJAE.	12 hours
	OR		
<Doc L.2>	B.3.1	Be in MODE 3.	12 hours
	AND		
	B.3.2	Be in MODE 4.	36 hours

<CTS>

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. ----- 1 251,100 Verify the gross gamma activity rate of the noble gases is ≤ ~~[240]~~ μCi/second [after decay of 30 minutes]. 1 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

<4.8.I>
<4.8.I footnote b>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.7.6 - MAIN CONDENSER OFFGAS

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

3.7 PLANT SYSTEMS

3.7.7 The Main Turbine Bypass System

⟨DOC M.1⟩ 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.] 1

1

TSTF-319 changes not adopted 2

⟨DOC M.1⟩ APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
⟨DOC M.1⟩	SR 3.7.7.1 Verify one complete cycle of each main turbine bypass valve.	~~31~~ days 92 3

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
⟨DOC M.1⟩	SR 3.7.7.2 Perform a system functional test.	~~[18]~~ 24 months 1
⟨DOC M.1⟩	SR 3.7.7.3 Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	~~[18]~~ 24 months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.7.7 - MAIN TURBINE BYPASS SYSTEM

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

2. TSTF-319 revised the Main Turbine Bypass System LCO (ISTS LCO 3.7.7) to require adjusting APLHGR limits, in addition to the ISTS LCO 3.7.7 requirement to adjust MCPR limits, when the Main Turbine Bypass System is inoperable. The plant-specific turbine bypass valve out-of-service analysis does not require adjustments of APLHGR or LHGR limits when the Main Turbine Bypass System is inoperable. Therefore, the change from TSTF-319 is not adopted.

3. ISTS SR 3.7.7.1, "Verify one complete cycle of each main turbine bypass valve," has a Frequency of "31 days." The Frequency is being changed to "92 days" based on the main turbine manufacturer's recommendations for functional testing of the turbine bypass valves.

⟨CTS⟩

3.7 PLANT SYSTEMS

3.7.8 Spent Fuel Storage Pool Water Level

⟨3.10.H⟩ LCO 3.7.8 The spent fuel storage pool water level shall be ≥ ~~[~~23~~]~~ ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. 19 1

⟨Appl 3.10.H⟩ APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool, 5 During movement of new fuel assemblies in the spent fuel storage pool with irradiated fuel assemblies seated in the spent fuel storage pool. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.10.H Act⟩ A. Spent fuel storage pool water level not within limit.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ------------------ Suspend movement of ~~irradiated~~ fuel assemblies in the spent fuel storage pool. 2	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
⟨4.10.H⟩ SR 3.7.8.1 Verify the spent fuel storage pool water level is ≥ ~~[~~23~~]~~ ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. 19 1	7 days

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.7.8 - SPENT FUEL STORAGE POOL WATER LEVEL

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

2. 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.

1 Insert ITS 3.7.9

<CTS>

3.7 PLANT SYSTEMS

3.7.9 Safe Shutdown Makeup Pump (SSMP) System

<3.8.J> LCO 3.7.9 The SSMP System shall be OPERABLE.

<App1 3.8.J> APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.8.J Act 1> A. SSMP System inoperable.	A.1 Restore SSMP System to OPERABLE status.	14 days
<3.8.J Act 1> B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Reduce reactor steam dome pressure to $\leq$ 150 psig.	12 hours 36 hours

Insert Page 3.7-21

<CTS> 1 Insert ITS 3.7.9 (continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.9.1 <4.8.J.1>	Verify each SSMP System 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
SR 3.7.9.2 <4.8.J.2>	Verify SSMP System pump develops a flow rate $\geq$ 400 gpm against a system head corresponding to reactor pressure > 1120 psig.	92 days

Insert Page 3.7-22

1. This Specification has been added consistent with the current licensing basis. The Safe Shutdown Makeup Pump System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

All changes are [1] unless otherwise indicated

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.

located on the RHR heat exchanger discharge header

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.

Each pump can provide sufficient flow to the heat exchanger (3500 gpm) and all auxiliary loads

[2] (1) U [1]

Insert BKGD-1

suction pipes which begin in the crib house

via

discharge [2]

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.~~

and [3]

occurs [3]

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.

and adequate electrical power is available

(continued)

1

Insert BKGD-1

The Ultimate Heat Sink (UHS) consists of the Mississippi River, the intake flume, the crib house, and the discharge structure and flume. The UHS is described in the UFSAR Section 9.2.5, Reference 2.

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 containment is discussed in ~~the~~ FSAR, ~~Chapters [6] and [15]~~ Section 6.2 (Ref~~s. 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.

1 4 2 1 1

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, Section ~~[6.2.1.4.3]~~ (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 ~~[4000]~~ 3500 gpm ~~per pump~~ to the associated heat exchanger with ~~two~~ one pump~~s~~ operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are ~~[206.4]~~ 177 °F and ~~[36.59]~~ 28 psig, respectively, well below the design temperature of ~~[340]~~ 281 °F and maximum allowable pressure of [62] psig.

2 3 1 4 2 2 2

The RHRSW System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii).

1

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)

Additionally, the RHRSW discharge header motor operated valves must be closed so that failure of one subsystem will not affect the OPERABILITY of the other subsystem

1

1

UHS

and separately to the associated safety related equipment

BASES

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.~~

7

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

1

1

3 4

An adequate suction source is not addressed in this LCO since the minimum net positive suction head ~~(X59 ft mean sea level in the pump well)~~ ~~is bounded by the plant service water pump requirements~~ (LCO 3.7.2, "~~Plant Service Water (PSW) System and~~ Ultimate Heat Sink (UHS)").

1

1

2

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.

4

7

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

5

Insert Appl

ACTIONS

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.

1

(continued)

5

Insert 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 3.4.8, "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

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

1 can provide

with concurrent worst case single failure

1 in each subsystem

1

and low probability of an event occurring requiring RHRSW during this time period 3

C.1

Required Action 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.

7 4

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

an 3

subsystem 2

D.1

With both RHRSW subsystems inoperable for reasons other than Condition B (e.g., both subsystems with inoperable flow

(continued)

BASES

ACTIONS

D.1 (continued)

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.

an 7 4

3 The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.8, be entered and Required Actions taken if ~~the~~ an inoperable RHRSW subsystem results in an inoperable [RHR shutdown cooling]. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. subsystem 2

E.1 and E.2

If any Required Action and associated Completion Time of Conditions A, B, C or D are not met 6

~~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.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual, and 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. 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

1. FSAR, Section [9.2.7].
2. UFSAR, Section 9.2.5.
2. FSAR, ~~Chapter [6]~~. Section 6.2
~~3. FSAR, Chapter [15].~~
4. FSAR, Section [6.2.1.1.3].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.1 - RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

1. 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.

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

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

4. The proper Quad Cities 1 and 2 LCO number has been provided.

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

6. Changes have been made to more closely reflect the Condition.

7. Typographical error corrected.

All changes are 3 unless otherwise indicated

B 3.7 PLANT SYSTEMS

B 3.7.2 Diesel Generator Cooling ~~(DG) [1B] Standby Service~~ Water (DGCW ~~SSW~~) System

BASES

BACKGROUND

The DGCW ~~DG [1B] SSW~~ System is designed to provide cooling water for the removal of heat from the heat exchangers ~~DG [1B]. DG [1B] is the only component served by the DG [1B] SSW System~~. Insert BKGD-1

The DGCW ~~DG [1B] SSW~~ pump autostarts upon receipt of a DGCW ~~diesel generator (DG)~~ start signal when power is available to the pump's electrical bus. Cooling water is pumped from the suction header of Residual Heat Removal Service Water (RHRSW) System ~~[Altamaha River]~~ by the DGCW ~~DG [1B] SSW~~ pump to the associated ~~essential~~ DG ~~components through the SSW supply header~~. After removing heat from the heat exchangers ~~components~~, the water is discharged to the discharge flume ~~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.~~ Insert BKGD-2 A complete description of the DGCW ~~DG [1B] SSW~~ System is presented in the UFSAR, Section ~~[~~9.5.5~~]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ability of the DGCW ~~DG [1B] SSW~~ System to provide adequate cooling to the heat exchangers and ECCS room emergency coolers ~~DG [1B]~~ is an implicit assumption for the safety analyses presented in the 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 ~~DG [1B] SSW~~ System to cool the DG~~[1B]~~s.

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

LCO

The OPERABILITY of the unit DGCW ~~DG [1B] SSW~~ System is required to provide a coolant source to ensure effective operation of the DGs ~~[1B]~~ and ECCS in the event of an accident or transient. The OPERABILITY of the DGCW ~~DG [1B] SSW~~ System is based on having an OPERABLE pump and an OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring cooling water to the associated DG heat exchangers and ECCS room emergency coolers. Insert LCO

An adequate suction source is not addressed in this LCO since the minimum net positive suction head of the DGCW ~~DG [1B] SSW~~ pump and the maximum suction source temperature are covered by the requirement specified in ~~is bounded by the PSW requirements~~ ~~[~~LCO 3.7.2 ~~[~~"~~[~~Unit Service Water (PSW)~~]~~ ~~System and [~~Ultimate Heat Sink (UHS)~~]~~".

(continued)

1 3

Insert BKGD-1

diesel generator (DG) heat exchangers and the Emergency Core Cooling System (ECCS) room emergency coolers. Each unit DGCW subsystem provides cooling water to its associated DG and the unit ECCS room emergency coolers. The DG 1/2 DGCW subsystem may be manually aligned to provide cooling to either unit's ECCS room emergency coolers.

3

Insert BKGD-2

The DGCW subsystem associated with DG 1 (DG 2) is also normally aligned to provide cooling water to the unit ECCS room emergency coolers. However, the DGCW subsystem associated with DG 1/2 can be aligned as an alternate source of cooling water to the Unit 1 or Unit 2 ECCS room emergency coolers. The DGCW subsystem associated with DG 1 can be aligned as an alternate source of cooling water to the DG 1/2 heat exchanger.

2

Insert LCO

The OPERABILITY of the opposite unit's DGCW subsystem is required to provide adequate cooling to ensure effective operation of the required opposite unit's DG heat exchanger in the event of an accident in order to support operation of the shared systems such as the Standby Gas Treatment System and Control Room Emergency Ventilation System.

All changes are 2 unless otherwise indicated

BASES (continued)

APPLICABILITY

Insert APPL

~~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").~~

ACTIONS

Insert ACTIONS

~~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.~~

A ~~B~~.1

2 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 1 DG ~~[1B]~~ and ECCS components supported by the affected ECCS room emergency coolers, 1 cannot perform ~~its~~ their 3 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 System (ECCS)—Operating," as applicable 4 ~~or LCO 3.8.2~~. ~~Additionally, if the DG [1B] SSW System is~~

(continued)

2

Insert APPL

In MODES 1, 2, and 3, the DGCW subsystems are required to be OPERABLE 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.

2

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.

BASES

ACTIONS

~~B.1 (continued)~~

~~not restored to OPERABLE status within 60 days, DG [1B] must be immediately declared inoperable.~~ 2

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 2 2

Verifying the correct alignment for manual, ~~power operated, and automatic~~ valves in the ~~DG [1B] SSW~~ DGCW System flow path sub s provides assurance that the proper flow paths will exist for ~~DG [1B] SSW~~ DGCW System operation. This SR does not apply to ~~valves that~~ sub 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. 2 ~~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.~~ 3 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 2 2

This SR ensures that each required ~~the~~ ~~DG [1B] SSW~~ DGCW 1 sub System pump will automatically start to provide required cooling to the associated 1 3 ~~DG [1B]~~ DG heat exchangers and ECCS room emergency coolers 1 3 when the DG (1B) 5 starts ~~and the respective bus is energized~~ 2.

Operating experience has shown that these components usually pass the SR when performed at the ~~[18]~~ 24 1 month Frequency, which is based at the refueling cycle. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES 3
1. FSAR, Section [9.5.5].
2. FSAR, Chapter [6].
3. FSAR, Chapter [15].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.2 - DIESEL GENERATOR COOLING WATER (DGCW) SYSTEM

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

2. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

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

5. The bracketed requirement/information has been deleted because it is not applicable to Quad Cities 1 and 2.

1

B 3.7 PLANT SYSTEMS

B 3.7.2 [Plant Service Water (PSW)] System and [Ultimate Heat Sink (UHS)] 2

3 ( )

BASES

BACKGROUND

1 4

Insert BKGD-1

~~The [PSW] System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), residual heat removal (RHR) pump coolers, and room coolers for Emergency Core Cooling System equipment, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The [PSW] System also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, nonessential loads are automatically isolated, the essential loads are automatically divided between [PSW] Divisions 1 and 2, and one [PSW] pump is automatically started in each division.~~

~~The [PSW] System consists of the [UHS] and two independent and redundant subsystems. Each of the two [PSW] subsystems is made up of a header, two [8500] gpm pumps, a suction source, valves, piping and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity to support the required systems with one pump operating. The two subsystems are separated from each other so failure of one subsystem will not affect the OPERABILITY of the other system.~~

4

Insert BKGD-2

~~Cooling water is pumped from the [Altamaha River] by the [PSW] pumps to the essential components through the two main headers. After removing heat from the components, the water is discharged to the circulating water flume to replace evaporation losses from the circulating water system, or directly to the river via a bypass valve.~~

APPLICABLE SAFETY ANALYSES

the RHRSW and the DGCW 2 5

Sufficient water inventory is available for ~~all [PSW]~~ System post LOCA cooling requirements ~~for a 30-day period with no additional makeup water source available~~. 4 This water source is provided by the UHS. 4 The ability of the ~~[PSW] System~~ UHS 2 to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the FSAR, ~~Chapters [4] and [6]~~ Section 6.2 4 7 (Refs. ~~1 and 2, respectively~~). These 4

(continued)

1 4

Insert BKGD-1

The Residual Heat Removal Service Water (RHRSW) and the Diesel Generator Cooling Water (DGCW) Systems are designed to provide cooling water to components required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is described in UFSAR, Section 9.2.1 (Ref. 1) while the DGCW System is described in UFSAR, Section 9.5.5 (Ref. 2). These systems are also described in the Bases for LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System," and LCO 3.7.2, "Diesel Generator Cooling Water (DGCW) System." The UHS provides a suction source and discharge pathway for the cooling water associated with these systems. The UHS is described in UFSAR, Section 9.2.5 (Ref. 3).

4

Insert BKGD-2

The Mississippi River provides an UHS with sufficient cooling capacity to either provide normal cooldown of the units, or mitigate the effects of accident conditions within acceptable limits for one unit while conducting a normal cooldown of the other unit. The water flows under a floating boom to the intake flume and into the intake bay of the crib house, where it is directed to various plant systems. There are seven bays within the crib house, one associated with each of the six circulating water pumps and another which houses the 1/2 B diesel-driven fire pump. The bay housing the 1/2 B diesel-driven fire pump receives its water from two of the bays associated with the circulating water pumps (bays 1A and 2C). This bay also supplies water to a suction header for each RHRSW subsystem (2 for each unit). Each DGCW subsystem also obtains a suction from one of these headers. The DGCW subsystem associated with DG 1/2 obtains a suction from one of the Unit 1 RHRSW suction headers. The RHRSW and DGCW Systems for both units can receive a sufficient amount of water from either bay 1A or 2C. The UHS also contains a discharge flume where the water is returned to the Mississippi River. A weir wall in the discharge flume maintains a minimum level in the discharge bay to ensure flow is directed to the river.

BASES

APPLICABLE SAFETY ANALYSES (continued)

analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The ability of the {PSW} System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 1 and 2. The ability to provide onsite emergency AC power is dependent on the ability of the {PSW} System to cool the DGs. The long term cooling capability of the RHR, core spray, and RHR service water pumps is also dependent on the cooling provided by the [PSW] System.

The [PSW] System, together with the {UHS}, satisfy Criterion 3 of the NRC Policy Statement.

UHS 2 4 DGCW 4 1 UHS 2 ies 10 CFR 50.36(c)(2)(ii) 4

LCO

The [PSW] subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one subsystem of [PSW] is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems of [PSW] must be OPERABLE. At least one subsystem will operate, if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered OPERABLE when it has an OPERABLE [UHS], two OPERABLE pumps, and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the appropriate equipment.

1

The OPERABILITY of the {UHS} is based on having a minimum water level in the pump well of the intake structure of {60.7} ft mean sea level and a maximum water temperature of {90}°F.

2 bay 4 568 2 95

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

1

APPLICABILITY

In MODES 1, 2, and 3, the [PSW] System and {UHS} are required to be OPERABLE to support OPERABILITY of the

1 2 is

(continued)

BASES

RHRSW and DG CW

APPLICABILITY (continued)

equipment serviced by the [PSW] System. Therefore, the [PSW] System and [UHS] are required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the [PSW] System and [UHS] are determined by the systems they support.

ACTIONS

A.1

With one [PSW] pump inoperable in each subsystem, the inoperable pump must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE [PSW] pumps (even allowing for an additional single failure) are adequate to perform the [PSW] heat removal function; however, the overall reliability is reduced. The 30 day Completion Time is based on the remaining [PSW] heat removal capability to accommodate additional single failures, and the low probability of an event occurring during this time period.

B.1

With one [PSW] pump inoperable in each subsystem, one inoperable pump must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE [PSW] pumps are adequate to perform the [PSW] heat removal function; however, the overall reliability is reduced. The 7 day Completion Time is based on the remaining [PSW] heat removal capability to accommodate an additional single failure and the low probability of an event occurring during this time period.

C.1

[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. The 7 day Completion Time is 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]

(continued)

BASES

ACTIONS

C.1 (continued)

the time required to reasonably complete the Required Action.

5

D.1

With one [PSW] subsystem inoperable for reasons other than Condition A and [Condition C] (e.g., inoperable flow path or both pumps inoperable in a loop), the [PSW] subsystem must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE [PSW] subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE [PSW] subsystem could result in loss of [PSW] function.

1

The 72 hour Completion Time is based on the redundant [PSW] System capabilities afforded by the OPERABLE subsystem, the low probability of an accident occurring during this time period, and is consistent with the allowed Completion Time for restoring an inoperable DG.

Required Action D.1 is modified by two Notes indicating that the applicable Conditions of LCO 3.8.1, "AC Sources—Operating," LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," be entered and Required Actions taken if the inoperable [PSW] subsystem results in an inoperable DG or RHR shutdown cooling subsystem, respectively. This is in accordance with LCO 3.0.6 and ensures the proper actions are taken for these components.

A
1 5

E.1 and E.2

1

If the [PSW] subsystem cannot be restored to OPERABLE status within the associated Completion Time, or both [PSW] subsystems are inoperable for reasons other than Condition B and [Condition C], [or the [UHS] is determined inoperable for reasons other than Condition C] 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 2

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR ensures adequate long term (30 days) cooling can be maintained. With the [UHS] water source below the minimum level, the affected [PSW] 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 3.7.2.2

This SR verifies the water level [in each pump well] of the intake [structure] to be sufficient for the proper operation of the [PSW] 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.

bay

RHRSW and DGCW

SR 3.7.2.3

Verification of the [UHS] temperature ensures that the heat removal capability of the [PSW] 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.

RARSW and DGCW

ies

are

SR 3.7.2.4

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 motorfailure, 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.4 (continued)

significant degradation of the cooling tower fans occurring between surveillances.

5

SR 3.7.2.5

Verifying the correct alignment for each manual, power operated, and automatic valve in each [PSW] subsystem flow path provides assurance that the proper flow paths will exist for [PSW] 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 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.

1

This SR is modified by a Note indicating that isolation of the [PSW] System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the [PSW] System. As such, when all [PSW] pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the [PSW] 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 3.7.2.6

This SR verifies that the automatic isolation valves of the [PSW] 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 the use of an actual or simulated

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.6 (continued)

initiation signal. This SR also verifies the automatic start capability of one of the two [PSW] pumps in each subsystem.

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. 1

REFERENCES 4

1. a FSAR, ~~Chapter [4]~~. Section 9.2.1 2
2. FSAR, ~~Chapter [6]~~. Section 9.5.5

3. UFSAR Section 9.2.5
4. UFSAR Section 6.2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.3 -ULTIMATE HEAT SINK (UHS)

1. Changes have been made to reflect changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

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

3. ISTS 3.7.2 has been renumbered as ITS 3.7.3, consistent with the sequence in the CTS.

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

5. ISTS 3.7.2 Action C and ISTS SR 3.7.2.1 and SR 3.7.2.4 have been deleted since the Quad Cities 1 and 2 design does not include cooling tower fans or a fan basin water level requirement.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Emergency Ventilation (CREV) ~~[Main Control Room Environmental Control (MCREC)]~~ System 1

BASES

BACKGROUND

The CREV ~~[MCREC]~~ System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA). 1 Insert BKGD-1

The safety related function of the CREV ~~[MCREC]~~ System ~~includes two independent and redundant~~ consists of a single high efficiency air filtration ~~subsystems~~ train for emergency treatment of ~~recirculated air or~~ outside supply air. ~~Each subsystem~~ The filter train 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~~ two 100% capacity booster fans in parallel, ~~an~~ the Train B air handling unit (excluding the refrigeration condensing unit), and the associated ductwork and dampers. ~~Demisters remove water droplets from the airstream.~~ The electric heater is used to limit the relative humidity of the air entering the filter train. Prefilters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The CREV ~~[MCREC]~~ System is a standby system, parts of which also operate during normal unit operations to maintain the control room emergency zone environment. 1 Upon receipt of ~~the initiation~~ an isolation signal~~(s)~~ (indicative of conditions that could result in radiation exposure to control room emergency zone personnel), the ~~[MCREC] System automatically switches to the pressurization mode of operation~~ Control room emergency zone is automatically isolated to ~~prevent~~ minimize infiltration of contaminated air into the control room emergency zone. ~~A system of dampers isolates the control room, and a part of the recirculated air is routed through either of the two filter subsystems.~~ Insert BKGD-2. Outside air is taken in at the ~~normal~~ ventilation intake and is mixed with the recirculated air ~~before being passed~~ through ~~one of the charcoal adsorber filter subsystems~~ the AFU for removal of airborne radioactive particles.

The CREV ~~[MCREC]~~ 1 System is designed to maintain the control room emergency zone environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. ~~A single [MCREC] subsystem~~ The CREV System will pressurize the control room to about ~~[0.1]~~ 0.125 inches water gauge to ~~prevent~~ minimize infiltration of air from ~~surrounding buildings~~ adjacent zones. 1 CREV ~~[MCREC]~~ System operation in maintaining control 1

(continued)

2

Insert BKGD-1

The control room emergency zone served by the CREV System consists of the main control room, cable spreading room, auxiliary electric equipment room, computer room, and the Train B Heating Ventilation and Air Conditioning (HVAC) equipment enclosure.

2

Insert BKGD-2

Operator action is required within one hour after an accident to verify isolation and activate the air filtration unit (AFU) of the CREV System to pressurize the control room emergency zone.

BASES

1

BACKGROUND (continued)

9.4 u 6.4, 1

room habitability is discussed in the FSAR, ~~Chapters~~ ~~[6]~~ and ~~[9]~~, (Refs. 1 and 2, respectively). Sections 15.6.5 1 3

APPLICABLE SAFETY ANALYSES

CREV 1 Sections

The ability of the ~~[MCREC]~~ System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, ~~Chapters~~ ~~[6]~~ and ~~[15]~~ (Refs. 1 and 3, respectively). The ~~pressurization~~ ~~mode~~ of the ~~[MCREC] System~~ is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the FSAR, Section ~~[6.4.1.2.2]~~ (Ref. ~~6~~). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 3. ~~No single active or passive failure will cause the loss of outside or recirculated air from the control room.~~

emergency zone u .4 isolation 6.5 1 control room emergency zone u 1 1

CREV

The ~~[MCREC]~~ System satisfies Criterion 3 of ~~the NRC Policy Statement~~.

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

LCO

CREV 15

~~Two redundant subsystems of~~ the ~~[MCREC]~~ 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.

1

CREV

The ~~[MCREC]~~ System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE ~~in both subsystems~~. ~~A~~ ~~sub~~system is considered OPERABLE when its associated: The

Insert LCO

~~a.~~ ~~Fan~~ a booster fan is OPERABLE;

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

3

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

In addition, the control room emergency zone boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that the pressurization limit of SR 3.7.4.4 can be met. However, it is acceptable for access doors to be open for normal control room emergency zone entry and exit and not consider it to be a failure to meet the LCO.

(continued)

2

<u>Insert LCO</u>

a. AFU is OPERABLE,

b. Train B air handling unit (fan portion only) is OPERABLE, including the ductwork, to maintain air circulation to and from the control room emergency zone; and

c. Outside air ventilation intake is OPERABLE.

The AFU is considered OPERABLE when

All changes are 2 unless otherwise noted

BASES (continued)

APPLICABILITY

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

1 CREV

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the [MCREC] System 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 potential for draining the reactor vessel (OPDRVs)~~;~~ .

b. During CORE ALTERATIONS; and

a ~~c~~. During movement of irradiated fuel assemblies in the {secondary} containment~~.~~ ;

4

ACTIONS

1 CREV 5

A.1

With ~~one~~ the CREV [MCREC] ~~sub~~system inoperable in MODES 1, 2, or 3, the inoperable [MCREC] ~~sub~~system must be restored to OPERABLE status within 7 days. ~~With the unit in this condition, the remaining OPERABLE [MCREC] 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 reduced [MCREC] System capability.~~ 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

CREV 1

In MODE 1, 2, or 3, if the inoperable [MCREC] ~~sub~~system 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)

All changes are 2 unless otherwise noted

BASES

ACTIONS (continued)

C.1, C.2.1, C.2.2, and C.2.3 5

4

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.~~

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

Insert C.1a

4

With the CREV System inoperable,

5

Insert C.1b

During movement of irradiated fuel assemblies in the {secondary} containment, during CORE ALTERATIONS, or during OPDRVs, ~~if the inoperable [MCREC] subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE [MCREC] 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.~~ 1 5

~~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. 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~~s~~ must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Action~~s~~ must continue until the OPDRVs are suspended. 1 4

~~D.1~~

~~If both [MCREC] subsystems are inoperable in MODE 1, 2, or 3, the [MCREC] System may not be capable of performing~~ 5

(continued)

4 Insert C.1a

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.

4 Insert C.1b

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.

BASES

ACTIONS

5

D.1 (continued)

the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

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.

During movement of irradiated fuel assemblies in the [secondary] containment, during CORE ALTERATIONS, or during OPDRVs, with two [MCREC] 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, actions must be initiated immediately to suspend OPDVRs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

the CREV 2

This SR verifies that ~~a subs~~ystem 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 dries out any

from the control room 2

the 2

This SR includes initiating flow through the HEPA filters and charcoal adsorbers. 2

(continued)

for ≥ 10 continuous hours, during system operation 2

All changes are 2 unless otherwise noted

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1 (continued)

moisture that has accumulated in the charcoal as a result of 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.]~~ 1 Furthermore, the 31 day Frequency is based on the known reliability of the equipment ~~and the two subsystem redundancy available~~. 2

SR 3.7.4.2

1 CREV Specification 5.5.7," 4

This SR verifies that the required [MCREC] testing is performed in accordance with ~~the~~ [Ventilation Filter Testing Program (VFTP)]. The [MCREC] filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). 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]. 1 6

SR 3.7.4.3

CREV 1 System isolation dampers close 5

This SR verifies that on an actual or simulated initiation signal, ~~each~~ the [MCREC] ~~subsystem starts and operates~~. 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.~~ 6 5 6 Insert SR 3.7.4.3

SR 3.7.4.4

emergency zone

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 ~~(the turbine building)~~, is periodically tested to verify proper function of the [MCREC] System. During the emergency ~~mode of operation~~, the [MCREC] System is designed to slightly pressurize the control room ≥ ~~[0.1]~~ inches water gauge positive pressure with respect to the ~~turbine building~~ to 1 CREV pressurization

emergency zone 0.125 adjacent areas

(continued)

6

<u>Insert SR 3.7.4.3</u>

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.

All changes are 2 unless otherwise noted

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.4 (continued)

minimize ~~prevent~~ unfiltered inleakage. The CREV ~~[MCREC]~~ System is designed to maintain this positive pressure at a flow rate of ≤ 2000 ~~[400]~~ cfm to the control room in the pressurization mode. The Frequency of 24 ~~[18]~~ months ~~on a STAGGERED TEST BASIS~~ 5 is consistent with emergency zone ~~industry practice~~ and other filtration systems SRs.

REFERENCES

1. UFSAR, Section ~~Chapter~~ 6.4.
2. UFSAR, Section ~~Chapter~~ 9.4.
3. UFSAR, Section ~~Chapter~~ 6.5.
4. ~~FSAR, Section [6.4.1.2.2].~~
4. ~~5.~~ Regulatory Guide 1.52, Revision 2, March 1978.

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

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. 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 3.7.4.4 (i.e., leakage can occur as long as a 0.125 inch pressure is maintained in the control room). Also, an allowance to open control room emergency zone access doors for entry and exit has been added.

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 reflect those changes made to the Specification.

6. Changes have been made to more closely match the LCO requirements.

All changes are 2 unless otherwise indicated

1

B 3.7 PLANT SYSTEMS

Emergency Ventilation

B 3.7.5 [Control Room Air Conditioning (AC)] System

portion of the control room area Heating, Ventilation, and Air Conditioning (HVAC) System (hereafter referred to as the Control Room Emergency Ventilation AC)

BASES

BACKGROUND

Emergency Ventilation

The [Control Room AC] System provides temperature control for the control room following isolation of the control room. emergency zone

1 Emergency Ventilation

Insert BKGD-1

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.~~

10

The [Control Room AC] System is designed to provide a controlled environment under both normal and accident conditions. ~~A single subsystem~~ The system provides the required temperature control to maintain a suitable control room emergency zone environment for a sustained occupancy of 12 persons. The design conditions for the habitability of the control room ~~environment~~ emergency zone are 75°F 70°F to 80 and 50% relative humidity. The [Control Room AC] System operation in maintaining the control room temperature is discussed in the FSAR, Section [6.4] (Ref. 1). 11 1 2 emergency zone

APPLICABLE SAFETY ANALYSES

The design basis of the [Control Room AC] System is to maintain the control room emergency zone temperature for a 30 day continuous occupancy following isolation of the control room emergency zone.

1 Emergency Ventilation

~~The [Control Room AC] 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 emergency zone. ~~A single 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.~~ Insert ASA-1 The [Control Room AC] System is designed in accordance with Seismic Category I requirements, except for a portion of the return ductwork 2. The [Control Room AC] System is capable of removing sensible and latent heat loads from the control room emergency zone 2, including consideration of equipment

(continued)

2

Insert BKGD-1

is a single zone system that services only those rooms that are a part of the control room emergency zone. The system provides cooling of the recirculated and outside air makeup for the control room emergency zone. The Control Room Emergency Ventilation AC System, addressed by this Specification, consists of the Train B air handling unit (AHU), ductwork, dampers, refrigeration condensing unit, and instrumentation and controls to provide for control room emergency zone temperature control.

2

Insert ASA-1

The safety related Control Room Emergency Ventilation AC System (Train B HVAC) is powered from diesel generator supported switchgear. Train B Control Room HVAC is normally in the standby condition and is used for accident mitigation. Train A Control Room HVAC is nonsafety related and is in operation during normal conditions. The Train B refrigeration condensing unit, normally served by the Service Water System, can be provided with cooling water from either the Unit 1 or 2 Residual Heat Removal Service Water (RHRSW) System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

Emergency Ventilation 1

The ~~Control Room AC~~ System satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) 2

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~~. (15) Total system failure could result in the equipment operating temperature exceeding limits.

Emergency Ventilation 1 emergency zone 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. Insert LCO 3

Emergency Ventilation 1

APPLICABILITY

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. emergency zone 2

1 Emergency Ventilation

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:

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 ~~secondary~~ containment~~.~~ ; 1

4

(continued)

3

Insert LCO

In addition, during conditions in MODES other than MODES 1, 2, and 3 when the Control Room Emergency Ventilation AC System is required to be OPERABLE (e.g., during CORE ALTERATIONS), the necessary portions of the RHRSW System and Ultimate Heat Sink capable of providing cooling to the refrigeration condensing unit are part of the OPERABILITY requirements covered by this LCO.

BASES (continued)

ACTIONS

A.1

Emergency Ventilation 1

in MODES 1, 2, or 3 5

the

With ~~one~~ [control room AC] ~~sub~~system 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 ~~safety and~~ nonsafety cooling methods. 2

emergency zone

B.1 and B.2

Emergency Ventilation 1

2 In MODE 1, 2, or 3, if the inoperable [control room AC] ~~sub~~system 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

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel 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.~~

Insert C.1a

4

With the Control Room Emergency Ventilation AC System inoperable 5

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. This action ensures that the remaining subsystem is OPERABLE,~~

Insert C.1b

(continued)

4

Insert C.1a

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.

4

Insert C.1b

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.

 1

BASES

ACTIONS

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

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. 5

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, 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. 1 4

D.1

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. 5

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 a sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the [secondary] containment, during CORE ALTERATIONS, or during OPDRVs, with two [control room AC] subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might

(continued)

Emergency Ventilation 1

BASES

ACTIONS

5

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

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, 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

2 emergency zone

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]~~ 24 month Frequency is appropriate since significant degradation of the [Control Room AC] System is not expected over this time period. 1

Emergency Ventilation

2 U

REFERENCES

1. [FSAR, Section [6.4].]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.5 - CONTROL ROOM EMERGENCY VENTILATION
AIR CONDITIONING (AC) SYSTEM

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

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 LCO Section of the Bases has been revised to add clarification regarding Operability requirements for the RHRSW System and Ultimate Heat Sink during Modes 4 and 5, since the ITS does not have LCOs for the RHRSW System and Ultimate Heat Sink in these Modes.

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 reflect those changes made to the Specification.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

main [1]

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 moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture 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, discussed in ~~the FSAR, Section [15.1.35] (Ref. 1)~~ Reference 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) ~~or the NRC staff approved licensing basis~~. [2] [3]

The main condenser offgas limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). [2]

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 established consistent with

(continued)

BASES

251,100 [3] 2511 [3]

LCO (continued)	this requirement (~~{2436}~~ MWt x 100 μCi/MWt-second = ~~{240}~~ μCi/second). (A)
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. [3] In MODES 4 and 5, steam is not being exhausted to the main condenser and the requirements are not applicable. [1] main
ACTIONS	<u>A.1</u> 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, 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. <u>B.1, B.2, B.3.1, and B.3.2</u> [1] significant If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, {all main steam lines or} the SJAE must be isolated. [3] This isolates the Main Condenser Offgas System from ~~the~~ source(s) of ~~the~~ radioactive steam. [1] 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

(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 3.7.6.1

2 representative

(taken at the recombiner outlet or the SJAE outlet if the recombiner is bypassed)

2 as indicated by the radiation monitors located prior to the offgas holdup line

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.

m

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. 3

REFERENCES

1. FSAR, Section [15.1.35].
2. 10 CFR 100.

Letter E-DAS-023-00 from D.A. Studky (Scientech-NUS) to R. Tsai (ComEd), dated January 24, 2000

3

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

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.

5

TSTF-319 changes not adopted

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 ~~[25]~~ 40% 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 nine ~~three~~ valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve ~~bypass valve chest~~. Each of these ~~three~~ nine valves is 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 [7.7.4] (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct 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 sequentially. When the bypass valves open, the steam flows from the ~~bypass chest~~ main steam equalizing header to the bypass manifold, through ~~connecting piping~~ its bypass line, to ~~the pressure breakdown assemblies~~ the bypass valve, where a ~~series of~~ orifices ~~are used to~~ further reduce the steam pressure before the steam enters the condenser.

1 2 2 3 2 2 2 2 1 2 5 4 2

15.2.3.2, 15.2.2.2, and 15.1.2

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the turbine trip, turbine generator load rejection ~~transient~~ and feedwater controller failure transients as discussed in the UFSAR, Sections ~~[15.1.7]~~ (Refs. 2, 3, and 4, 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.

5 4 2 2

The Main Turbine Bypass System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii).

2

(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, so 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 this LCO to be met.}~~ The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. 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 analyses (Ref. 2, 3, and 4). 1 2

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 generator load rejection transient, turbine trip, and feedwater controller failure. 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 at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level. 2 3

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), ~~or~~ 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. 4 1

(continued)

BASES

ACTIONS (continued)

B.1

and [4] turbine trip, and feedwater controller failure transients [2]

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 generator load rejection ~~transient~~. 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

[5] 92

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 ~~92~~ day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the ~~92~~ day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

92 [5]

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 system initiation signals, the valves will actuate to their required position. 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.

[1] 24

[1] 24

[2] Therefore, the Frequency was concluded to be

that these components usually pass the SR when performed at [2]

(continued)

BASES

as defined in the transient analysis inputs for the cycle 2

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in ~~[unit specific documentation]~~ the Technical Requirements Manual (Ref. 5). The [18] 24 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 [18] 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

2 1 2 1 2

REFERENCES

1. UFSAR, Section ~~[7.7.4]~~ 2.3.2 1

2. UFSAR, Section [15.1.2].

3. UFSAR, Section 15.2.2.2
4. UFSAR, Section 15.1.2

5. Technical Requirements Manual

2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.7 - MAIN TURBINE BYPASS SYSTEM

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

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. This LCO is needed to ensure the MCPR limit is not exceeded. The cladding 1% plastic strain limit is an LHGR concern, not a MCPR concern. Therefore, this statement has been deleted. In addition, the statement that refers to the APHLGR Bases has also been deleted since this LCO is only concerned with MCPR.

4. Typographical/grammatical error corrected.

5. Changes have been made consistent with changes made to the Specification.

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 FSAR, Section [ ] (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section [15.1.4] (Ref. 2).

1 a 9.1.2 2 Reference 2

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. 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% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800 (Ref. 4). 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).

and less than the 10 CFR 50, Appendix A, GDC 19 limits (Ref. 6)

and 5 1 7

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 no more severe than those of the fuel handling accident over the reactor core, as discussed in the FSAR, Section [9.1.2.2.2] (Ref. 6).~~ 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.

1

The spent fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

TSTF-139 Change not adopted 3

10 CFR 50.36 (c)(2)(ii) 1

(continued)

BASES (continued)

LCO

The specified water level preserves the assumptions 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 during movement of irradiated fuel assemblies in the spent fuel storage pool, since the potential for a release of fission products exists.

4

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,

---

ACTIONS

A.1

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

5

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 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.~~

5

Insert A.1

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. 4
Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. 4
This effectively precludes a spent fuel handling accident from occurring.

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 all water level changes are controlled by unit procedures.

(continued)

5

Insert 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)

1 9.1.2 2

REFERENCES

u 1. FSAR, Section ~~1~~ ~~7~~.
2. ~~FSAR, Section [15.1.4].~~
5 ~~3.~~ NUREG-0800, Section 15.7.4, Revision 1, July 1981.
~~4.~~ 10 CFR 100.
1 7 ~~5.~~ Regulatory Guide 1.25, March 1972.
~~6. FSAR, Section [9.1.2.2.2].~~

6. 10 CFR 50, Appendix A, GDC 19. 1

5. NUREG-0800, Section 6.4, Revision 2, July 1981. 1

Letter E-DAS-00-0048 from D. A. Studley (Scientech) to Robert Tsai (ComEd), "Submittal of Calculation in Support of Improved Tech. Spec. Program," dated February 17, 2000 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.8 - SPENT FUEL STORAGE POOL WATER LEVEL

1. 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.

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

3. 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.

4. Changes have been made to be consistent with 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.

1

B 3.7 PLANT SYSTEMS

B 3.7.9 Safe Shutdown Makeup Pump (SSMP) System

BASES

BACKGROUND

The SSMP System is designed to operate manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the Feedwater System to provide makeup water to the RPV. Under these conditions, the High Pressure Coolant Injection (HPCI), the Reactor Core Isolation Cooling (RCIC) and the SSMP Systems perform similar functions. The SSMP System design requirements ensure that the criteria of 10 CFR 50, Appendix R, Section III.G (Ref. 1) are satisfied.

The SSMP System (Ref. 2) consists of a motor driven pump unit, as well as piping and valves to transfer water from the suction source to the RPV through the Feedwater System line via the HPCI System line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the contaminated condensate storage tanks (CCSTs). An alternate source of makeup water is available from the Fire Protection System header in the turbine building.

The SSMP System is designed to provide makeup water for a wide range of reactor pressures, 150 psig to 1120 psig. The SSMP System injection valves are interlocked to allow injection into only one RPV at a time since the system is common to Units 1 and 2. Electric power for the system is normally fed from Division 2 of Unit 1, however an alternate source is available from Division 2 of Unit 2.

The SSMP System does not include a minimum flow line therefore a trip will occur if the flow control valve closes. This will prevent pump damage due to overheating in low flow conditions. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the SSMP System discharge piping is kept full of water. The SSMP System is normally aligned to the CCST. The height of water in the CCST is sufficient to maintain the piping full of water up to the unit injection valves. The feedwater header pressure ensures the remaining portion of the SSMP System discharge line is full of water. Therefore, the SSMP System does not require a "keep fill" system.

(continued)

II

BASES (continued)

APPLICABLE SAFETY ANALYSES	The function of the SSMP System is to respond to transient events by providing makeup coolant to the reactor. The SSMP System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for SSMP System operation. The system provides a backup to the Unit 1 and 2 RCIC Systems to satisfy the requirements of criteria of 10 CFR 50, Appendix R, Section III.G (Ref. 1). Based on its contribution to the reduction of overall plant risk, the system satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii) and is therefore included in the Technical Specifications.
LCO	The OPERABILITY of the SSMP System ensures sufficient reactor water makeup is provided in the event of RPV isolation accompanied by a loss of feedwater flow. The SSMP System has sufficient capacity for maintaining RPV inventory during an isolation event.
APPLICABILITY	The SSMP System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since the SSMP System provides a non-Emergency Core Cooling System water source for makeup when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, the SSMP System is not required to be OPERABLE since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the RPV and since the plant risk associated with fire is also reduced during these MODES.
ACTIONS	A.1 and A.2 If the SSMP System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, the SSMP System must be restored to OPERABLE status within 14 days. In this Condition, loss of the SSMP System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the RCIC and HPCI System are required to be OPERABLE. The 14 day Completion Time is consistent with the Completion Time for a RCIC System inoperability, because of similar functions of the RCIC and

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

SSMP Systems. The same Completion Time for RCIC is also applied to the SSMP System since the SSMP System and the RCIC System have the same post-fire shutdown functionality goals to provide reactor water makeup (Ref. 3).

B.1 and B.2

If the SSMP System cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SSMP System flow path provides assurance that the proper flow path will exist for SSMP System 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. 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. For the SSMP System, this SR also includes the flow controller position, since it controls the pump discharge flow control valve position.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the SSMP System. This Frequency has been shown to be acceptable through operating experience.

(continued)

1 Insert ITS B 3.7.9 (continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.7.9.2 The SSMP System pump flow rate ensures that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow test is performed by utilizing the full flow test line to the CCST. The requirements include verifying that the pump discharge pressure is greater than or equal to a pressure that would produce the desired injection flow including allowances for the flow and elevation head losses of the injection line. This provides adequate assurance of SSMP System OPERABILITY based on performance at nominal conditions. A 92 day Frequency for SR 3.7.9.2 is consistent with the Inservice Testing Program requirements.
REFERENCES	1. 10 CFR 50, Appendix R, Section III.G. 2. UFSAR, Section 5.4.6.5. 3. Letter from J.A. Grobe (NRC) to O.D. Kingsley (ComEd), "NRC Inspection Report 50-254/98011 (DRS); 50-265/98011 (DRS)," dated July 2, 1998.

Insert Page B 3.7-43

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.9 - SAFE SHUTDOWN MAKEUP PUMP SYSTEM

1. This proposed Bases has been added to match the addition of the Specification.

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 Quad Cities 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-1433, 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-1433, 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.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.1 - RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

There were no plant specific less restrictive changes identified for this Specification.

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.3 - ULTIMATE HEAT SINK (UHS)

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.4 - CONTROL ROOM EMERGENCY VENTILATION (CREV) 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 phrase "actual or," in reference to the simulated automatic isolation test 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.

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 deletes the Control Room Manual Initiation Function requirements from the Technical Specifications. The Control Room Manual Initiation Function is not assumed to be the initiator of analyzed events and is not assumed to mitigate accident or transient events. The automatic isolation Functions are credited to provide the appropriate isolation signal to the control room isolation dampers. The requirements for the automatic portion of the control room isolation logic are being retained in ITS 3.7.4, Control Room Emergency Ventilation (CREV) System, and ITS 3.3.7.1, CREV Isolation Instrumentation. 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. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not significantly reduce a margin of safety because it has no impact on any safety analysis assumptions. The automatic isolation Functions are credited to provide the appropriate isolation signal to the control room isolation dampers. The requirements for the automatic portion of the control room isolation logic are being retained in ITS 3.7.4, CREV System, and ITS 3.3.7.1, CREV Isolation Instrumentation. Therefore, no significant reduction in a margin of safety will be permitted.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.5 - CONTROL ROOM EMERGENCY VENTILATION
AIR CONDITIONING (AC) SYSTEM

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?

   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 4 hours to isolate 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 additional 4 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 significantly increase the probability or consequences of a previously analyzed accident.

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?

   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 limit, the ability to isolate the main steam lines or SJAE manually if an event occurs, and the minimization of plant transients. The proposed 4 hour extension will allow the MSIVs or the SJAE to be isolated in an orderly manner. As a result, the potential for human error and the risk associated with challenging plant systems will be reduced. Any reduction in a margin of safety will be insignificant and offset by the benefit gained from avoiding potential plant transients. 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 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 8 hours. The method of placing the plant outside the Applicability of the Specification and the associated 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.

L.2 CHANGE (continued)

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 8 hours. 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. The proposed alternative action may help avoid a plant transient caused by isolating the main steam isolation valves in the 8 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.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 will require the determination of gross gamma activity rate of noble gases to be within limits 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 instead of any activity increase > 50%. 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. The proposed Surveillance Frequency is still considered to be adequate to determine whether the gross gamma activity rate of the noble gases is within limits. Main condenser offgas activity levels are expected to increase as a result of THERMAL POWER level increases. However, the increase is expected to stabilize. Offgas activity increases due to changes in THERMAL POWER are not necessarily indicative of fuel failure. The intent of the Surveillance is to trend and determine the extent of fuel failure so that alternative plant operating strategies are taken. This change will therefore reduce the number of times the test must be performed when the main condenser offgas activity is expected to change (i.e., during THERMAL POWER increases) and not necessarily be indicative of fuel failure and only require it to be performed at this 4 hour Frequency when the increase in the activity level is not expected, and therefore fuel failure may exist. The proposed Surveillance Frequencies are considered to be sufficient in determining whether the LCO is being met and to trend the activity when there is an unexpected increase. Therefore, this change in the Surveillance Frequency does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not 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.

L.3 CHANGE (continued)

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 gross gamma activity rate is still required to be within limit. This change will require the determination of gross gamma activity rate of noble gases to be within limits once within 4 hours after a $\geq$ 50% increase in the normal steady state fission gas release after factoring out increases due to changes in THERMAL POWER instead of any activity increase $\geq$ 50%. The proposed Surveillance Frequency is still considered to be adequate to determine whether the gross gamma activity rate of noble gases is within limits. Main condenser offgas activity levels are expected to increase as a result of THERMAL POWER level increases. However, the increase is expected to stabilize. Offgas activity increases due to changes in THERMAL POWER are not necessarily indicative of fuel failures. The intent of the Surveillance is to trend and determine the extent of fuel failure so that alternative plant operating strategies are taken. This change will therefore reduce the number of times the test must be performed when the main condenser offgas activity is expected to change (i.e., during THERMAL POWER increases) and not necessarily be indicative of fuel failure and only require it to be performed at this 4 hour Frequency when the increase in the activity level is not expected and therefore fuel failure may exist. The proposed Surveillance Frequencies are considered to be sufficient for determining whether the LCO is being met and to trend the activity when there is an unexpected increase. Therefore, this change does not result in 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 determination of the gross radioactivity rate of noble gases has been postponed until 31 days after any main steam line is not isolated and a SJAE is in operation. The offgas gross radioactivity rate Surveillance is not considered to be an initiator of any accident. Therefore, this change does not increase the probability of an accident previously evaluated. With all main steam lines isolated or the SJAE not in service, this Surveillance provides no meaningful information. The gross radioactivity rates are only expected to be high when operating close to full power conditions where the main steam lines are open and a SJAE is in operation. With the main steam lines isolated or a SJAE not in service the reactor power is low and thus the resulting offgas activity is insignificant. During a reactor startup (when the main steam lines are opened and the SJAE placed in service) the gross radioactivity rate of noble gases should be considered to be nearly equivalent to the levels before shutting down. In addition, if offgas activity exceeds the setpoint of the offgas radiation monitors, alarms would annunciate and operators will be required to take action according to procedures. The system is designed to automatically isolate within 15 minutes if the activity remains above the setpoint. Therefore, entering the conditions of the Applicability without determining the gross radioactivity rate is acceptable and does not increase 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 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 determination of the gross radioactivity rate of noble gases has been postponed until 31 days after any main steam line is not isolated and a SJAE is in operation. The offgas gross radioactivity rate Surveillance is not considered to be an initiator of any accident. Therefore, this change does not increase the probability of an accident previously evaluated. With all main steam lines isolated or the SJAE not in service, this Surveillance provides no meaningful information. The gross radioactivity rates are only expected to be high when operating close to full power conditions where the main steam lines are open and a SJAE is in operation. With the main steam lines isolated or the SJAE not in service the reactor power is low and thus the resulting offgas activity is insignificant. During a reactor startup (when the main steam lines are opened and the SJAE placed in service) the gross radioactivity rate of noble gases should be considered to be nearly equivalent to the levels before shutting down. In addition, if offgas activity exceeds the setpoint of the offgas radiation monitors, alarms would annunciate and operators will be required to take action according to procedures. The system is designed to automatically isolate within 15 minutes if the activity remains above the setpoint. Therefore, entering the conditions of the Applicability without determining the gross radioactivity rate is acceptable and 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
ITS: 3.7.8 - SPENT FUEL STORAGE POOL WATER LEVEL

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.7.9 - SAFE SHUTDOWN MAKEUP PUMP SYSTEM

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.8.E - FLOOD PROTECTION

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.8.F - SNUBBERS

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.8.G - SEALED SOURCE CONTAMINATION

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.