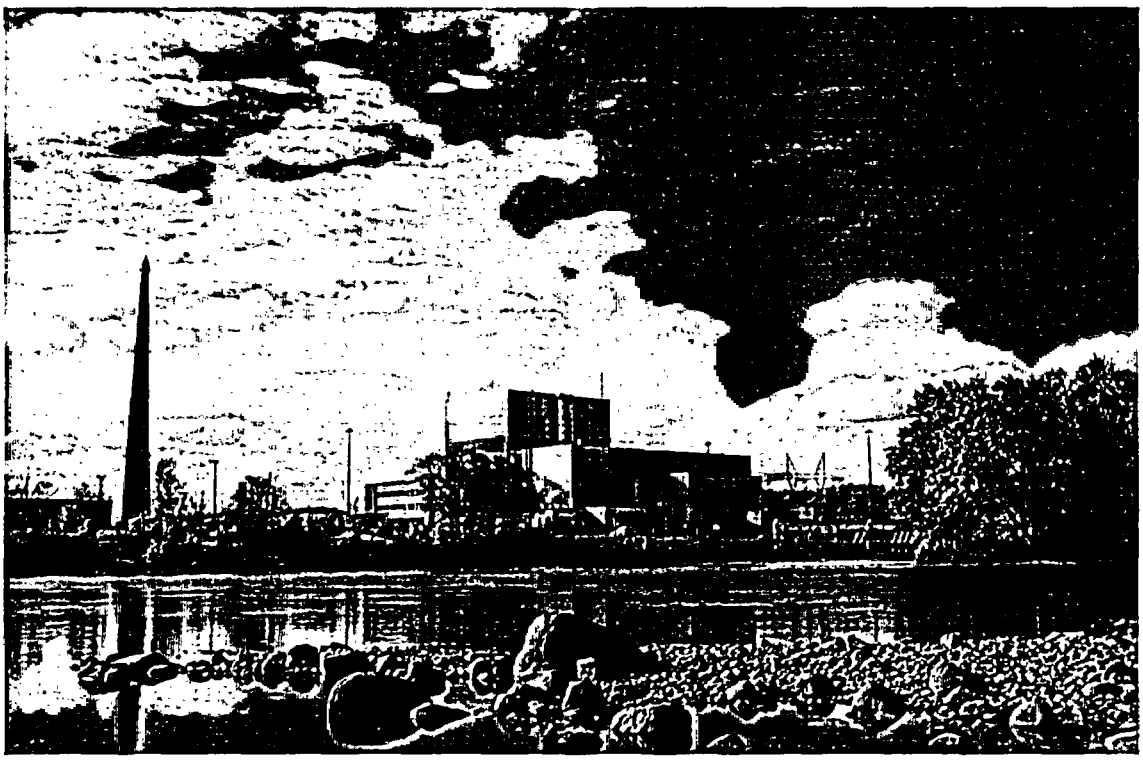
IMPROVED TECHNICAL SPECIFICATIONS



MONTICELLO NUCLEAR GENERATING PLANT

VOLUME 12

ITS Section 3.7,
Plant Systems

NMC
Committed to Nuclear Excellence

ATTACHMENT 1

VOLUME 12

MONTICELLO IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.7 PLANT SYSTEMS

Revision 0

LIST OF ATTACHMENTS

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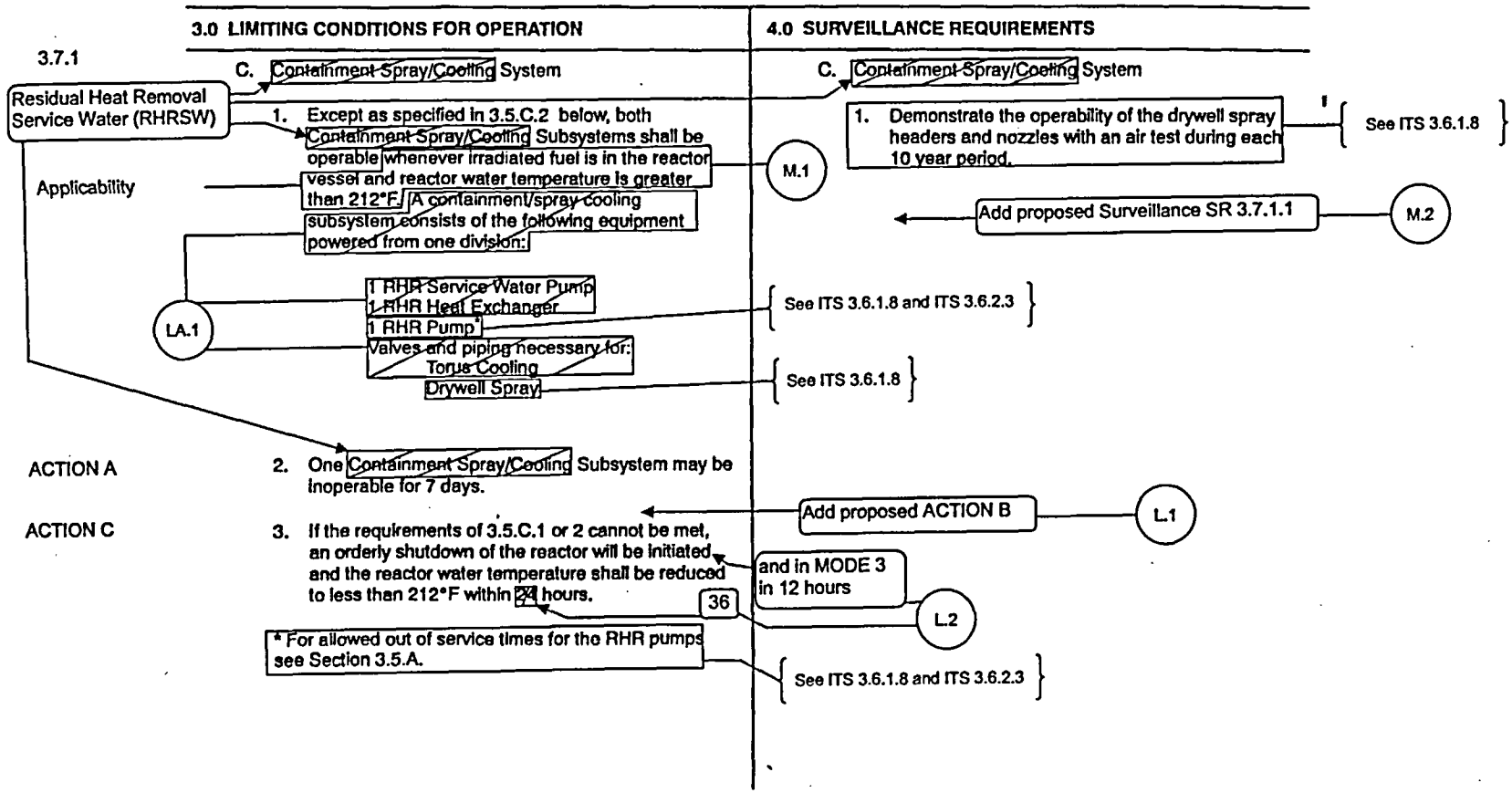
ATTACHMENT 1

ITS 3.7.1, Residual Heat Removal Service Water (RHRSW) System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS



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

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.5.C.1 is applicable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. ITS LCO 3.7.1 is applicable in MODES 1, 2, and 3. This changes the CTS by requiring two RHRSW subsystems to be OPERABLE in MODE 2 when reactor water temperature is less than or equal to 212°F.

The purpose of CTS 3.5.C.1 is to ensure the RHRSW subsystems are OPERABLE to mitigate the consequences of a design basis accident. The RHRSW subsystems are required to be OPERABLE during MODES 1, 2, and 3 when there is considerable energy in the reactor core and a DBA could cause a significant heat up of the suppression pool. In MODES 1 and 3, the reactor coolant temperature will always be above 212°F. In MODE 2, the reactor coolant temperature may be less than or equal to 212°F when the reactor is subcritical but control rods are withdrawn. Therefore, it is necessary and acceptable to require the RHRSW subsystems to be OPERABLE. This change is designated as more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

M.2 Currently, the CTS does not provide any specific Surveillances to verify OPERABILITY of the RHRSW subsystems. ITS SR 3.7.1.1 requires verification that each RHRSW subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position every 31 days. This changes the CTS by adding this Surveillance Requirement to the Technical Specifications.

The purpose of ITS SR 3.7.1.1 is to provide assurance that the proper flow paths will exist for RHRSW System operation. This change is acceptable because it provides additional assurance that the RHRSW System will be capable of performing its function. This change is designated as more restrictive because it adds Surveillance Requirements to the CTS.

RELOCATED SPECIFICATIONS

None

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

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.5.C.1 states that an RHRSW subsystem consists of the following equipment powered from one division: 1 RHR Heat Exchanger, 1 RHR Service Water Pump, and valves and piping necessary for torus cooling. ITS 3.7.1 requires two RHRSW subsystems to be OPERABLE, but the details of what constitutes an OPERABLE subsystem are moved to the Bases. This changes the CTS by moving the details of what constitutes an OPERABLE subsystem to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for two RHR service water subsystems to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5 of the ITS. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* When two RHRSW subsystems are inoperable, a unit shutdown is required by CTS 3.5.C.3; no time is provided to restore a subsystem prior to requiring the unit shutdown. With two RHRSW subsystems inoperable, ITS 3.7.1 ACTION B will allow 8 hours to restore one inoperable RHRSW subsystem prior to requiring a unit shutdown. This changes the CTS by allowing 8 hours to restore one of two inoperable RHRSW subsystems prior to requiring a unit shutdown.

The purpose of CTS 3.5.C is to require sufficient containment cooling to ensure the primary containment conditions for the safety analyses are met. The proposed 8 hour Completion Time is acceptable since an immediate shutdown has the potential to result in a unit scram and discharge of steam to the suppression pool, when both RHRSW subsystems are inoperable and incapable of removing the generated heat. The 8 hours provides some time to restore one of the subsystems prior to requiring a shutdown (thus precluding the potential problem described above), yet is short enough that it does not significantly increase the probability of an accident to occur during this additional time. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 3 - Relaxation of Completion Time)* CTS 3.5.C.3 requires the unit to be shutdown and reactor water temperature reduced to less than 212°F within 24 hours if the requirements of CTS 3.5.C.1 or CTS 3.5.C.2 are not met. ITS 3.7.1 ACTION C requires the reactor be in MODE 3 in 12 hours and in MODE 4 in 36 hours. This changes the CTS by requiring the unit to be in

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

MODE 3 in 12 hours and by extending the time to reduce reactor water temperature to < 212°F (i.e., MODE 4) from 24 hours to 36 hours.

The purpose of CTS 3.5.C.3 is to place the unit outside the Applicability of the Specification within a reasonable amount of time. This change is acceptable because the Completion Time is consistent with the safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. 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. This change is also acceptable because it requires the unit to be in an intermediate condition (MODE 3) sooner than is currently required (12 hours versus 24 hours). This portion of the change reduces the amount of time the unit would be allowed to continue to operate in MODES 1 and 2 once the condition is identified. The consequences of a pressurization event are significantly reduced when the reactor is shutdown and a controlled cooldown is already in progress. This change is designated as less restrictive because additional time is allowed to place the unit outside the LCO Applicability than is allowed in the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.5.C 3.7.1 Residual Heat Removal Service Water (RHRSW) System

3.5.C.1 LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

3.5.C.1 APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One RHRSW pump inoperable.	A.1 Restore RHRSW pump to OPERABLE status.	30 days	
B. One RHRSW pump in each subsystem inoperable.	B.1 Restore one RHRSW pump to OPERABLE status.	7 days	1
3.5.C.2 A. One RHRSW subsystem inoperable for reasons other than Condition A.	A.1 ----------NOTE---------- 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 RHRSW System. ----------		1 2 3
	Restore RHRSW subsystem to OPERABLE status.	7 days	

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
DOC L.1	B D. Both RHRSW subsystems inoperable for reasons other than Condition B.	D.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of 7 LCO 3.4.8 for RHR shutdown cooling made inoperable by RHRSW System. ----------		1 2 3
		Restore one RHRSW subsystem to OPERABLE status.	[8] hours	3
3.5.C.3	C E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND	12 hours	1
	C	E.2 Be in MODE 4.	36 hours	1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
DOC M.2	SR 3.7.1.1	Verify each RHRSW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.1, RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

1. ISTS 3.7.1 ACTIONS A and B have been deleted because they are not applicable to Monticello. The Monticello USAR, Section 5.2.3.2.3 (for the long term primary containment response after a design basis loss of coolant accident analysis) only credits one RHRSW pump in one subsystem (i.e., a flow rate of 3500 gpm, which is the flow rate for one RHRSW pump). The following requirements have been renumbered, where applicable, to reflect this deletion.

2. Changes have been made to reflect changes made to other Specifications.

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

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

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. (3)

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. (1) (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. (1) (2)

The system is initiated manually from the control room. If operating during a loss of coolant accident (LOCA), the system is automatically tripped to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time 10 minutes after the LOCA, or manually started any time the LOCA signal is manually overridden or clears. (2) (2)

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

discussed in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

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] gpm per pump with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are [206.4] °F and [36.59] psig, respectively, well below the design temperature of [340] °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

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."

INSERT 1

An adequate suction source is not addressed in this LCO since the minimum net positive suction head ([59] ft mean sea level in the pump well) is bounded by the plant service water pump requirements (LCO 3.7.2, "[Plant Service Water (PSW)] System and [Ultimate Heat Sink (UHS)]").

2 **INSERT 1**

may be opened since the cross tie valve is only 1 inch in size and the RHRSW pump flow requirements (tested per the requirements of the Inservice Testing Program) account for the flow through the open cross tie valve.

BASES

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. (3) (7)

and therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. The LCOs of the system supported by the RHRSW System will govern RHRSW System OPERABILITY requirements in MODES 4 and 5

In MODES 4 and 5, the OPERABILITY requirements of the RHRSW System are determined by the systems it supports. (2)

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. (3)

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.

BASES

ACTIONS (continued)

A

C.1

3

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 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 inoperable [RHR shutdown cooling]. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

3

1

B

D.1

3

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.

3

2

7 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 inoperable [RHR shutdown cooling]. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

3

1

BASES

ACTIONS (continued)

C

F.1 and F.2 (3)

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

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

REFERENCES

U

1. FSAR, Section 9.2.7. 10.4.2 (1) (2)
2. FSAR, Chapter 6. 5.2.3 (1) (2)
3. ~~FSAR, Chapter 15.~~ (2)
4. ~~FSAR, Section 6.2.1.4.3.~~ (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.1 BASES, RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) 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. Changes are made to reflect those changes made to the Specifications.

4. These punctuation corrections have been made consistent with the Writers Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

5. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.1, RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) SYSTEM

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 2

ITS 3.7.2, Emergency Service Water (ESW) System and Ultimate Heat Sink (UHS)

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

Add proposed ITS 3.7.2

M.1

DISCUSSION OF CHANGES
ITS 3.7.2, EMERGENCY SERVICE WATER (ESW) AND ULTIMATE HEAT SINK (UHS)

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M.1 The CTS does not have any specific requirements for the Emergency Service Water (ESW) System or ultimate heat sink (UHS). The ESW System and UHS requirements are governed by the systems they support. ITS LCO 3.7.2 requires two ESW subsystems and the UHS to be OPERABLE. Appropriate ACTIONS and Surveillance Requirements are also provided. This changes the CTS by incorporating the requirements of ITS 3.7.2.

The ESW System and UHS are necessary to support the equipment required for long term cooling of the reactor containment following a Design Basis Accident (DBA). The requirement to maintain two ESW subsystems and the UHS OPERABLE assures adequate cooling capacity is available for the removal of heat from equipment such as residual heat removal and core spray pump coolers, and ECCS room coolers required for safe shutdown following a DBA. The ESW System, together with the UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). This change is acceptable because the ability of the ESW System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analysis evaluated for a DBA. The ITS restoration actions for when an ESW subsystem or the UHS is inoperable is consistent with or more restrictive than the Technical Specification Systems they support. In addition, specific temperature and level requirements are now specified, as well as Surveillance Requirements. This change is designated as more restrictive because it adds new requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

[PSW] System and [UHS] 3.7.2 (1)

E

CTS

3.7 PLANT SYSTEMS

Emergency

3.7.2 [Plant] Service Water ([E][PSW]) System and [Ultimate Heat Sink (UHS)] (1)

E

DOC M.1 LCO 3.7.2 Two [E][PSW] subsystems and [UHS] shall be OPERABLE. (1)

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 (2)
C. [One or more cooling towers with one cooling tower fan inoperable.	C.1 Restore cooling tower fan(s) to OPERABLE status.	7 days] (3)
-----REVIEWER'S NOTE----- The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit. ---------- D. [Water temperature of the UHS > [90]°F and ≤ [ ]°F.	D.1 Verify water temperature of the UHS is ≤ [90]°F averaged over the previous 24 hour period.	Once per hour] (4)

[PSW] System and [UHS] 3.7.2 (1)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M.1 (A) (E) E. One [PSW] subsystem inoperable for reasons other than Condition[s] A [and C].	(A) E.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 (7) LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for [RHR shutdown cooling] made (E) inoperable by [PSW] ---------- (E) Restore the [PSW] subsystem to OPERABLE status.	(7 days) 72 hours

(4) (1) (5) (5) (6) (1) (1) (7)

[PSW] System and [UHS] 3.7.2 (1)

CTS

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	[Verify the water level of each [PSW] cooling tower basin is ≥ [ ] ft.	24 hours] (3)
DOC M.1 SR 3.7.2.[2] [1]	[Verify the water level [in each PSW pump well of the intake structure] is ≥ [60.1] [899] ft [mean sea level].	24 hours [(3) (1)
DOC M.1 SR 3.7.2.[3] [2]	[Verify the average water temperature of [UHS] is ≤ [/] [90] °F.	24 hours [(3) (1)
SR 3.7.2.4	[Operate each [PSW] cooling tower fan for ≥ [15] minutes.	31 days] (3)

[PSW] System and [UHS] 3.7.2 (E) (1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
DOC M.1	SR 3.7.2.5 (3)	----------NOTE---------- (3) (E) Isolation of flow to individual components does not render [PSW] System inoperable. (1) ----------	
	(E) 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 (1) (8)	
DOC M.1	SR 3.7.2.6 (4) (E) Verify each [PSW] subsystem actuates on an actual or simulated initiation signal.	(24) [18] months	(3) (1)

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JUSTIFICATION FOR DEVIATIONS
ITS 3.7.2, EMERGENCY SERVICE WATER (ESW) AND ULTIMATE HEAT SINK (UHS)

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

2. ISTS 3.7.2 ACTIONS A and B have been deleted because they are not applicable to Monticello. The Monticello ESW System design contains only one ESW pump per subsystem.

3. The bracketed ISTS 3.7.2 ACTION C has been deleted because it is not applicable to Monticello. Monticello does not include cooling towers for OPERABILITY of the ESW System or UHS.

4. The bracketed ISTS ACTION D has been deleted as it is not part of the plant specific ITS. The 90°F limit in ITS SR 3.7.2.2 is the maximum water temperature assumed in the accident analysis. Therefore, when 90°F is exceeded, the UHS is inoperable and ISTS 3.7.2 ACTION F (ITS 3.7.2 ACTION B) must be taken. The following requirements have been renumbered to reflect this and other ACTION deletions.

5. ISTS 3.7.2 Required Action E.1 Note 1 has been deleted because it is not applicable to Monticello. The Emergency Diesel Generator-Emergency Service Water System provides the Monticello emergency diesel generator cooling water. The following Note has been renumbered to reflect this deletion.

6. Changes have been made to reflect changes made to other Specifications.

7. The ISTS 3.7.2 Required Action E.1 Completion Time has been changed from 72 hours to 7 days. The requirement for 72 hours was based on the more limiting Completion Time associated with restoring an inoperable DG. The Monticello EDGs are supported by the EDG-ESW system. The Completion Time has been revised to be consistent with the time provided to restore an inoperable RHR and core spray subsystem, which are the subsystems supported by the ESW System.

8. Each ESW subsystem includes both manual and automatic valves. There are no power operated valves other than automatic valves: therefore, reference to "power operated" valves in ISTS SR 3.7.2.5 (ITS SR 3.7.2.3) has been deleted.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

[PSW] System and [UHS]
B 3.7.2

B 3.7 PLANT SYSTEMS

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

BASES

BACKGROUND

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.

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

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. The ability of the [PSW] System 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]] (Refs. 1 and 2, respectively). These 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

[PSW] System and [UHS] (3)
B 3.7.2

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. (2) (2) (3)

The [PSW] System, together with the [UHS], satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). (3)

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. (3) (3) (3)

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. (3) (2)

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. (3) (3) (899)

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. (3) (3)

APPLICABILITY

In MODES 1, 2, and 3, the [PSW] System and [UHS] are required to be OPERABLE to support OPERABILITY of the equipment serviced by the [PSW] System. Therefore, the [PSW] System and [UHS] are required to be OPERABLE in these MODES. (3)

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

BASES

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. (1)

[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 the time required to reasonably complete the Required Action.]

[D.1

---REVIEWER'S NOTE---
The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.

[PSW] System and [UHS]
B 3.7.2

BASES

ACTIONS (continued)

With water temperature of the UHS > [90]°F, the design basis assumption associated with initial UHS temperature are bounded provided the temperature of the UHS averaged over the previous 24 hour period is ≤ [90]°F. With the water temperature of the UHS > [90]°F, long term cooling capability of the ECCS loads and DGs may be affected. Therefore, to ensure long term cooling capability is provided to the ECCS loads when water temperature of the UHS is > [90]°F, Required Action D.1 is provided to more frequently monitor the water temperature of the UHS and verify the temperature is ≤ [90]°F when averaged over the previous 24 hour period. The once per hour Completion Time takes into consideration UHS temperature variations and the increased monitoring frequency needed to ensure design basis assumptions and equipment limitations are not exceeded in this condition. If the water temperature of the UHS exceeds [90]°F when averaged over the previous 24 hour period or the water temperature of the UHS exceeds [ ]°F, Condition F must be entered immediately.]

A → E.1

E → 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.

7 days →

E →

7 day →

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.

RHR or CS subsystem →

A → a → Required Action E.1 is modified by two Notes indicating that the applicable Conditions of LCO 3.8.1, "AC Sources - Operating," 7 → 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.

E →

[PSW] System and [UHS]
B 3.7.2

BASES

ACTIONS (continued)

B
E.1 and E.2

E
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 or D~~], 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 3.7.2.1~~

~~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.]~~

1
[SR 3.7.2.2

E
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.]

2
[SR 3.7.2.3

E
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.]

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BASES

SURVEILLANCE REQUIREMENTS (continued)

[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 motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the cooling tower fans occurring between surveillances.]

SR 3.7.2.5

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.

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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.

REFERENCES

1. FSAR, Chapter [4]. Section 5.2.3
2. FSAR, Chapter [6].

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.2 BASES, EMERGENCY SERVICE WATER (ESW) AND ULTIMATE HEAT SINK (UHS)

1. Changes have been made to reflect those changes made to the Specification.

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

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

4. The Reviewer's Note is deleted as it is not part of the plant-specific ITS.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.2, EMERGENCY SERVICE WATER (ESW) AND ULTIMATE HEAT SINK (UHS)

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 3

ITS 3.7.3, Emergency Diesel Generator-Emergency Service Water (EDG-ESW) System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.7.3

Add proposed ITS 3.7.3

M.1

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DISCUSSION OF CHANGES
ITS 3.7.3, EMERGENCY DIESEL GENERATOR-EMERGENCY SERVICE WATER (EDG-ESW) SYSTEM

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M.1 The CTS does not have any specific requirements for the Emergency Diesel Generator-Emergency Service Water (EDG-ESW) System. The EDG-ESW System requirements are governed by the EDG Technical Specifications. ITS LCO 3.7.3 requires two EDG-ESW subsystems to be OPERABLE. An appropriate ACTION and Surveillance Requirements are also provided. This changes the CTS by incorporating the requirements of ITS 3.7.3.

The EDG-ESW System is necessary to support the emergency diesel generator requirement following a Design Basis Accident (DBA). The requirement to maintain two EDG-ESW subsystems OPERABLE assures adequate cooling capacity is available for the removal of heat from the emergency diesel generators required for safe shutdown following a DBA. The EDG-ESW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). This change is acceptable because the ability of the EDG-ESW System to provide adequate cooling to the emergency diesel generators is an implicit assumption for the safety analysis evaluated for a DBA. The ITS restoration actions for when an EDG-ESW subsystem is inoperable is consistent with the EDG CTS Actions, since the ITS requires the associated EDG to be declared inoperable. In addition, specific Surveillance requirements are now specified. This change is designated as more restrictive because it adds new requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

EDG-ESW → DG [1B] SSW System 3.7.3 (1)

CTS

3.7 PLANT SYSTEMS

3.7.3 Diesel Generator (DG) [1B] Standby Service Water (SSW) System (1)

DOC M.1 LCO 3.7.3 The DG [1B] SSW System shall be OPERABLE. (1)

APPLICABILITY: When DG [1B] is required to be OPERABLE. (3)

NOTE
Separate Condition entry is allowed for each EDG-ESW subsystem. (5)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M.1 A. DG [1B] SSW System inoperable. (One or more EDG-ESW subsystems)	A.1 Align cooling water to DG [1B] from a Unit [1] plant service water (PSW) subsystem.	8 hours
	AND	(2)
	A.2 Verify cooling water is aligned to DG [1B] from a Unit [1] PSW subsystem.	Once per 31 days
	AND	
	A.3 Restore DG [1B] SSW System to OPERABLE status.	60 days
B. Required Action and associated Completion Time not met.	B.1 (A) Declare DG [1B] inoperable. (associated EDG)	Immediately (2) (1)

EDG-ESW → ~~DG [1B] SSW~~ System 3.7.3 (1)

CTS

SURVEILLANCE REQUIREMENTS

		SURVEILLANCE	FREQUENCY	
DOC M.1	SR 3.7.3.1	Verify each ~~DG [1B] SSW System~~ (EDG-ESW subsystem) manual, power ~~operated, and automatic~~ valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	(1) (4)
DOC M.1	SR 3.7.3.2	Verify ~~the DG [1B] SSW System~~ (each EDG-ESW subsystem) pump starts automatically when ~~DG [1B]~~ (the associated EDG) starts and energizes the respective bus.	~~[18]~~ (24) months	(4) (1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.3, EMERGENCY DIESEL GENERATOR-EMERGENCY SERVICE WATER (EDG-ESW) SYSTEM

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

2. ISTS 3.7.3 Required Actions A.1, A.2 and A.3 have been deleted because they are not applicable to Monticello. The Monticello EDG-ESW System design does not provide for credited alternate cooling sources. In addition, ISTS 3.7.3 Condition B has been deleted to reflect the ISTS 3.7.3 ACTION A requirements for inoperability of one or more EDG-ESW subsystems. The following requirements have been renumbered, where applicable, to reflect this deletion.

3. ISTS 3.7.3 provides the cooling water requirement for the DG 1B SSW System. This Specification was a plant-specific Specification, written for the E. I. Hatch Units 1 and 2 design. Hatch Units 1 and 2 design includes a SSW System for the common DG (1B DG), separate from the cooling water for the remaining DGs. ISTS 3.7.2, Plant Service Water (PSW) System and Ultimate Heat Sink (UHS) provided the cooling water requirement for the remaining DGs. This is shown in the ISTS 3.7.2 Bases, Background section, which states that the PSW System provides cooling water for the removal of heat from equipment, such as the DGs. The ISTS 3.7.2 Applicability is MODES 1, 2, and 3. As stated in the ISTS 3.7.2 Bases, Applicability section, the MODES 4 and 5 OPERABILITY requirements of the PSW System and UHS are determined by the systems they support. Thus, for all DGs except the common 1B DG, no specific PSW requirements are provided in the ISTS for MODES other than MODES 1, 2, and 3. In addition, NUREG-1431, Rev.3, the Westinghouse ISTS, the ISTS includes only MODES 1, 2, 3, and 4 (which is equivalent to the BWR MODES 1, 2, and 3) for the DG cooling water Specification (ISTS 3.7.6, Service Water System). The Monticello design includes an EDG-ESW System that only provides cooling water to the EDGs. Therefore, consistent with the ISTS intent, the Applicability for ISTS 3.7.3 (ITS 3.7.3) has been changed to MODES 1, 2, and 3.

4. Each EDG-ESW subsystem includes only manual valves. There are no power operated or automatic valves; therefore, reference to "power operated" and "automatic" valves in ISTS SR 3.7.3.1 has been deleted.

5. An ACTIONS Note has been added to ISTS 3.7.3 (ITS 3.7.3) to allow separate Condition entry for each inoperable EDG-ESW subsystem. This change is intended to ensure that each occurrence of an inoperable EDG-ESW subsystem be assessed in accordance with the applicable Conditions and Required Actions of LCO 3.8.1 for its impact on the EDG capability to function as an AC power source. This change is acceptable since the ITS 3.7.3 Required Action is to declare the associated EDG inoperable.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

EDG-ESW → DG [1B] SSW System B 3.7.3 (1)

B 3.7 PLANT SYSTEMS

B 3.7.3 Diesel Generator (DG) [1B] Standby Service Water (SSW) System (1)

BASES

BACKGROUND

The DG [1B] SSW System is designed to provide cooling water for the removal of heat from the DG [1B]. DG [1B] is the only component served by the DG [1B] SSW System. (1)

The DG [1B] SSW pump autostarts upon receipt of a diesel generator (DG) start signal when power is available to the pump's electrical bus. Cooling water is pumped from the Altamaha River by the DG [1B] SSW pump to the essential DG components through the SSW supply header. After removing heat from the components, the water is discharged to the unit service water (PSW) discharge header. The capability exists to manually cross connect the PSW System to supply cooling to the DG [1B] during times when the SSW pump is inoperable. A complete description of the DG [1B] SSW System is presented in the FSAR, Section 9.5.5 (Ref. 1). (2) (3)

APPLICABLE SAFETY ANALYSES

The ability of the DG [1B] SSW System to provide adequate cooling to the DG [1B] is an implicit assumption for the safety analyses presented in the FSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). The ability to provide onsite emergency AC power is dependent on the ability of the DG [1B] SSW System to cool the DG [1B]. (1) (2) (3)

The DG [1B] SSW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). (1)

LCO

The OPERABILITY of the DG [1B] SSW System is required to provide a coolant source to ensure effective operation of the DG [1B] in the event of an accident or transient. The OPERABILITY of the DG [1B] SSW System is based on having an OPERABLE pump and an OPERABLE flow path. (2)

An adequate suction source is not addressed in this LCO since the minimum net positive suction head of the DG [1B] SSW pump is bounded by the PSW requirements (LCO 3.7.2, "Unit Service Water (PSW) System and Ultimate Heat Sink (UHS)"). (1) (4)

APPLICABILITY

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

INSERT 1

INSERT 1A

INSERT 2

2 **INSERT 1**

In addition, the Service Water System to EDG-ESW System isolation valves open upon receipt of an associated EDG start signal. If offsite power is not lost, then the Service Water System will supply cooling water to the associated EDG (i.e., the EDG-ESW and Service Water pumps are in parallel, and the discharge pressure of the Service Water pump is higher than the EDG-ESW pump). However, the Service Water System capability is not required to be OPERABLE as part of this Specification.

2 **INSERT 1A**

The EDG-ESW subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one subsystem does not depend on the other subsystem. In the event of a DBA, one EDG-ESW subsystem is required to provide the minimum heat removal capability assumed in the safety analysis for the associated EDG. To ensure this requirement is met, two EDG-ESW subsystems 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 pump and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the associated EDG.

1 **INSERT 2**

In MODES 1, 2, and 3, the EDG-ESW System is required to be OPERABLE to support OPERABILITY of the EDGs. Therefore, the EDG-ESW System is required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the EDG-ESW System are determined by the systems it supports. Therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. LCO 3.8.2, "AC Sources - Shutdown," will govern EDG-ESW System OPERABILITY requirements in MODES 4 and 5.

EDG-ESW → DG [1B] SSW System B 3.7.3 (1)

BASES

ACTIONS

INSERT 3

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

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. (1)

A → B.1 (1)

With one or more EDG-ESW subsystems 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 DG [1B] cannot perform its intended function and must be immediately declared inoperable. (1)

associated EDG (2)

In accordance with LCO 3.0.6, this also requires entering into the Applicable Conditions and Required Actions for LCO 3.8.1 or LCO 3.8.2 (5) (1) (2) Additionally, if the DG [1B] SSW System is not restored to OPERABLE status within 60 days, DG [1B] must be immediately declared inoperable.

of

, "AC Sources-Operating."

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

each EDG-ESW subsystem → Verifying the correct alignment for manual, power operated, and automatic valves in the DG [1B] SSW System flow path provides assurance that the proper flow paths will exist for DG [1B] SSW 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. A valve is also (1) (1)

1 **INSERT 3**

A Note has been provided to modify the ACTIONS related to EDG-ESW subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EDG-ESW subsystems provide appropriate compensatory measures for separate inoperable EDG-ESW subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable EDG-ESW subsystem.

EDG-ESW → DG [1B] SSW System B 3.7.3 (1)

BASES

SURVEILLANCE REQUIREMENTS (continued)

allowed to be in the nonaccident position, and yet be considered in the correct position provided it can be automatically realigned to its accident position, within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. (1)

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

SR 3.7.3.2 each EDG-ESW sub

This SR ensures that the DG [1B] SSW System pump will automatically start to provide required cooling to the DG [1B] when the DG [1B] starts and the respective bus is energized. (1)

associated EDG

24 Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency, which is based at the refueling cycle. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint. (3)

REFERENCES

1. FSAR, Section [9.5.5]. 10.4.4 (2) (3)
2. FSAR, Chapter [6]. Section 14.7.2.3.1.1 (2) (3)
3. FSAR, Chapter [15]. (3)

U

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.3 BASES, EMERGENCY DIESEL GENERATOR-EMERGENCY SERVICE WATER (EDG-ESW) SYSTEM

1. Changes have been made to reflect those changes made to the Specification.

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

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

4. Changes have been made to reflect changes made to other Specifications.

5. Typographical error corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.3, EMERGENCY DIESEL GENERATOR-EMERGENCY SERVICE WATER (EDG-ESW) SYSTEM

There are no specific NSHC discussions for this Specification.

ATTACHMENT 4

ITS 3.7.4, Control Room Emergency Filtration (CREF) System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

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ITS

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.a With both control room ventilation trains inoperable, restore at least one train to operable status within 24 hours. 3.b If 3.a is not met, then be in hot shutdown within the next 12 hours and in cold shutdown within 24 hours following the 12 hours. 3.c If 3.a is not met during movement of irradiated fuel assemblies in the secondary containment, core alterations, or activities having the potential for draining the reactor vessel then immediately suspend these activities.	See ITS 3.7.5
3.7.4 B. Control Room Emergency Filtration System (Add LCO Note)	B. Control Room Emergency Filtration System
LCO 3.7.4 1. Except as specified in 3.17.B.1.a through d below, two control room emergency filtration system filter trains shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel.	SR 3.7.4.1 1. At least once per month, initiate from the control room 1000 cfm (±10%) flow through both trains of the emergency filtration treatment system. The system shall operate for at least 10 hours with the heaters operable.
Applicability	

L.1 · LA.1 · L.2 · M.1 · A.2 · operating · recently · A.6

3.17/4.17

229v 8/18/00
Amendment No. 65, 89, 112

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION		4.0 SURVEILLANCE REQUIREMENTS
ACTION A ACTION C	a. When one control room emergency filtration system filter train is made or found to be inoperable for any reason, restore the inoperable train to operable status within seven days/or be in hot shutdown within the next 12 hours following the seven days and either reduce the reactor coolant temperature to below 212°F/or initiate and maintain the operable emergency filtration system filter train in the pressurization mode/within the following 24 hours.	M.2
ACTION B ACTION C	b. When/both filter trains of the control room emergency filtration system/are inoperable, restore at least one train to operable status within 24 hours or/be in hot shutdown within the next 12 hours following the 24 hours and reduce the reactor coolant water temperature to below 212°F within the following 24 hours.	M.3 Add proposed ACTION E
ACTION A ACTION D	c. With one control room emergency filtration system filter train inoperable during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel, restore the inoperable train to operable status within 7 days/or immediately after the 7 days initiate and maintain the operable emergency filtration system filter train in the pressurization mode or immediately suspend these activities.	recently A.6 LCO 3.0.3 is not applicable. A.3 A.6
ACTION F	d. With both control room emergency filtration system filter trains inoperable during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel, immediately suspend these activities.	recently LCO 3.0.3 is not applicable. A.3
3.17/4.17		229v v 8/18/00 Amendment No. 112

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A.1

Add proposed SR 3.7.4.2

A.4

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Performance Requirements a. Acceptance Criteria - Periodic Requirements (1) The results of the in-place DOP tests at 1000 cfm (±10%) shall show ≤1% DOP penetration on each individual HEPA filter and shall show ≤0.05% DOP penetration on the combined HEPA filters. (2) The results of in-place halogenated hydrocarbon tests at 1000 cfm (±10%) shall show ≤1% penetration on each individual charcoal adsorber and shall show ≤0.05% penetration on the combined charcoal banks. (3) The results of laboratory carbon sample analysis shall show ≤0.5% methyl iodide penetration when tested at 30°C and 95% relative humidity.	2. Performance Requirement Test The in-place performance testing of HEPA filter banks and charcoal adsorber banks shall be conducted in accordance with Sections 10 and 11 of ASME N510-1989. The carbon sample test for methyl iodide shall be conducted in accordance with ASTM D 3803-1989. Sample removal shall be in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978. a. At least once per operating cycle, but not to exceed 18 months; or following painting, fire, or chemical release while the system is operating that could contaminate the HEPA filters or charcoal adsorbers, perform the following: (1) In-place DOP test the HEPA filter banks. (2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer. (3) Remove one carbon test sample from each charcoal adsorber bank. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency. (4) Initiate from the control room 1000 cfm (±10%) flow through both trains of the emergency filtration treatment system.

See ITS 5.5

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A.1

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
b. Acceptance Criteria - System Operation Requirements The results of laboratory carbon sample analysis shall show ≤ 0.5% methyl iodide penetration when tested at 30°C and 95% relative humidity.	b. At least once per 720 hours of system operation, remove one carbon test sample from each charcoal adsorber bank. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency. { See ITS 5.5 }

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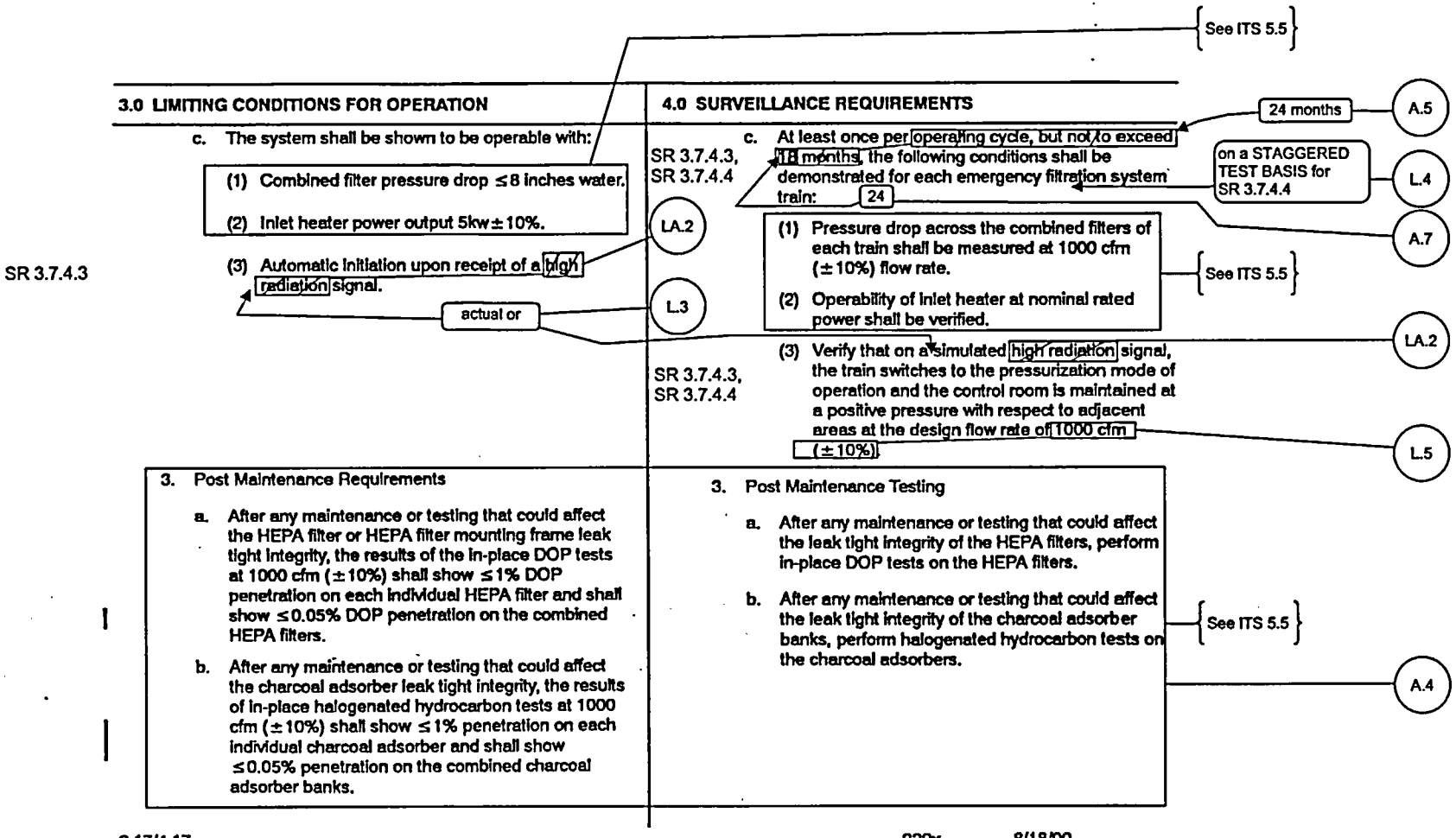
A.1

ITS

ITS



3.17/4.17

229x 8/18/00
Amendment No. ~~65, 101, 108,~~ 112

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DISCUSSION OF CHANGES
ITS 3.7.4, CONTROL ROOM EMERGENCY FILTRATION (CREF) SYSTEM

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 4.17.B.1 states to operate each CREF subsystem for at least 10 hours with the heaters "operable." ITS SR 3.7.4.1 requires the CREF System to operate with the heaters "operating." This changes the CTS by requiring the CREF heaters to be "operating" in lieu of being "operable" during the test.

The purpose of CTS 4.17.B.1 is to ensure each CREF subsystem remains OPERABLE to support the safety analyses. This change requires the CREF heaters to be "operating" in lieu of being "operable" during the test. Moisture may accumulate on the high efficiency particulate air filters and the charcoal adsorber when the system is in standby. Performing this Surveillance on a periodic frequency will remove the moisture since it is necessary for the heaters to actually operate to reduce moisture from the adsorbers and HEPA filters. Therefore it is necessary that the heaters are actually operating during the test. This change is consistent with current practice. The heaters are still required to be OPERABLE, as required by ITS SR 3.7.4.2 and ITS 5.5.6, Ventilation Filter Testing Program (VFTP). This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 Under certain conditions, CTS 3.17.B.1.c and 3.17.B.1.d, in part, require the immediate suspension of movement of recently irradiated fuel assemblies in the secondary containment. ITS 3.7.4 ACTIONS D and F include the same requirement, however a Note has been added that states that LCO 3.0.3 is not applicable. This changes the CTS by adding this Note.

The purpose of CTS 3.17.B.1.c and 3.17.B.1.d, in part, is to provide the appropriate actions when one or two CREF subsystems, as applicable, are inoperable while moving recently irradiated fuel in the secondary containment. This change adds a Note that states LCO 3.0.3 is not applicable. This Note has been added because ITS LCO 3.0.3 has been added to ITS Section 3.0 in accordance with ITS Section 3.0 DOC M.1. This Note is necessary because if moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. Since ITS LCO 3.0.3 is not currently included in the CTS this change is considered administrative. This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 3.7.4, CONTROL ROOM EMERGENCY FILTRATION (CREF) SYSTEM

A.4 CTS 3/4.17.B.2 specifies the performance requirements for the CREF subsystems while CTS 3/4.17.B.3 specifies the post maintenance requirements for the CREF subsystems. ITS SR 3.7.4.2 requires the performance of the required CREF filter testing in accordance with the Ventilation Filter Testing Program (VFTP). CTS 3/4.17.B does not include a VFTP, but the requirements that make up the program are being moved to ITS 5.5. This changes the CTS by requiring testing in accordance with the VFTP, whose requirements are being moved to ITS 5.5.

This change is acceptable because filter testing requirements are being moved to the VFTP as part of ITS 5.5 and ITS SR 3.7.4.2 references the VFTP for performing these tests. This change is designated as administrative because it does not result in technical changes to the CTS.

A.5 CTS 4.17.B.2.c requires verification of the OPERABILITY of each CREF subsystem each "operating cycle." ITS SR 3.7.4.3 and ITS SR 3.7.4.4 require the same testing however the Surveillances are required to be performed every "24 months." This changes the CTS by changing the Frequency from "operating cycle" to "24 months."

This change is acceptable because the current "operating cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 to 24 months and the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. 4.17.B.2.c was included in this evaluation. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.6 These changes to CTS 3.17.B.1, CTS 3.17.B.1.c, and CTS 3.17.B.1.d are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-04-023, from Thomas J. Palmisano (NMC) to USNRC, dated April 29, 2004. As such, these changes are administrative.

A.7 This change to CTS 4.17.B.2.c is provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-04-036, from Thomas J. Palmisano (NMC) to USNRC, dated June 30, 2004. As such, this change is administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.17.B.1, is applicable, in part, whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. ITS LCO 3.7.4 is applicable in MODES 1, 2, and 3. This changes the CTS by requiring the CREF System to be OPERABLE in MODE 2 when reactor water temperature is less than or equal to 212°F.

DISCUSSION OF CHANGES
ITS 3.7.4, CONTROL ROOM EMERGENCY FILTRATION (CREF) SYSTEM

The purpose of CTS 3.17.B.1 is to ensure the CREF System is OPERABLE to mitigate the consequences of a design basis accident. The CREF System is required to be OPERABLE during MODES 1, 2, and 3 when there is considerable energy in the reactor core or there is a potential for the reactor core to become critical and the CREF System would be required to mitigate the consequences of a design basis accident. In MODES 1 and 3, the reactor coolant temperature will always be above 212°F. In MODE 2, the reactor coolant temperature may be less than or equal to 212°F when the reactor is subcritical but control rods are withdrawn. Therefore, it is necessary and acceptable to require the CREF System to be OPERABLE. This change is designated as more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

M.2 CTS 3.17.B.1.a allows 7 days to restore an inoperable CREF subsystem. If this cannot be met, it requires the unit to be in hot shutdown (i.e., MODE 3) in 12 hours and to be below 212°F (i.e., be in MODE 4) or initiate and maintain the OPERABLE CREF subsystem in the pressurization mode within the following 24 hours. ITS 3.7.4 ACTION C does not include the option to place the OPERABLE CREF subsystem in operation in lieu of being in MODE 4. This changes the CTS by deleting the allowance to place the OPERABLE CREF subsystem in operation in lieu of achieving MODE 4 conditions.

The purpose of the 3.17.B.1.a, in part, is to place the unit in a MODE in which the LCO is no longer applicable in a reasonable amount of time. ITS 3.7.4 ACTION C does not include the option to place the OPERABLE CREF subsystem in operation. ITS 3.7.4 ACTION C requires the unit to be in MODE 3 in 12 hours and MODE 4 in 36 hours. This change deletes the allowance to place the OPERABLE CREF subsystem in operation and continue to operate in MODE 3. CTS 3.17.B.1 allows the unit to operate in MODE 3 conditions indefinitely with an inoperable CREF subsystem as long as the OPERABLE CREF subsystem is operating. With the system operating, automatic functions of the systems may not be required since the system is operating, however, there is a potential that an operating component can still fail if required (for example, power is not available). In MODE 3 conditions, the potential for a design basis accident still exists, therefore the allowance has been deleted since the failure of the operating subsystem may result in a failure to satisfy the requirements of the system. This change is designated as more restrictive since the option to start the OPERABLE CREF subsystem has been deleted.

M.3 When both CREF subsystems are inoperable, CTS 3.17.B.1.b allows 24 hours to restore an inoperable CREF subsystem to OPERABLE status prior to initiating a reactor shutdown. When both CREF subsystems are inoperable due to an inoperable control room boundary, ITS 3.7.4 ACTION B allows 24 hours to restore the control room boundary to OPERABLE status. When both CREF subsystems are inoperable for reasons other than an inoperable control room boundary, ITS 3.7.4 ACTION E requires immediate entry into LCO 3.0.3. This will require the unit to initiate action within 1 hour to place the unit in MODE 2 within 7 hours, MODE 3 within 13 hours, and MODE 4 within 37 hours. This changes the CTS by requiring an immediate entry into LCO 3.0.3 when two

DISCUSSION OF CHANGES
ITS 3.7.4, CONTROL ROOM EMERGENCY FILTRATION (CREF) SYSTEM

CREF subsystems are inoperable for any reason other than the control room boundary being inoperable.

The purpose of the CTS 3.17.B.1.b is to provide appropriate compensatory action when both CREF subsystems are inoperable. This change requires immediate entry into LCO 3.0.3 when two CREF subsystems are inoperable for any reason other than the control room boundary being inoperable. Under the same conditions in the CTS, 24 hours is allowed to restore one CREF subsystem prior to initiating a reactor shutdown. This change is acceptable because it does not allow the unit to operate for 24 hours with two inoperable CREF subsystems in MODES 1, 2, and 3. This is necessary because if there is an accident, the CREF subsystem will not be able to pressurize the control room boundary and ensure the control room dose remains within the safety analyses dose calculations. This change is designated as more restrictive because with two CREF subsystems inoperable in MODES 1, 2, and 3 the reactor will have to be shutdown sooner in the ITS than is currently allowed in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.17.B.1 states to "initiate from the control room" flow through CREF subsystem and operate for at least 10 hours. ITS SR 3.7.4.1 includes the same requirement, however, the statement to "initiate from the control room" is not included. This changes the CTS by moving the requirement to "initiate from the control room" from the CTS to the ITS Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to operate the CREF subsystem for $\geq$ 10 hours in ITS SR 3.7.4.1. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.17.B.2.c.(3) and CTS 4.17.B.2.c.(3) require each CREF subsystem be shown to be OPERABLE with automatic initiation upon receipt of a "high radiation" signal. ITS SR 3.7.4.3 requires verification that each CREF subsystem actuates on an initiation signal. This changes the CTS by moving the specific type of actuation signal to the ITS Bases.

DISCUSSION OF CHANGES
ITS 3.7.4, CONTROL ROOM EMERGENCY FILTRATION (CREF) SYSTEM

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to verify each CREF subsystem actuates on an initiation signal. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirement)* CTS 3.17.B.1 requires two CREF subsystems to be OPERABLE, but does not allow the main control room boundary to be opened intermittently under administrative controls. If it is opened, both CREF subsystems are inoperable. ITS LCO 3.7.4 also requires the two CREF subsystems to be OPERABLE, however a Note to the LCO is included that allows the main control room boundary to be opened intermittently under administrative controls. This changes the CTS by allowing the main control room boundary to be opened intermittently under administrative controls and not consider the CREF System to be inoperable.

The purpose of CTS 3.17.B.1 is to ensure the CREF System remains OPERABLE to support the safety analyses. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses. The LCO is modified by a Note allowing the main control room boundary to be opened intermittently under administrative controls. The control room boundary is often opened intermittently to allow entry and exit and also periodically opened to perform maintenance. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the main control room. This individual will have a method to rapidly close the opening when a need for main control room isolation is indicated. This change is acceptable because the ITS Bases requires these administrative controls to be met in order to utilize the proposed LCO Note. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L.2 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.17.B.1 requires that each CREF subsystem be initiated with 1000 cfm (± 10%) of flow and operated for 10 hours. ITS SR 3.7.4.1 includes a similar requirement, except the flow rate is not specified. This changes the CTS by deleting the flow rate requirement from the Surveillance acceptance criteria.

The purpose of 4.17.B.1 is to ensure the each CREF subsystem remains OPERABLE to support the safety analyses. This change is acceptable because it

has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. This change deletes the flow rate requirement from the Surveillance acceptance criteria. Moisture may accumulate on the high efficiency particulate air filters and the charcoal adsorber when the system is in standby. Performing this Surveillance on a periodic frequency will remove the moisture since it is necessary for the heaters to actually operate (cycle properly when required) to reduce moisture from the adsorbers and HEPA filters. The actual flow rate is not specifically required to remove the moisture. Therefore, the explicit flow rate requirement has been deleted. Each CREF subsystem is still required to be tested at the specified flow rate as required by ITS SR 3.7.4.2 and ITS 5.5.6. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L.3 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 3.17.B.2.c.(3) and 4.17.B.2.c.(3) require verification of the automatic actuation of each CREF subsystem upon a receipt of the specified inputs (i.e., simulated signal). ITS SR 3.7.4.3 specifies that the signal may be from either an "actual" or "simulated" initiation signal. This changes the CTS by explicitly allowing the use of an actual signal for the test.

The purpose of CTS 3.17.B.2.c.(3) and 4.17.B.2.c.(3) is to ensure that each CREF subsystem operates correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L.4 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.17.B.2.c.(3) requires each CREF subsystem to be operating in the pressurization mode of operation while maintaining the control room at a positive pressure with respect to adjacent areas, at least once per operating cycle. ITS SR 3.7.4.4 requires this same test, however it is required to be performed every 24 months "on a STAGGERED TEST BASIS." This changes the CTS by requiring the test to be performed using each CREF subsystem at least once per 48 months.

The purpose of the CTS 4.17.B.2.c.(3), in part, is to ensure the integrity of the control room boundary. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The change is acceptable since the proposed Surveillance Frequency will continue to require performance of the test every 24 months. This will ensure the control room boundary integrity is maintained. The status of the integrity of the control room boundary can be determined with either CREF

subsystem. The secondary purpose of this test will now also serve to ensure that each CREF subsystem is being tested every 48 months (based on the definition of STAGGERED TEST BASIS). ITS SR 3.7.4.3 requires the performance of a test to ensure each CREF subsystem actuates on an actual or simulated initiation signal. Therefore, each subsystem will continue to be tested to ensure it can be automatically aligned to the pressurization mode, however the pressurization verification will only be required with one subsystem in operation. This change is designated as less restrictive because the pressurization Surveillance will only be required to be performed on one CREF subsystem each Surveillance interval instead of on both CREF subsystems.

L.5 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.17.B.2.c.(3) requires each CREF subsystem to be operating in the pressurization mode of operation while maintaining the control room at a positive pressure with respect to adjacent areas with a flow rate of 1000 cfm (± 10%). ITS SR 3.7.4.4 requires this same test, however the flow rate limit is specified to be ≤ 1100 cfm. This changes the CTS by deleting the minimum flow rate limit.

The purpose of the CTS 4.17.B.2.c.(3), in part, is to ensure the integrity of the control room boundary. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. The CREF flow rate limit range has been changed to a maximum flow rate limit. The CTS requires the CREF flow rate to be from 900 cfm to 1100 cfm during the control room boundary integrity test. In the ITS, the same test may be performed at ≤ 1100 cfm. This change is acceptable because if the control room boundary is positive with a lower flow rate, the control room boundary is intact. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CREF
[MCREC] System 3.7.4 (1)

CTS

3.7 PLANT SYSTEMS

Emergency Filtration (CREF)

3.17.B 3.7.4 [Main] Control Room [Environmental Control (MCREC)] System (1)

CREF

3.17.B.1 LCO 3.7.4 Two [MCREC] subsystems shall be OPERABLE. (1)

---NOTE---

DOC L.1 The main control room boundary may be opened intermittently under administrative control.

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.17.B.1.a, 3.17.B.1.c	A. One [MCREC] subsystem inoperable. (CREF)	A.1 Restore [MCREC] subsystem to OPERABLE status. (CREF)	7 days	(1)
3.17.B.1.b	B. Two [MCREC] subsystems inoperable due to inoperable control room boundary in MODE 1, 2, or 3. (CREF)	B.1 Restore control room boundary to OPERABLE status.	24 hours	(1)
3.17.B.1.a, 3.17.B.1.b	C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours	

CREF [MCREC] System 3.7.4 (1)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.17.B.1.c D. Required Action and associated Completion Time of Condition A not met during movement of (1) [recently] irradiated fuel assemblies in the [secondary] containment or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	D.1 ----------NOTE---------- [Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.] ----------	(2)
	Place OPERABLE CREF [MCREC] subsystem in [pressurization] mode.	Immediately (1)
	OR	
	D.2.1 Suspend movement of [recently] irradiated fuel assemblies in the [secondary] containment.	Immediately (1)
	AND	
	D.2.2 Initiate action to suspend OPDRVs.	Immediately
3.17.B.1.b E. Two [MCREC] CREF subsystems inoperable in MODE 1, 2, or 3 for reasons other than Condition B.	E.1 Enter LCO 3.0.3.	Immediately (1)

CREF
[MCREC] System 3.7.4 (1)

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.17.B.1.d	F. Two [MCREC] subsystems inoperable during movement of [recently] irradiated fuel assemblies in the [secondary] containment or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------		
		F.1 Suspend movement of [recently] irradiated fuel assemblies in the [secondary] containment.	Immediately	(1)
		AND		
		F.2 Initiate action to suspend OPDRVs.	Immediately	

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.17.B.1	SR 3.7.4.1	Operate each [MCREC] subsystem for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days	(1)
4.17.B.2, 4.17.B.3	SR 3.7.4.2	Perform required [MCREC] filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]	(1)
3.17.B.2.c.(3), 4.17.B.2.c, 4.17.B.2.c.(3)	SR 3.7.4.3	Verify each [MCREC] subsystem actuates on an actual or simulated initiation signal.	[18] months	(1)
4.17.B.2.c, 4.17.B.2.c.(3)	SR 3.7.4.4	[Verify each [MCREC] subsystem can maintain a positive pressure of ≥ [0.1] inches water gauge relative to the [turbine building] during the [pressurization] mode of operation at a flow rate of ≤ [400] cfm.	[18] months on a STAGGERED TEST BASIS [	(1) (3)

CREF
24
adjacent areas
1100

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.4, CONTROL ROOM EMERGENCY FILTRATION (CREF) SYSTEM

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

2. Required Action D.1 is modified by a Note that states to place the CREF System in the toxic gas protection mode if automatic transfer to toxic gas protection modes is inoperable. The Monticello CREF design does not include an automatic transfer to the toxic gas protection mode. Therefore, the Note is deleted.

3. ISTS SR 3.7.4.4 specifies a bracketed positive pressure criterion of 0.1 inches water gauge relative to the turbine building. ITS SR 3.7.4.4 only requires maintaining a positive pressure relative to adjacent areas. This difference was accepted by the NRC in a letter dated May 30, 1989 from John F. Stepano (NRC) to Mr. Musolf (NSPC) and discussed in Section 2, page 7, of the associated Safety Evaluation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

CREF
[MCREC] System 1
B 3.7.4

B 3.7 PLANT SYSTEMS

Emergency Filtration (CREF)
B 3.7.4 [Main] Control Room Environmental Control (MCREC)] System 1

BASES

CREF

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

CREF
The safety related function of [MCREC] System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of recirculated air or outside supply air. Each subsystem consists of a demister, an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a booster fan, an air handling unit (excluding the condensing unit), and the associated ductwork and dampers. Demisters remove water droplets from the airstream. 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. 1 2

low efficiency filter
s
two
an emergency filter
an exhaust/recirculation fan,
Low efficiency filters

CREF
boundary 2
The [MCREC] System is a standby system, parts of which also operate during normal unit operations to maintain the control room environment. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the [MCREC] System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and a part of the recirculated air is routed through either of the two filter subsystems. 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 for removal of airborne radioactive particles. 1 2

a Control Room Air Inlet Radiation - High
boundary (the main control room and portions of the first and second floors of the Emergency Filtration Train (EFT) building). 2
CREF
INSERT 1
INSERT 2

CREF
boundary 2 1
The [MCREC] System is designed to maintain the control room 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 will pressurize the control room to about [0.1] inches water gauge to prevent infiltration of air from surrounding buildings. [MCREC] System operation in maintaining control room habitability is discussed in the FSAR, Chapters [6] and [9], (Refs. 1 and [2], respectively). 1 2 1 2 2, 3, 4

CREF
CREF
U
Sections 14.7.2.4.3, 14.7.6.3.2, 14.7.3.2.4, and 14.7.1.6

2 **INSERT 1**

boundary from untreated outside air.

2 **INSERT 2**

This air is then combined with return air from the control room boundary and passed through an exhaust/recirculation fan, which is then passed through the air handling unit into the control room boundary.

BASES

APPLICABLE SAFETY ANALYSES

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 2, respectively). The pressurization mode of the [MCREC] System is assumed to operate following a loss of coolant accident, fuel handling accident [involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X days]], main steam line break, and control rod drop accident, as discussed in the FSAR, Section [6.4.1.2.2] (Ref. 4). [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.

The [MCREC] System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

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.

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

a. Fan is OPERABLE[, exhaust/recirculation fan, and air handling unit (excluding the condenser unit) are];

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

c. Heater, [demister,] ductwork, [valves,] and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

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

[MCREC] System
B 3.7.4

BASES

APPLICABILITY

In MODES 1, 2, and 3, the [MCREC] 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 [MCREC] 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 operations with potential for draining the reactor vessel (OPDRVs) and;

b. During movement of [recently] irradiated fuel assemblies in the [secondary] containment. [Due to radioactive decay, the MCREC System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days).]

ACTIONS

A.1

With one [MCREC] subsystem inoperable, the inoperable [MCREC] subsystem 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

---REVIEWER'S NOTE---

Adoption of Condition B is dependent on a commitment from the licensee to have written procedures available describing compensatory measures to be taken in the event of an intentional or unintentional entry into Condition B.

CREF
[MCREC] System
B 3.7.4 (1)

BASES

ACTIONS (continued)

CREF

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

10 CFR 50, Appendix A,

C.1 and C.2

CREF

In MODE 1, 2, or 3, if the inoperable [MCREC] subsystem or control room boundary 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. (1)

D.1, D.2.1 and D.2.2 (5)

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

CREF
[MCREC] System (1)
B 3.7.4

BASES

ACTIONS (continued)

During movement of [recently] irradiated fuel assemblies in the [secondary] containment 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)

Required Action D.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]. (4)

An alternative to Required Action D.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. (2)

If applicable, movement of [recently] 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 the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended. (1)

E.1

If both [MCREC] subsystems are inoperable in MODE 1, 2, or 3 for reasons other than an inoperable control room boundary (i.e., Condition B), the [MCREC] System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately. (1)

F.1 and F.2

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

BASES

ACTIONS (continued)

During movement of [recently] irradiated fuel assemblies in the [secondary] containment 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. (1) (2)

CREF

boundary

If applicable, movement of [recently] 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. (1)

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

from the control room

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any 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.] Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available. (2) (1)

SR 3.7.4.2

CREF

This SR verifies that the required [MCREC] testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP]. (1)

CREF
[MCREC] System (1)
B 3.7.4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.3

CREF
This SR verifies that on an actual or simulated initiation signal, each [MCREC] subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.5 overlaps this SR to provide complete testing of the safety function. The [18] month Frequency is specified in Reference 5. (1) (5) (7)

LCO 3.3.7.1, "Control Room Emergency Filtration (CREF) Instrumentation,"

INSERT 3

SR 3.7.4.4 (1)

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 prevent unfiltered inleakage. The [MCREC] System is designed to maintain this positive pressure at a flow rate of ≤ [400] cfm to the control room in the pressurization mode. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration systems SRs. (2) (4) (1) (24)

boundary (2)
CREF (4) (1)
boundary (2)
CREF (1)
1100
adjacent areas (4)
boundary (2)
24
(1)

REFERENCES

U 1. FSAR, Chapter [6]. Section 14.7.2.4.3 (1) (2)

U 2. FSAR, Chapter [9]. Section 14.7.6.3.2 (1) (2)

U 3. FSAR, Chapter [15]. Section 14.7.3.2.4 (1) (2)

U 4. FSAR, Section [6.4.1.2.2]. 14.7.1.6 (1) (2)

5. Regulatory Guide 1.52, Rev. [2]. (7)

7 **INSERT 3**

Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.4 BASES, CONTROL ROOM EMERGENCY FILTRATION (CREF) 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 Reviewer's Note is deleted as it is not part of the plant-specific ITS.

4. Changes made to be consistent with changes made to the Specification.

5. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

6. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

7. ISTS SR 3.7.4.3 verifies that each [MCREC] subsystem (changed to CREF subsystem in ITS SR 3.7.4.3) actuates on an actual or simulated initiation signal every 18 months. The justification for the 18 month Frequency is that it is specified in Regulatory Guide 1.52. Regulatory Guide 1.52 addresses filtration requirements. The Surveillance verifies mechanical requirements and the Bases have been modified to correctly state the basis of the Frequency.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.4, CONTROL ROOM EMERGENCY FILTRATION (CREF) SYSTEM

There are no specific NSHC discussions for this Specification.

ATTACHMENT 5

ITS 3.7.5, Control Room Ventilation System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.17 CONTROL ROOM HABITABILITY	4.17 CONTROL ROOM HABITABILITY
Applicability: Applies to the control room ventilation system equipment necessary to maintain habitability.	Applicability: Applies to the periodic testing requirements of systems required to maintain control room habitability.
Objectives: To assure the control room is habitable both under normal and accident conditions.	Objectives: To verify the operability of equipment related to control room habitability.
Specification:	Specification:
3.7.5 A. Control Room Ventilation System	A. Control Room Ventilation System
LCO 3.7.5 — 1. Except as specified in 3.17.A.2 and 3.17.A.3 below, both trains of the control room ventilation system shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel. — Applicability	1. Once per 12 hours check control room temperature.
ACTION A — 2.a With one control room ventilation train inoperable, restore the inoperable train to operable status within 30 days.	
ACTION B — 2.b If 2.a is not met, then be in hot shutdown within the next 12 hours following the 30 days and in cold shutdown within 24 hours following the 12 hours.	LCO 3.0.3 is not applicable.
ACTION C — 2.c If 2.a is not met during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel then immediately place the operable control room ventilation train in operation or immediately suspend these activities.	

M.1 — A.3 — L.1 — Add proposed SR 3.7.5.1 — M.2 — A.2 — A.3

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION		4.0 SURVEILLANCE REQUIREMENTS
ACTION D	3.a With both control room ventilation trains inoperable, restore at least one train to operable status within 24 hours.	M.3
ACTION D	3.b If 3.a is not met, then be in hot shutdown within the next 12 hours and in cold shutdown within 24 hours following the 12 hours.	Enter LCO 3.0.3 immediately; M.3
		With both control room ventilation trains inoperable, A.3
ACTION E	3.c If 3.a is not met during movement of irradiated fuel assemblies in the secondary containment, core alterations, or activities having the potential for draining the reactor vessel then immediately suspend these activities.	LCO 3.0.3 is not applicable. A.2; A.3
	B. Control Room Emergency Filtration System 1. Except as specified in 3.17.B.1.a through d below, two control room emergency filtration system filter trains shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel.	B. Control Room Emergency Filtration System 1. At least once per month, initiate from the control room 1000 cfm (±10%) flow through both trains of the emergency filtration treatment system. The system shall operate for at least 10 hours with the heaters operable. — See ITS 3.7.4

3.17/4.17

229v 8/18/00
Amendment No. 65, 89, 112

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DISCUSSION OF CHANGES
ITS 3.7.5, CONTROL ROOM VENTILATION SYSTEM

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 Under certain conditions, CTS 3.17.A.2.c and CTS 3.17.A.3.c, in part, require immediate suspension of movement of irradiated fuel assemblies in the secondary containment. ITS 3.7.5 ACTIONS C and E include the same requirement, however a Note has been added that states that LCO 3.0.3 is not applicable. This changes the CTS by adding this Note.

The purpose of CTS 3.17.A.2.a and CTS 3.17.A.3.c, in part, is to provide the appropriate actions when one or two control room ventilation subsystems, as applicable, are inoperable while moving irradiated fuel in the secondary containment. This change adds a Note that states LCO 3.0.3 is not applicable. This Note has been added because ITS LCO 3.0.3 has been added to ITS Section 3.0 in accordance with ITS Section 3.0 DOC M.1. This Note is necessary because if moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. Since ITS LCO 3.0.3 is not currently included in the CTS this change is considered administrative. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 These changes to CTS 3.17.A.1, CTS 3.17.A.2.c, and CTS 3.17.A.3.c are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval in NMC letter L-MT-04-023, from Thomas J. Palmisano (NMC) to USNRC, dated April 29, 2004. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.17.A.1 is applicable, in part, whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. ITS LCO 3.7.5 is applicable in MODES 1, 2, and 3. This changes the CTS by requiring the Control Room Ventilation System to be OPERABLE in MODE 2 when reactor water temperature is less than or equal to 212°F.

The purpose of CTS 3.17.A.1 is to ensure the Control Room Ventilation System is OPERABLE to mitigate the consequences of a design basis accident. The Control Room Ventilation System is required to be OPERABLE during MODES 1,

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DISCUSSION OF CHANGES
ITS 3.7.5, CONTROL ROOM VENTILATION SYSTEM

2, and 3 when there is considerable energy in the reactor core or there is the potential for the reactor core to become critical and the Control Room Ventilation System would be required to mitigate the consequences of a design basis event. In MODES 1 and 3, the reactor coolant temperature will always be above 212°F. In MODE 2, the reactor coolant temperature may be less than or equal to 212°F when the reactor is subcritical but control rods are withdrawn. Therefore, it is necessary and acceptable to require the Control Room Ventilation System to be OPERABLE. This change is designated as more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

M.2 CTS 4.17.A does not provide a requirement to verify the capability of the Control Room Ventilation System to remove the assumed heat load. ITS 3.7.5 includes a Surveillance Requirement to cover this requirement. ITS SR 3.7.5.1 requires verification that each control room ventilation subsystem has the capability to remove the assumed heat load every 24 months. This changes the CTS by adding an additional OPERABILITY requirement for the Control Room Ventilation System.

The purpose of the new Surveillance Requirement is to ensure that the heat removal capability of the system is sufficient to remove the control room boundary 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 Ventilation System is not expected over this time period. This change is acceptable because it provides additional assurance that the Control Room Ventilation System will perform correctly during a design basis event. This change is designated as more restrictive because it adds a new Surveillance Requirement to the CTS.

M.3 When both control room ventilation subsystems are inoperable, CTS 3.17.A.3.a allows 24 hours to restore an inoperable control room ventilation subsystem to OPERABLE status. If CTS 3.17.A.3.a is not met, CTS 3.17.A.3.b requires the unit to be in MODE 3 in 12 hours and in MODE 4 within the following 24 hours. ITS 3.7.5 ACTION D requires immediate entry into LCO 3.0.3 under the same conditions. This will require the unit to initiate action within 1 hour to place the unit in MODE 2 within 7 hours, MODE 3 within 13 hours, and MODE 4 within 37 hours. This changes the CTS by requiring an immediate entry into LCO 3.0.3 when two control room ventilation subsystems are inoperable.

The purpose of the CTS 3.17.A.3.a and CTS 3.17.A.3.b is to provide appropriate compensatory actions when both control room ventilation subsystems are inoperable and place the unit in a MODE in which the LCO is no longer applicable in a reasonable amount of time if at least one subsystem is not restored to OPERABLE status. This change requires immediate entry into LCO 3.0.3 when two control room ventilation subsystems are inoperable. ITS 3.7.5 ACTION D does not include any specific time for restoration and requires immediate entry into LCO 3.0.3. This change is acceptable because the equipment in the control room boundary may not function correctly if a design basis event were to occur. This change places the unit outside of the applicability of the Specification where the system is not required to mitigate the consequences of those design basis events postulated during operating

DISCUSSION OF CHANGES
ITS 3.7.5, CONTROL ROOM VENTILATION SYSTEM

conditions. This change is designated as more restrictive since the restoration time of 24 hours has been deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.17.A.1 requires verification that the control room temperature is within limit. ITS 3.7.5 does not include this requirement. This changes the CTS by eliminating the Surveillance Requirement to verify control room temperature.

The purpose of CTS 4.17.A.1 is to ensure the continuous duty rating for the instrumentation and equipment cooled by this system is not exceeded. This change is acceptable because the deleted Surveillance Requirement is not necessary to ensure the Control Room Ventilation System can perform its safety function. ITS SR 3.7.5.1 has been added in accordance with DOC M.2 to verify each control room ventilation subsystem has the capability to remove the assumed heat load. This SR will ensure the Control Room Ventilation System can perform its safety function. Temperature is not always the appropriate method to verify the system capability to remove its design basis heat load because the conditions in the control room boundary do not always reflect the assumptions of the accident (e.g., personnel assumed to be in the control room boundary during an accident, the system does not normally operate in the pressurization mode of operation). ITS SR 3.7.5.1 will ensure each control room ventilation subsystem has sufficient cooling capability to meet the safety analyses assumptions. This change is designated as less restrictive because a Surveillance that is required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

Ventilation

[Control Room AC] System
3.7.5 (1)

CTS

3.7 PLANT SYSTEMS

Ventilation

3.17.A 3.7.5 [Control Room Air Conditioning (AC)] System (1)

ventilation

3.17.A.1 LCO 3.7.5 Two [control room AC] subsystems shall be OPERABLE. (1)

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.17.A.2.a	A. One [control room AC] subsystem inoperable.	A.1 Restore [control room AC] subsystem to OPERABLE status.	30 days	(1)
3.17.A.2.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	
3.17.A.2.c	C. Required Action and associated Completion Time of Condition A not met during movement of [recently] irradiated fuel assemblies in the [secondary] containment or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ---------- C.1 Place OPERABLE [control room AC] subsystem in operation. (ventilation) OR	Immediately	(1)

Ventilation
[Control Room AC] System (1)
3.7.5

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.17.A.2.c		C.2.1 Suspend movement of [recently] irradiated fuel assemblies in the [secondary] containment. AND	Immediately	(1)
		C.2.2 Initiate action to suspend OPDRVs.	Immediately	
3.17.A.3.a, 3.17.A.3.b	D. Two [control room AC] (ventilation) subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately	(1)
3.17.A.3.c	E. Two [control room AC] (ventilation) subsystems inoperable during movement of [recently] irradiated fuel assemblies in the [secondary] containment or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ---------- E.1 Suspend movement of [recently] irradiated fuel assemblies in the [secondary] containment. AND	Immediately	(1)
		E.2 Initiate actions to suspend OPDRVs.	Immediately	

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
DOC M.2	SR 3.7.5.1	Verify each [control room AC] (ventilation) subsystem has the capability to remove the assumed heat load.	[18] (24) months	(1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.5, CONTROL ROOM VENTILATION SYSTEM

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

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.5 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Control Room AC System provides temperature control for the control room following isolation of the control room.

The Control Room AC System consists of two independent, redundant subsystems that provide cooling and heating of recirculated control room air. Each subsystem consists of heating coils, cooling coils, fans, chillers, compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.

The Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. A single subsystem provides the required temperature control to maintain a suitable control room environment for a sustained occupancy of 12 persons. The design conditions for the control room environment are 76°F 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).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for a 30 day continuous occupancy.

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. 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. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

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

1

INSERT 1

The system is designed to maintain the control room boundary at 78°F during the summer and 72°F in the winter. The maximum design condition in the control room and most of the EFT is 104°F.

BASES

LCO

Two independent and redundant subsystems of the [Control Room AC] System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

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.

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.

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:

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

b. During movement of [recently] irradiated fuel assemblies in the [secondary] containment. [Due to radioactive decay, the Control Room AC System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days).]

ACTIONS

A.1

With one [control room AC] subsystem inoperable, the inoperable [control room AC] subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE [control room AC] subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

Ventilation

[Control Room AC] System (1)
B 3.7.5

BASES

ACTIONS (continued)

B.1 and B.2

ventilation

In MODE 1, 2, or 3, if the inoperable [control room AC] subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. (1)

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

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

ventilation

During movement of [recently] irradiated fuel assemblies in the [secondary] containment 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, that no failures that would prevent actuation will occur, and that any active failure will be readily detected. (1)

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, movement of [recently] 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)

BASES

ACTIONS (continued)

D.1

ventilation
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.

E.1 and E.2

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

ventilation
During movement of [recently] irradiated fuel assemblies in the [secondary] containment 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 require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, handling of [recently] 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

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

REFERENCES
U
1. [F]SAR, Section [6.4].
6.7

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.5 BASES, CONTROL ROOM VENTILATION 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. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.5, CONTROL ROOM VENTILATION SYSTEM

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 6

ITS 3.7.6, Main Condenser Offgas

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.8 MAIN CONDENSER OFFGAS	**4.8 MAIN CONDENSER OFFGAS**
Applicability:	Applicability:
Applies to the radioactive release rate from the main condenser offgas.	Applies to the sampling and monitoring of radioactive effluents discharged from the main condenser offgas.
Objective:	Objective:
To limit the doses received at the site boundary from main condenser offgas in the event that effluent is discharged with less than full treatment.	To limit the doses received at the site boundary from main condenser offgas in the event that effluent is discharged with less than full treatment.
Specification:	Specification:
3.7.6 A. Main Condenser Offgas Activity	A. Main Condenser Offgas Activity
Applicability — 1. Whenever the Steam Jet Air Ejectors (SJAEs) are in operation, the gross gamma activity rate of the noble gases measured at the main condenser offgas system pretreatment monitor station shall be	NOTE: Not required to be performed until 31 days after the SJAEs are in operation.
LCO 3.7.6 — ≤ 2.6 x 10^5 µCi/second after a decay of 30 minutes.	1. Verify the gross gamma activity rate of the noble gases is ≤ 2.6 x 10^5 µCi/second after a decay of 30 minutes:
	a. Once every month.
	b. 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.
ACTION A — 2. When the gross gamma activity rate of the noble gases is not within the limit of 3.8.A.1 above, restore gross gamma activity rate of the noble gases to within the limit within 72 hours.	SR 3.7.6.1

A.2

A.2

any main steam line not isolated and

3.8/4.8

192 07/25/01
Amendment No. ~~15, 40, 120,~~ 121

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION		4.0 SURVEILLANCE REQUIREMENTS
ACTION B	3. When 3.8.A.2 cannot be met, either: a. Isolate all main steam lines within 12 hours; or b. Isolate the SJAEs within 12 hours; or c. Be in hot shutdown within 12 hours and cold shutdown within the following 24 hours.	

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DISCUSSION OF CHANGES
ITS 3.7.6, MAIN CONDENSER OFFGAS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.8.A.1 requires the main condenser offgas activity to be within limit "Whenever the Steam Jet Air Ejectors (SJAEs) are in operation." CTS 4.8.A requires the main condenser offgas activity Surveillance to be performed "after the SJAEs are in operation." ITS LCO 3.7.6 also requires the main condenser offgas activity to be within limit, however the Applicability is MODE 1, and MODES 2 and 3 with any main steam line not isolated and steam jet air ejector (SJAE) in operation. ITS SR 3.7.6.1 includes the same Surveillance requirement to verify the main condenser offgas activity, however a Note has been included that requires the Surveillance to be performed "after any main steam line is not isolated and SJAE in operation." This changes the CTS by clarifying that the LCO is always applicable in MODE 1, and only in MODES 2 and 3 when any main steam line is opened and a SJAE is in operation, and also allows the Surveillance to be performed only after both a main steam line is opened and a SJAE is in service.

The purpose of CTS 3.8.A.1, in part, is to state when the main condenser offgas activity is applicable. The purpose of CTS 4.8.A, in part, is to state when the Surveillance is required to be performed. This change clarifies that the LCO is always applicable in MODE 1, and only in MODES 2 and 3 when any main steam line is opened and a SJAE is in operation, and also allows the Surveillance to be performed only after both a main steam line is opened and a SJAE is in service. The SJAEs cannot be placed in service without main steam pressure (i.e., any main steam line not isolated). This proposed Applicability is consistent with the CTS 3.8.A.3 default action to isolate the main steam lines, isolate the SJAEs, or be in cold shutdown. The Surveillance will be required to be performed at the same time as in the CTS since the SJAE cannot be placed in service without steam line pressure. This change is also acceptable because it matches the Actions and Applicability. In addition, the main condenser offgas activity limit is only necessary when the main steam is directed to the main condenser and the offgas system is in operation since this is when there is a potential for the Main Condenser Offgas System to rupture. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

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DISCUSSION OF CHANGES
ITS 3.7.6, MAIN CONDENSER OFFGAS

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.8.A

3.7.6 Main Condenser Offgas

3.8.A.1 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]. (1)

offgas

260

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.8.A.2	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.A.3	B. Required Action and associated Completion Time not met.	B.1 [Isolate all main steam lines.] (1) OR B.2 Isolate SJAE. OR B.3.1 Be in MODE 3. AND B.3.2 Be in MODE 4.	12 hours [(1) 12 hours 12 hours 36 hours

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.8.A	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)
		Verify the gross gamma activity rate of the noble gases is ≤ [240] (260) mCi/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 (1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.6, MAIN CONDENSER OFFGAS

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

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

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 condenser. Air and noncondensible gases are collected in the 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. at the outlet of the offgas condenser (1)

APPLICABLE SAFETY ANALYSES

INSERT 1

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). 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. (1)

(1)

The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

INSERT 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 this requirement ([2436] MWt x 100 µCi/MWt-second = [240] mCi/second). (1) (2)

must be ≤ 260

after decay of 30 minutes

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, steam is not being exhausted to the main condenser and the requirements are not applicable. (2)

1

INSERT 1

an event that inadvertently releases the main condenser effluent directly to the environment without treatment.

1

INSERT 2

an event that inadvertently releases the main condenser effluent directly to the environment without treatment,

BASES

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. INSERT 3 (1)

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 the source of the 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. (2)

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR, 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.

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. (2)

1 **INSERT 3**

an event that inadvertently releases the main condenser effluent directly to the environment without treatment

BASES

REFERENCES

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

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.6 BASES, MAIN CONDENSER OFFGAS

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.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.6, MAIN CONDENSER OFFGAS

There are no specific NSHC discussions for this Specification.

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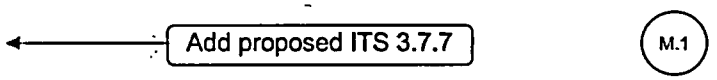
ATTACHMENT 7

ITS 3.7.7, Main Turbine Bypass System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)



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DISCUSSION OF CHANGES
ITS 3.7.7, MAIN TURBINE BYPASS SYSTEM

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M.1 The CTS does not have any requirements for Main Turbine Bypass System. ITS LCO 3.7.7 requires the Main Turbine Bypass System to be OPERABLE. This changes the CTS by incorporating the requirements of ITS 3.7.7. Appropriate ACTIONS and Surveillance Requirements are also provided.

The Monticello transient analyses credit the opening of the turbine bypass valves during certain events. If the Main Turbine Bypass System is inoperable, the safety analyses are not met. This change is acceptable since the Monticello transient analyses credit the Main Turbine Bypass System; therefore, the LCO ensures that the assumptions of the safety analyses are conserved. This change is designated as more restrictive because it adds new requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.7.7 The Main Turbine Bypass System (1)

DOC M.1 LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

[a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the [COLR]; and] (4)

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

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].	A.1 [Satisfy the requirements of the LCO or restore Main Turbine Bypass System to OPERABLE status].	2 hours	(4)
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 (92) days	(3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
DOC M.1	SR 3.7.7.2	Perform a system functional test.	[18] 24 months	2
DOC M.1	SR 3.7.7.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	[18] 24 months	2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.7, MAIN TURBINE BYPASS SYSTEM

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

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

3. ISTS SR 3.7.7.1 has a Frequency of 31 days. The Frequency is being changed to 92 days in ITS SR 3.7.7.1. The Monticello CTS does not require this test to be performed, however, the turbine bypass valves are currently cycled every 92 days in accordance with plant procedures, and this 92 day Frequency has been shown to be acceptable.

4. Monticello is not currently analyzed to operate with the Main Turbine Bypass System inoperable. Therefore, this bracketed allowance has not been adopted.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

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]% 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 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 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 FSAR, 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. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

14

two

Electrical Pressure Regulator or the Mechanical Pressure Regulator

U

7.7.2.2

reducer

is reduced

1 2 2 2 2

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the turbine generator load rejection transient, as discussed in the FSAR, Section [15.1.1] (Ref. 2). 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 APLHGR and MCPR penalties.

The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

s

U

feedwater controller failure (maximum demand) and pneumatic system degradation, turbine trip with bypass - reduced scram speeds

s 14.4.4 and 14A.4 (Refs. 2 and 3), respectively

2 1 3

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 APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL

1 3

BASES

LCO (continued)

POWER RATIO (MCPR)") may be applied to allow this LCO to be met.] The APLHGR and MCPR limits for the inoperable Main Turbine Bypass System are 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 analysis (Ref. 2). (3) (e) (s. 2 and 3) (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 not violated during the turbine generator load rejection transient. As discussed in the Bases for LCO 3.2.1 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) (s)

feedwater controller failure (maximum demand) and pneumatic system degradation, turbine trip with bypass - reduced scram speeds

ACTIONS

A.1 (1)

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR and 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 APLHGR and 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. (3) (3) (1)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR and 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. (3) (s) (1)

feedwater controller failure (maximum demand) and pneumatic system degradation, turbine trip with bypass - reduced scram speeds

BASES

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 31 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 31 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 [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.

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 [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.

REFERENCES

1. FSAR, Section [7.7.4].
2. FSAR, Section [15.1.1].
3. USAR, Section 14A.4.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.7 BASES, 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. Change made to reflect changes made to the Specification.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.7, MAIN TURBINE BYPASS SYSTEM

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 8

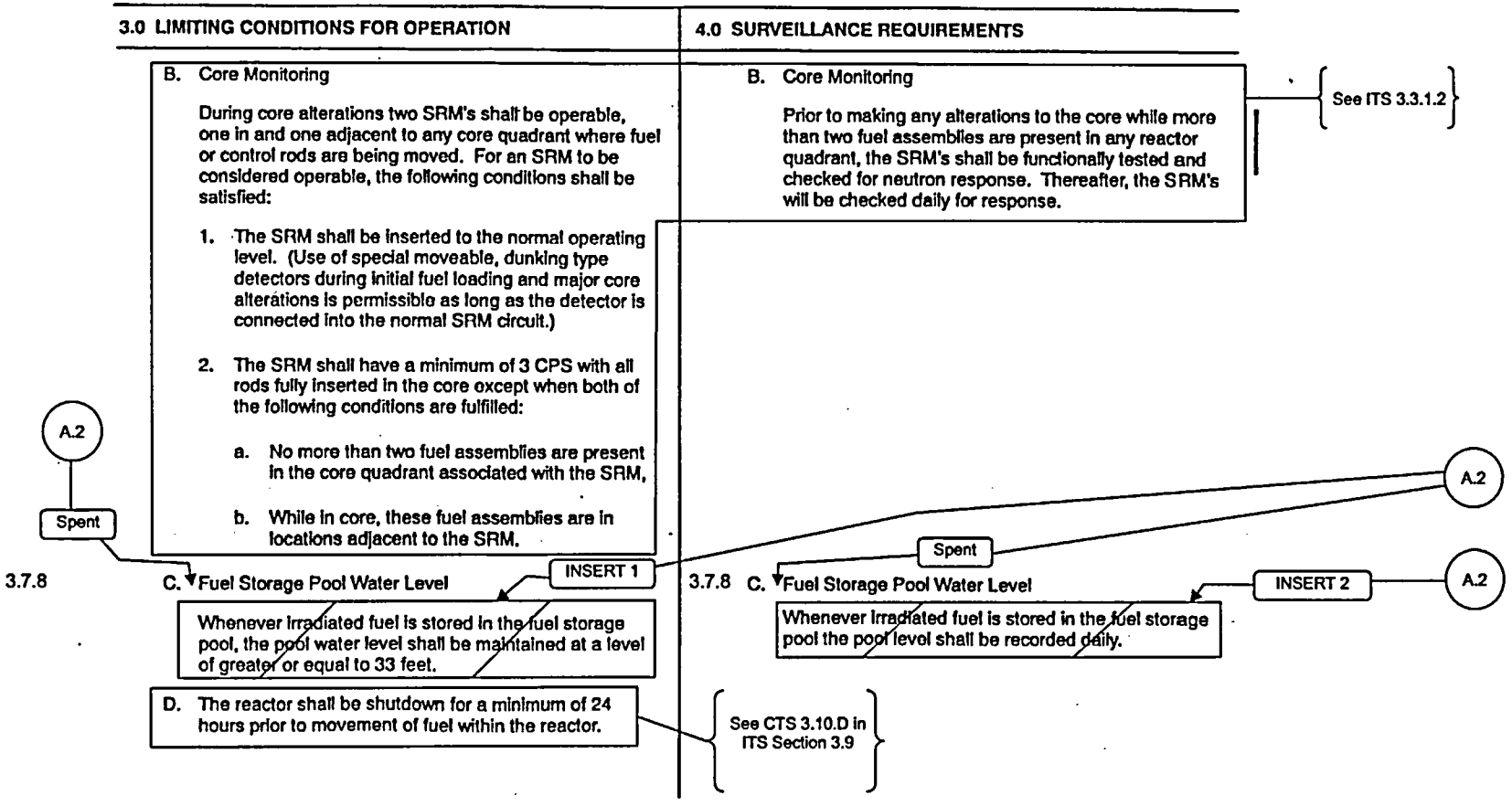
ITS 3.7.8, Spent Fuel Storage Pool Water Level

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A.1

ITS



3.10/4.10

207

Amendment No. 20, 123

10/26/01

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ITS 3.7.8

A.1

ITS

A.2 **INSERT 1**

Applicability — During movement of irradiated fuel assemblies, the spent
LCO 3.7.8 — fuel storage pool water level shall be maintained ≥ 37 feet above the bottom of the spent fuel storage pool.

LCO 3.0.3 is not applicable. — A.3

ACTION A — If the spent fuel storage pool water level is made or found not to be within limits, immediately suspend movement of irradiated fuel assemblies.

A.2 **INSERT 2**

SR 3.7.8.1 — Verify that the spent fuel storage pool water level is ≥ 37 feet above the bottom of the spent fuel storage pool:

7 days — L.1

1. Once every 24 hours, during movement if irradiated fuel assemblies, or

2. Once every 7 days, when irradiated fuel assemblies are stored in the spent fuel storage pool. — A.4

Insert Page 207

DISCUSSION OF CHANGES
ITS 3.7.8, SPENT FUEL STORAGE POOL WATER LEVEL

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 These changes to CTS 3.10.C and CTS 4.10.C are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-05-013, from Thomas J. Palmisano (NMC) to USNRC, dated April 12, 2005. As such, these changes are administrative.

A.3 Under certain conditions, CTS 3.10.C, in part, requires immediate suspension of movement of irradiated fuel assemblies. ITS 3.7.8 ACTION A includes the same requirement, however a Note has been added that states that LCO 3.0.3 is not applicable. This changes the CTS by adding this Note.

The purpose of CTS 3.10.C, in part, is to provide the appropriate actions when the spent fuel storage pool water level is not within limit. This change adds a Note that states LCO 3.0.3 is not applicable. This Note has been added because ITS LCO 3.0.3 has been added to ITS Section 3.0 in accordance with ITS Section 3.0 DOC M.1. This Note is necessary because if moving irradiated fuel assemblies while in MODE 4 or 5, ITS LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. Since ITS LCO 3.0.3 is not currently included in the CTS, this change is considered administrative. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.4 CTS 4.10.C.2 requires verification that the spent fuel storage pool water level is within limit once every 7 days when irradiated fuel assemblies are stored in the spent fuel storage pool. This Surveillance is not included in ITS 3.7.8. This changes the CTS by deleting this Surveillance.

The purpose of CTS 4.10.C.2 is to ensure there is sufficient water in the spent fuel storage pool when irradiated fuel assemblies are stored in the spent fuel storage pool. The applicability specified in CTS 3.10.C is "during movement of irradiated fuel assemblies," not "when irradiated fuel assemblies are stored in the spent fuel storage pool." CTS 4.10.C.1 requires verification that the spent fuel storage pool water level is within limit once every 24 hours during movement of irradiated fuel assemblies. CTS 4.0.C states that whenever the plant condition is such that a system or component is not required to be OPERABLE, the Surveillance testing associated with that system or component may be discontinued. This allowance is retained in SR 3.0.1. Therefore, CTS 4.10.C.2 is not required to be met or performed. The requirements in CTS 3/4.10.C have

recently been proposed to be modified (as discussed in DOC A.2) to ensure that when irradiated fuel assemblies are being moved over the spent fuel storage pool, the spent fuel storage pool level is adequate to ensure the consequences of fuel handling accident over the spent fuel pool will be bounded by the safety analysis. This specific Surveillance should not have been added to the Technical Specifications as part of this Amendment request. Therefore, this Surveillance is not required and is deleted. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 7 - Relaxation of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.10.C.1 requires a verification every 24 hours that the spent fuel storage pool water level is within the limit. ITS SR 3.7.8.1 requires verifying the spent fuel storage pool water level is within the limit every 7 days. This changes the CTS by extending the Surveillance Frequency from 24 hours to 7 days.

The purpose of CTS 4.10.C.1 is to ensure the spent fuel storage pool water level is within limit. This change extends the Surveillance Frequency from 24 hours to 7 days. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of reliability. The change in Surveillance Frequency from 24 hours to 7 days maintains a Frequency that is adequate to trend changes in the spent fuel storage pool water level. In addition, a control room alarm exists to alert the operator to a decreasing water level. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.10.C 3.7.8 Spent Fuel Storage Pool Water Level

3.10.C 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.

37 1 above the bottom of 2

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L.1 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.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.10.C 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.	7 days

37 above the bottom of 1 2

JUSTIFICATION FOR DEVIATIONS
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. ISTS LCO 3.7.8 requires the spent fuel storage water level to be 23 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks and ISTS SR 3.7.8.1 requires verification that the spent fuel storage water level to be 23 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. The value of 23 ft is bracketed. ITS LCO 3.7.8 requires the spent fuel storage water level to be $\geq$ 37 ft above the bottom of the spent fuel storage pool and ITS SR 3.7.8.1 requires verification that the spent fuel storage level is $\geq$ 37 ft above the bottom of the spent fuel storage pool. The proposed level will ensure there is $\geq$ 21 ft 4 inches of water over the top of irradiated fuel assemblies seated in the spent storage pool racks. The proposed level is acceptable because the safety analyses shows that the radiological consequences of a refueling accident occurring over the core bounds the same accident over the spent fuel storage pool. The value and method of referencing the spent fuel storage pool level in ITS LCO 3.7.8 and ITS SR 3.7.8.1 is consistent with a recent NMC License Amendment Request, as provided in NMC letter L-MT-05-013, from Thomas J. Palisano (NMC) to USNRC, dated April 12, 2005.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

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

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.

The spent fuel storage pool water level satisfies Criteria 2 and 3 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.

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.

BASES

ACTIONS

A.1

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

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. Suspension of this activity shall not preclude completion of movement of an irradiated 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.

REFERENCES

1. FSAR, Section [7]. (U) — 10.2.1 — (1) (2)
2. FSAR, Section [15/1.4]. (U) — 14.7.6 — 10 CFR 50.67 — (1) (2)
3. NUREG-0800, Section 15.7.4, Revision 1, July 1981. — (1)
4. 10 CFR 100. — (1)
5. Regulatory Guide 1.25, March 1972. (4) — (1)
6. FSAR, Section [9.1/2.2.2]. (5) (U) — 14.7.6.3.1 — (1) (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.8 BASES, 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

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.8, SPENT FUEL STORAGE POOL WATER LEVEL

There are no specific NSHC discussions for this Specification.

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