JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.4

Control Room Air Conditioning (AC) System

MARKUP OF NUREG-1433, REVISION 1, BASES

PA1 4 PA2

B 3.7 PLANT SYSTEMS

B 3.7.[5] [Control Room Air Conditioning (AC)] System

4 PA2

BASES

PA1

BACKGROUND

The [Control Room AC] System provides temperature control for the control room ~~following isolation of the control room~~. Insert BK6D-1 DB1

PA1 DB1

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. DB3

DB1 filtered and

A heater is located in the ductwork associated with each control room area. DB3

PA1

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 7[5]°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). 20 5 DB1 PA1 9.9.3.11 DB2

Insert BK6D-2 DB1

PA3 U

APPLICABLE SAFETY ANALYSES

The design basis of the [Control Room AC] System is to maintain the control room temperature for a 3[0] day continuous occupancy. PA1 0 DB1

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 PA1

PA4 active component

(continued)

DB1

INSERT BKGD-1

while the Control Room Emergency Ventilation Air Supply (CREVAS) System (a mode of the Control Room AC) provides a radiologically controlled environment (refer to the Bases of for LCO 3.7.3, "Control Room Emergency Ventilation Air Supply (CREVAS) System).

DB1

INSERT BKGD-2

This can be accomplished when a control room chiller is providing the cooling medium to the cooling coils of an air handling unit. The control room chillers are non-safety related; however, the Control Room AC System still meets safety-related QA Category I requirements when the Emergency Service Water System is aligned to directly supply the cooling coils. The resulting maximum Control Room environmental conditions when the Emergency Service Water System is supplying the air handling unit cooling coils is 104°F assuming a lake temperature of 85°F. This satisfies the OPERABILITY requirements of the Control Room equipment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The [Control Room AC] System satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) (Ref. 2) X1 PA1

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 active component failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits. DB1 PA1 PA4

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 air handling units, ~~cooling coils~~, ~~fans~~, ~~chillers~~, ~~compressors~~, recirculation exhaust fans, air handling unit fans, ductwork, dampers, and associated instrumentation and controls. INSERT LCO PA1 DB1

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. PA1

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: PA1

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

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the [secondary] containment. PA1

(continued)

(DB1)

INSERT LCO

The cooling coils of the air handling units may be cooled by the Control Room chillers, but to satisfy this LCO, the Emergency Service Water System must be capable of alignment to provide cooling water directly to the cooling coils.

BASES (continued)

ACTIONS

A.1

(PA3) plant

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. (PA1)

(PA4) active component

B.1 and B.2

(PA1) (PA3) plant

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.

plant (PA3)

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

(PA4) LCO 3.0.3 is not applicable while in MODE 4 and 5. However, since irradiated fuel assembly movement can occur in MODES 1, 2, or 3

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

During movement of irradiated fuel assemblies in the [secondary] containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE [control room AC] subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, (PA1) (PA1)

(continued)

BASES

ACTIONS

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

that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit plant PA3 in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the [secondary] containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action~~s~~ must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action~~s~~ must continue until the OPDRVs are suspended. PA1 PA5

D.1

If both [control room AC] subsystems are inoperable in MODE 1, 2, or 3, the [Control Room AC] System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately. PA1

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

PA4 LCO 3.0.3 is not applicable when in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2 or 3

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

During movement of irradiated fuel assemblies in the [secondary] containment, during CORE ALTERATIONS, or during OPDRVs, with two [control room AC] subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might PA1

(continued)

BASES

PA3

ACTIONS

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

plant

require isolation of the control room. This places the ~~unit~~ in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the [secondary] containment must be suspended PA1 immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action~~s~~ must be initiated PA5 immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action~~s~~ must continue until the OPDRVs are suspended. PA5 PA4

PA2

SURVEILLANCE REQUIREMENTS

SR 3.7.~~5~~.1 4

PA1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the [safety analyses]. The SR consists of a combination of testing and calculation. The ~~[18]~~ month Frequency is appropriate since significant degradation of the [Control Room AC] System is not expected over this time period.

with ESW providing water to the cooling coils of the air handling units

24

X2

It is acceptable to perform the test using chilled water as the cooling medium to the cooling coils, but a calculation must be performed to ensure that the heat load can be removed with ESW at 85°F

PA4

PA1

PA3

REFERENCES

1. [U]FSAR, Section [~~6.4~~]. 9.9.3.11 DB2

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

X1

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.4

Control Room Air Conditioning (AC) System

JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.4 - CONTROL ROOM AIR CONDITIONING (AC) SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The brackets have been removed and the proper plant specific information has been provided.

PA2 ISTS 3.7.5 has been renumbered as ISTS 3.7.4 to reflect deletion of ISTS 3.7.3. The Surveillance has been renumbered as a result of this change.

PA3 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA4 Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

PA5 Editorial change made to be consistent with the Specification.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 Changes have been made to reflect the plant specific design and analysis.

DB2 The brackets have been removed and the proper plant specific references provided.

DB3 The design does not include redundant heating coils therefore reference to "heating" and "heating coils" have been deleted. However, a heater is located in the ductwork associated with each Control Room area and each heater is supplied by a safety related power supply. This detail was added to the Background Section.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

<u>DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE</u> (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X2 The bracketed Surveillance Frequency of 18 months in ITS SR 3.7.4.1 has been removed and the Frequency changed to 24 months in conjunction with the current operating cycle. This proposed Frequency is considered adequate since significant degradation of the Control Room AC System is not expected over this time period.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.4

Control Room Air Conditioning (AC) System

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two control room AC subsystems shall be OPERABLE.

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

ACTIONS

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

(continued)

ACTIONS (continued)

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

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	24 months

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Control Room AC System provides temperature control for the control room while the Control Room Emergency Ventilation Air Supply (CREVAS) System (a mode of the Control Room AC) provides a radiologically controlled environment (refer to the Bases of for LCO 3.7.3, "Control Room Emergency Ventilation Air Supply (CREVAS) System").

The Control Room AC System consists of two, redundant subsystems that provide cooling of recirculated control room air. Each subsystem consists of cooling coils, fans, chillers, compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control. A heater is located in the ductwork associated with each control room area.

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 20 persons. The design conditions for the control room environment are 75°F and 50% relative humidity. This can be accomplished when a control room chiller is providing the cooling medium to the cooling coils of an air handling unit. The control room chillers are non-safety related; however the Control Room AC System still meets safety-related QA Category I requirements when the Emergency Service Water System is aligned to directly supply the cooling coils. The resulting maximum control room environmental conditions when the Emergency Service Water System is supplying the air handling unit cooling coils is 104°F assuming a lake temperature of 85°F. This satisfies the OPERABILITY requirements of the control room equipment. The Control Room AC System operation in maintaining the control room temperature is discussed in the UFSAR, Section 9.9.3.11 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for a 31 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 active component 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) (Ref. 2).

LCO

Two redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single active component 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 air handling units, recirculation exhaust fans, air handling unit fans, ductwork, dampers, and associated instrumentation and controls. The cooling coils of the air handling units may be cooled by the control room chillers, but to satisfy this LCO the Emergency Service System must be capable of alignment to provide cooling water directly to the cooling coils.

(continued)

BASES (continued)

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);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

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 plant 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 active component 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.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable control room AC subsystem cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

MODE that minimizes risk. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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

LCO 3.0.3 is not applicable while in MODE 4 and 5. However, since irradiated fuel assembly movement can occur in MODES 1, 2, or 3 the Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE control room AC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

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

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

(continued)

BASES

ACTIONS (continued)

D.1

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

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

LCO 3.0.3 is not applicable when in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3 the Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

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

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

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses with ESW providing water to the cooling coils of the air handling units. The SR consists of a combination of testing and calculation. It is

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.4.1 (continued) acceptable to perform the test using chilled water as the cooling medium to the cooling coils, but a calculation must be performed to ensure that the heat load can be removed with ESW at 85°F. The 24 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.
REFERENCES	1. UFSAR, Section 9.9.3.11. 2. 10 CFR 50.36(c)(2)(ii).

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.7.5

A1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

[3.7.5] 3.5 **MAIN CONDENSER STEAM JET AIR EJECTOR (SJAE)**

[Applicability]

Applicability

Applies to main condenser offgas discharge rate for noble gases when the reactor is in the run, startup/hot standby or hot shutdown mode of operation and the SJAE is in service.

Objective

To ensure that the SJAE release rates are maintained at a level compatible for further treatment and release.

Specifications

A2

[LCO 3.7.5] a. The gross radioactivity (~~beta and/or~~ gamma) rate of noble gases measured at the SJAE is ~~given on Table 3.10-1~~.

≤ 600,000 µCi/sec

L2

add Note to SR 3.7.5.1

L3

3.5 **MAIN CONDENSER STEAM JET AIR EJECTOR (SJAE)**

Applicability

Applies to the point of discharge at the SJAE when the reactor is in the run, startup/hot standby or hot shutdown mode of operation and the SJAE is in service.

Objective

To ensure that the SJAE release rates are properly monitored.

Specifications

A2

a. The gross radioactivity (~~beta and/or~~ gamma) rate of noble gases from the SJAE shall be determined to be within the limits of Specification 3.5.a by performing an isotopic analysis of a representative sample of gases taken at the discharge (prior to dilution and/or discharge) of the SJAE, or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity) as follows: LA1 {SR 3.7.5.1}

[SR 3.7.5.1 Frequency] 1. At least monthly.

[SR 3.7.5.1 Frequency] M1 2. With the SJAE Monitor reading at 5,000 µCi/sec or greater, within 4 hours following an increase of greater than or equal to 50% (after factoring out increases due to changes in thermal power level) in the nominal steady state fission gas release from the primary coolant.

Amendment No. ~~98~~, ~~127~~, 211

[RETS]

Page 1 of 6

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. Except as specified in 1. and 2. below, both SJAE system radiation monitors shall be operable during reactor power operation. The trip time delay setting for closure of the SJAE isolation valve shall not exceed 15 min. 1. A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function. Otherwise, in the event that one of the two SJAE radiation monitors is made or found to be inoperable, continued reactor power operation is permissible provided that the inoperable monitor is tripped. 2. Upon the loss of both SJAE system radiation monitors, either temporarily monitor radiation levels at the SJAE or initiate an orderly shutdown and have the main steam isolation valves closed within 8 hours.	b. Operation of the SJAE radioactive offgas monitors shall be verified by performing instrument surveillance as specified on Table 3.10-2.

R1

Specification 3.7.5

A1

JAFNPP

R1

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design	Action
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
(j)	SJAE Radiation Monitors [LCO 3.7.5]	≤ 500,000 μCi/sec L2	~~2~~	(e)
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)
(k)	Main Control Room Ventilation	≤4 x 10³ cpm[a]	1	(g)
(h)	Mechanical Vacuum Pump Isolation	≤3 x Normal Full Power Background	4	(h)

[LCO 3.7.5]

See ITS: 3.3.6.2

R1

See CTS RETS 3.1

See ITS 3.3.7

See ITS: 3.3.7.2

NOTES FOR TABLE 3.10-1

(a) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function, that is, trip capability is maintained.
An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to operable status within 24 hours, or the indicated action shall be taken.

See ITS: 3.3.6.2
CTS RETS: 3.1

(b) Trip level setting is in accordance with the methods and procedures of the ODCM.

(c) Cease operation of the refueling equipment. See ITS: 3.3.6.2

(d) Isolate secondary containment and start the SBGTS.

[ACTION B]

[ACTION A] (e) Bring the SJAE release rate below the trip level within 72 hours or isolate either the SJAE or all main steam lines within the next 12 hours.

add RA B.3.1 and B.3.2 L1

Specification 3.7.5

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Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

NOTES FOR TABLE 3.10-1 (cont'd.)

(f) Refer to Appendix B LCO 3.1.c and LCO 3.1.d. See CTS RETS 3.1

(g) Control room isolation is manually initiated. See ITS: 3.3.7.1

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required. see ITS: 3.3.2.2

(i) Conversion factor is 8.15 x 10^7 cpm - 1μCi/cc. see ITS: 3.3.7.1

(j) See RETS LCO 3.5.b.1 and 3.5.5.2. (R1)

(k) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip Function in the tripped condition, or the indicated action shall be taken. see ITS: 3.3.7.1

Specification 3.7.5

A1

R1

TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

Instrument Channels	Instrument Check[b]	Instrument Channel Functional Test[c]	Instrument Channel Calibration	Logic System Function Test[f][g]	
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	- -	See CTS RETS 3.1
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	- -	See ITS: 3.3.6.2
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months	
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	- -	see CTS RETS: 3.1
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	- -	
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months	R1
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	- -	See ITS: 3.3.7.1
Mechanical Vacuum Pump Isolation[g]	- -	- -	- -	Once per 24 Months	see ITS 3.3.7.2
Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months	see CTS RETS: 2.1
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	- -	
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	- -	
Normal Service Water Effluent	Daily	Quarterly	Quarterly	- -	
SBGTS Actuation	- -	- -	- -	Once per 24 Months	See ITS: 3.6.4.2 3.6.4.3

R1

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release. see CTS RETS : 2.1

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

R1

(f) ~~Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.~~

see ITS: 3.3.7.2

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

R1

(h) ~~The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.~~

(i) ~~This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.~~ R1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.7.5 - MAIN CONDENSER STEAM JET AIR EJECTOR OFFGAS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS RETS 3.5.a (LCO and Surveillance Requirement) specifies the limitations and surveillance requirements for gross radioactivity (beta and/or gamma) rate of noble gases. ITS 3.7.5 only places limitations on the gross gamma activity rate of the noble gases instead of "beta and/or gamma". The option to measure the beta rate of activity has been deleted since JAFNPP utilizes the gross gamma approach which is consistent with industry practice. This change is considered administrative and is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS RETS 3.5.a.2 (Surveillance Requirement) requires the gross activity to be determined within 4 hours following an increase of greater than 50% (factoring out increases due to changes in thermal power level) in the nominal steady state fission gas release. In ITS SR 3.7.5.1, this frequency has been changed to include an increase equivalent to 50%. This is an inconsequential change that is considered more restrictive since technically it increases the range of releases to be considered. However, no additional performances of the Surveillance would be expected since the increase is insignificant. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details in CTS RETS 3.5.a (Surveillance Requirement) of the method of performing Surveillance (by performing an isotopic analysis on a representative sample of gases) is proposed to be relocated to the Bases. These details are not necessary to ensure the air ejector offgas activity rate limit is maintained. The requirements of Specification 3.7.5 and SR 3.7.5.1 are adequate to ensure the air ejector offgas activity rate is maintained within the limit. The requirement to perform an isotopic analysis of a representative sample of gases is included in the Bases of ITS SR 3.7.5.1. As such, these details are not

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS RETS Table 3.10-1 Note (e) requires the plant to isolate the SJAE or all main steam lines within the next 12 hours if the SJAE release rate is not below the trip level within 72 hours. These actions have been included in ITS 3.7.5 as ACTION A and B. An option has been included in proposed ACTION B allowing the plant to be in MODE 3 within 12 hours and in MODE 4 within 36 hours. This is acceptable since these alternative actions will result in a power reduction which will reduce the coolant activity levels and place the plant in a condition where the Specification does not apply (MODE 4). This change is less restrictive on plant operation since the option is provided and the overall time to exit the applicability is longer.

L2 CTS RETS 3.5.a (LCO) specifies that the limits of gross radioactivity rate of noble gases is given on Table 3.10-1. CTS RETS Table 3.10-1 specifies the trip level setting for the SJAE Radiation Monitors. This limit has been increased from 500,000 to 600,000 μCi/sec consistent with the value used in the Offgas System Failure accident of UFSAR, Section 11.4.7.2. Since a higher value has been included in proposed Specification 3.7.5 this change is considered less restrictive but acceptable since the limit is consistent with the analysis. The trip level setting of the SJAE Radiation Monitors has been relocated as identified in the Discussion of Changes for CTS 3/4.2.D, "Radiation Monitoring Systems - Isolation and Initiation Functions". This change to include the analytical limit in the ITS is consistent with the requirements and format of NUREG-1433, Revision 1.

L3 CTS RETS 3.5.a (Surveillance Requirement) must be performed prior to entry into the mode of applicability in accordance with CTS 3.0.D. A Note has been added to CTS RETS 3.5.a (proposed SR 3.7.5.1) which clarifies when the surveillance must be performed. The Note specifies that the surveillance is not required to be met until after 31 days after any main steam line is not isolated and the SJAE are in operation since in this condition radioactive fission gases may be in the Main Steam Offgas System at significant rates. This change is considered less restrictive since CTS 4.0.D (ITS SR 3.0.4) requires the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 (continued)

surveillance to be met prior to entry into the modes of Applicability. This change is acceptable since a test with the valves isolated provides no meaningful information. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - RELOCATIONS

R1 CTS RETS 3.5.1.b (LCO and Surveillance Requirement), CTS RETS Table 3.10-1, and Table 3.10-2 specify the requirements for the Steam Jet Air Ejector (SJAE) System radiation monitors. This instrumentation is neither a safety system nor is it connected to the reactor coolant. The primary function of this instrumentation is to show conformance to the discharge limits of 10 CFR Part 20. This instrumentation is not installed to detect excessive reactor coolant leakage. The SJAE System monitors are used to provide a continuous check on the releases of radioactive gaseous effluents from the Main Condenser Steam Jet Air Ejector. These Technical Specifications require the Licensee to maintain Operability of various effluent monitors and establish setpoints in accordance with the Offsite Dose Calculation Manual (ODCM). The alarm/trip setpoints are established to ensure that the alarm/trip will occur to prevent exceeding the limits of 10 CFR 20. Plant Design Basis Accident (DBA) analyses do not assume any action, either automatic or manual, resulting from the Steam Jet Air Ejector (SJAE) monitors. ITS 3.7.5, Main Condenser Steam Jet Air Ejector Offgas, will be included in the ITS to ensure the SJAE Offgas failure event will remain within the calculated values of UFSAR, Section 11.4.7.2. Additional administrative controls are also proposed to be added to the Technical Specifications to ensure compliance with the applicable regulatory requirements is maintained. ITS 5.5.1 specifies that future changes to the ODCM will be reviewed to ensure that such changes will "maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations."

CTS RETS 3.5.1.b (LCO and Surveillance Requirement), CTS RETS Table 3.10-1 and able 3.10-2 do not identify a parameter which is an initial condition or assumption for a DBA or transient, identify a significant abnormal degradation of the reactor coolant pressure boundary, provide any mitigation of a design basis event and is not a structure system or component which operating experience or PRA has shown to be significant to public health and safety. Therefore, the requirements specified in

TECHNICAL CHANGES - RELOCATIONS

R1 (continued)

CTS RETS 3.5.1.b (LCO and Surveillance Requirement), CTS RETS Table 3.10-1 and Table 3.10-2 did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the JAFNPP Technical Specifications and will be relocated to the ODCM. Changes to the ODCM will be controlled by the provisions of the ODCM change control process described in Chapter 5 of the ITS. This change is consistent with Generic Letter 89-01 for removal of Radiological Effluent Technical Specification (RETS) and relocation to the ODCM.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.7.5 - MAIN CONDENSER STEAM JET AIR EJECTOR OFFGAS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   Alternative ACTIONS have been provided to exit the applicability by taking the plant to MODE 4 instead of isolating main steam lines or SJAE. High offgas activity is not an initiator of the analyzed offgas system failure or any design basis accident or transient. Therefore this change does not involve a significant increase in the probability of an accident previously evaluated. The proposed ACTIONS remove the plant from the conditions in which an accident in the offgas system may result in exceeding the limits. Although the time allowance to achieve MODE 4 is longer than allowed by CTS RETS Table 3.10-1 Note (e), the consequences of an accident occurring during this additional time period is the same as in the current time allowance provided. However, the change also added an additional requirement to be in MODE 3 in 12 hours, therefore, the consequences of an accident will be less with a shutdown already in progress. In addition, if offgas activity exceeded the setpoint of the offgas radiation monitors for more than 15 minutes, the offgas outlet isolation valves will automatically close. The closure of these valves may cause a loss of condenser vacuum, resulting in a main steam isolation and subsequent reactor protection system trip. Therefore, continued operation with offgas activity at above the offgas radiation monitor setpoint will normally cause an isolation of the main steam lines placing the plant outside the applicability of the specification. Therefore this change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   The proposed changed does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change allows the plant to cooldown to MODE 4 instead of either isolating the SJAE or all main steam lines. The offgas system will be operated in the same manner as during a normal cooldown. Although the offgas activity

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

2. (continued)

levels may be higher, this does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since placing the plant outside the applicable conditions of the LCO is an acceptable alternative to isolating the main steam lines or steam jet air ejectors. The potential of exceeding the offgas activity limits assumed in the analysis is minimized in MODE 4. In addition, the change also adds the requirement to be in MODE 3 in 12 hours. The consequences of an accident are significantly reduced when a cooldown is in progress. The additional time to exit the Applicability is acceptable since it avoids isolating the main condenser which is used as a heat sink during shutdown.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The Technical Specification Limit has been increased to 600,000 μCi/sec consistent with the value used in the Offgas System Failure accident of Section 11.4.7.2 (Table 11.4-1) of the UFSAR. This limit is not an initiator of any accident previously evaluated therefore this change will not increase the probability of an accident previously evaluated. Verification that the SJAE release is within the proposed Technical Specification limit will ensure a Offgas System Failure Accident will be bounded by the UFSAR analysis therefore this change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   The proposed changed does not introduce a new mode of offgas system operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change does not involve a significant reduction in a margin of safety since the current UFSAR assumptions have been considered in the proposed Technical Specification Limit. The calculated offsite doses resulting from an Offgas System failure event will be well within the limits of 10 CFR 100.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The determination of the gross radioactivity rate of noble gases has been postponed until after 31 days after any main steam line is not isolated and a SJAE is in operation. The offgas gross radioactivity rate surveillance is not considered to be an initiator of any accident. Therefore this change does not increase the probability of an accident previously evaluated. With all main steam lines isolated or if the SJAE is not in-service, this surveillance provides no meaningful information. The gross radioactivity rates are only expected to be high when operating close to full power conditions where the main steam lines are open and a SJAE is in operation. With any main steam line isolated or a SJAE not in service the reactor power is low and thus the resulting offgas gross radioactivity rate of noble gases are insignificant. During a reactor startup (when the main steam lines are opened and the SJAE placed in service) the gross radioactivity rate of noble gases should be considered to be nearly equivalent to the levels before shutting down. In addition, if offgas activity exceeded the setpoint of the offgas radiation monitors, alarms would annunciate and operators will be required to take action according to procedures. The system is designed to automatically isolate within 15 minutes if the activity remains above the setpoint. Therefore, entering the conditions of the Applicability without determining the gross radioactivity rate is acceptable and does not increase the consequences of an accident previously evaluated.

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

   The proposed changed does not introduce a new mode of offgas system operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

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

   The determination of the gross radioactivity rate of noble gases has been postponed until after 31 days after any main steam line is not isolated and a SJAE is in operation. Offgas gross radioactivity rate of noble gases are normally well below the limit when operating at full power conditions. With all main steam lines isolated or if the SJAE is not in-service reactor power must be far from rated conditions. In these conditions the gross radioactivity rate of noble gases is expected to be very low. In addition, if offgas activity exceeded the setpoint of the offgas radiation monitors, alarms would annunciate and operators will be required to take action according to procedures. The system is designed to automatically isolate within 15 minutes if the activity remains above the setpoints. Therefore, by postponing this surveillance until 31 days after entering the Applicability of the Specification is acceptable and does not significantly reduce the margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

3.7 PLANT SYSTEMS

Steam Jet Air Ejector (SJAE) PA1

3.7.6 Main Condenser Offgas

PA2 5

INSERT LCO

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~~.

600,000 M DBI DBI

[3.5.a (LCO) RETS] [Table 3.10-1 RETS]

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

DBI

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[Table 3.10-1 Footnote (e)] 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
[Table 3.10-1 Footnote (e)] B. Required Action and associated Completion Time not met.	~~B.1 Isolate all main steam lines.~~	~~12 hours~~ DBI
	~~OR~~	
	B.2 Isolate SJAE.	12 hours
	OR	
[L1]	B.3.1 Be in MODE 3.	12 hours
	AND	
	B.3.2 Be in MODE 4.	36 hours

D131

INSERT LCO

the discharge of the SJAE (prior to dilution and/or discharge), or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity)

PA1

SJAE

PA2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1 [L3] PA2 → 5	-----NOTE----- Not required to be performed until 31 days after any main steam line not isolated and SJAE in operation. -------------- DB1	
[3.5.1.a (SR) RETS] [M1]	Verify the gross gamma activity rate of the noble gases is ≤ ~~[240]~~ 600,000 μCi/second ~~[after decay of 30 minutes]~~. DB1	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 CLB1 With gross gamma activity rate > 5000 μCi/second

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.7.5 - MAIN CONDENSER STEAM JET AIR EJECTOR OFFGAS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 This threshold level was included to reduce the number of unnecessary grab samples and analyses. This allowance was approved in the NRC Safety Evaluation for Amendment 211 of the JAFNPP Operating License.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description or analysis description.

PA2 ISTS 3.7.6 has been renumbered as ITS 3.7.5 to reflect deletion of ISTS 3.7.3.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value/nomenclature has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

MARKUP OF NUREG-1433, REVISION 1, BASES

SJAE PA1

PA2

B 3.7 PLANT SYSTEMS

Steam Jet Air Ejector (SJAE) PA1

B 3.7.~~8~~ 5 Main Condenser Offgas

PA2

BASES

plant PA1 main

BACKGROUND

During ~~unit~~ plant operation, steam from the main 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.

PA3 PA2 plant SJAE PA2 and operates in three modes DB1

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.~~

INSERT BKGD-1 DB1 INSERT BKGD-2 at the discharge SJAE and in the main stack

SJAE PA1 DB2

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Section ~~[15.1.35]~~ 11.4.7.2 (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~~.

SJAE PA1 DB1 X1

The main condenser offgas limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 3).

PA1 SJAE

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 nominal noble gas release to the reactor coolant ~~of 100 µCi/MWt-second after decay of 30 minutes~~. The LCO is established consistent with

DB3

(continued)

Typical all pages

DB1

INSERT BKGD-1

During the startup mode, the SJAE offgas is directed to a 24 inch holdup pipe. During the intermediate mode the SJAE offgas is first directed to a recombiner and then to the same 24 inch holdup pipe. Finally in the normal mode of operation, the SJAE offgas is directed to the recombiner and then to charcoal beds. In all modes, before discharging to the main stack the offgas passes through a parallel set of HEPA filters.

DB1

INSERT BKGD-2

from the radiolytic dissociation of reactor coolant and other sources. After the recombiner, the offgas is cooled by two condensers in series and then delivered to one of two dryers to reduce the moisture content before being passed through the charcoal beds for delay and decay of noble gas activity.

a nominal production rate of 600,000 μCi/sec with no decay. DB3

BASES

LCO (continued)

~~this requirement [2436] MWt x 100 μCi/MWt-second = [240] mCi/second~~.

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 SJAE Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, main steam is not being exhausted to the main condenser and the requirements are not applicable. PA1 DB4 PA3

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 SJAE Offgas System rupture. PA1

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 SJAE Offgas System from the significant 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 primary containment isolation 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~~ plant systems. DB4 PA3 PA3 DB1 PA1

An alternative to Required Actions B.1 and B.2 is to place the ~~unit~~ plant in a MODE in which the LCO does not apply. To achieve this status, the ~~unit~~ plant must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The PA1

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. plant PA1

CLB1

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1 5 PA2

With the measured rate of radioactivity ≥ 5,000 μCi/second and

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. providing offgas isolation on excessive activity PA3

PA4 X2

, taken at the discharge (prior to dilution and/or discharge) of the SJAE, or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity)

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. DB4

PA1 (SJAE) PA1

REFERENCES

1. UFSAR, Section ~~15.1.35~~. 11.4.7.2 DB2

2. 10 CFR 100.

3. 10 CFR 50.36 (c)(2)(ii) X1

4. 10 CFR 50, Appendix I

CLB1

The 5,000 μCi/second threshold level is an administrative control to reduce the number of unnecessary grab samples. This value is approximately 1% of the SJAE trip level setting and operating at or below the threshold level will ensure the site boundary annual radiation exposures remain within the 10 CFR 50, Appendix I guidelines (Ref. 4)

CLB1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.5 - MAIN CONDENSER STEAM JET AIR EJECTOR OFFGAS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 This threshold level was included to reduce the number of unnecessary grab samples and analyses. This allowance was approved in the NRC Safety Evaluation for Amendment 211 of the JAFNPP Operating License.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA2 ISTS 3.7.6 has been renumbered as ITS 3.7.5 to reflect deletion of ISTS 3.7.3.

PA3 Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

PA4 Editorial changes made to be consistent with the Specification.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 Changes have been made to reflect the plant specific design.

DB2 The brackets have been removed and the proper plant specific references provided.

DB3 Changes have been made to reflect the plant specific analysis.

DB4 The brackets have been removed and the proper plant specific information has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X2 The list of noble gases actually sampled is more extensive, therefore these details are not included.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.5

Main Condenser Steam Jet Air Ejector (SJAE) Offgas

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.7 PLANT SYSTEMS

3.7.5 Main Condenser Steam Jet Air Ejector (SJAE) Offgas

LCO 3.7.5 The gross gamma activity rate of the noble gases measured at the discharge of the SJAE (prior to dilution and/or discharge), or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity) shall be $\leq$ 600,000 μCi/second.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	------------------NOTE------------------ Not required to be performed until 31 days after any main steam line not isolated and SJAE in operation. -- Verify the gross gamma activity rate of the noble gases is ≤ 600,000 μCi/second.	31 days AND With gross gamma activity rate > 5,000 μCi/second, 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

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Steam Jet Air Ejector (SJAE) Offgas

BASES

BACKGROUND

During plant operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser (SJAE) Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser SJAE Offgas System has been incorporated into the plant design to reduce the gaseous radwaste emission and operates in three modes. During the startup mode, the SJAE offgas is directed to a 24 inch holdup pipe. During the intermediate mode the SJAE offgas is first directed to a recombiner and then to the same 24 inch holdup pipe. Finally in the normal mode of operation, the SJAE offgas is directed to the recombiner and then to charcoal beds. In all modes, before discharging to the main stack the offgas passes through a parallel set of HEPA filters.

This system uses a catalytic recombiner to recombine hydrogen and oxygen from the radiolytic dissociation of reactor coolant and other sources. After the recombiner, the offgas is cooled by two condensers in series and then delivered to one of two dryers to reduce the moisture content before being passed through the charcoal beds for delay and decay of noble gas activity. The radioactivity of the gaseous mixture is monitored at the discharge of the recombiner and in the main stack.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser SJAE Offgas System failure event, discussed in the UFSAR, Section 11.4.7.2 (Ref. 1). The analysis assumes a gross failure in the Main Condenser SJAE Offgas System that results in the rupture of the Main Condenser SJAE 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).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).
LCO	To ensure compliance with the assumptions of the Main Condenser SJAE Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a nominal noble gas release to the reactor coolant. The LCO is established consistent with a nominal production rate of 600,000 μCi/sec with no decay.
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 SJAE Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, main steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

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

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

If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser SJAE Offgas System from significant sources of radioactive steam. The main steam lines are considered

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain primary containment isolation valve 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 plant systems.

An alternative to Required Actions B.1 and B.2 is to place the plant in a MODE in which the LCO does not apply. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample, taken at the discharge (prior to dilution and/or discharge) of the SJAE, or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity) to ensure that the required limits are satisfied. With the measured rate of radioactivity > 5,000 μCi/second and 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 providing offgas isolation on excessive activity, and is acceptable, based on operating experience. The 5,000 μCi/second threshold level is an administrative control to reduce the number of unnecessary grab samples. This value is 1% of the SJAE trip level setting and operating at or below the threshold level will ensure the site boundary annual radiation exposures remain within the 10 CFR 50, Appendix I guidelines (Ref. 4).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

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 SJAE Offgas System at significant rates.

REFERENCES

1. UFSAR, Section 11.4.7.2.
2. 10 CFR 100.
3. 10 CFR 50.36(c)(2)(ii).
4. 10 CFR 50, Appendix I.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

MI

Insert New Specification 3.7.6

Insert new Specification 3.7.6, "Main Turbine Bypass System", as shown in the JAFNPP Improved Technical Specifications.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A new Specification requiring the Main Turbine Bypass System to be Operable is proposed to be added. The proposed Specification will require the Main Turbine Bypass System to be Operable or LHGR and MCPR penalties to be applied. This proposed change is an additional restriction on plant operations since the CTS does not provide any restrictions with the Main Turbine Bypass System inoperable. This Specification will help ensure the safety analyses assumptions of certain events are maintained by limiting the resulting LHGR and MCPR if the event were to occur during power operation.

TSTF-319 E

TSTF-319

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.7.6 - MAIN TURBINE BYPASS SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

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

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

MARKUP OF NUREG-1433, REVISION 1 SPECIFICATION

PA1

E

TSTF-319

3.7 PLANT SYSTEMS

3.7.6 The Main Turbine Bypass System

PA1

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

OR

INSERT LCO-1

TA1

a. TA1

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

DB1 TA1 DB1

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

Insert LCO-2 TA1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met ~~or Main Turbine Bypass System inoperable.~~ DB1	A.1 Satisfy the requirements of the LCO ~~or restore Main Turbine Bypass System to OPERABLE status.~~ DB1	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify one complete cycle of each main turbine bypass valve.	~~31 days~~ 24 months X3 E

PA1

(continued)

TAC

INSERT LCO-1

The following limits are made applicable:

TAC

INSERT LCO-2

TSTF-379

b. LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.2	Perform a system functional test.	24 months X1
PA1 SR 3.7.6.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.7.6 - MAIN TURBINE BYPASS SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 ISTS 3.7.7 has been renumbered as ITS 3.7.6 to reflect deletion of ISTS 3.7.3. The Surveillances have been renumbered to reflect this change.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the information retained. Subsequent reload analyses may be performed without taking credit for the Main Turbine Bypass System. Therefore, the option to adjust the MCPR operating limit is retained.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 319, Revision 0, have been incorporated into the revised Improved Technical Specifications. However, in lieu of adding the APLHGR limits, which at JAFNPP is an accident limit, the LHGR limits, which may impact certain transients, is the proper limit and is being added. TSTF-319

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 The bracketed Surveillances Frequencies in ITS SRs 3.7.6.2 and 3.7.6.3 have been changed from 18 months to 24 months consistent with the length of the current operating cycle. The proposed Frequency is consistent with the bases justification for these surveillances.

X2 Not Used. 1 E

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X3 The Frequency of ITS SR 3.7.6.1 has been changed from 31 days to 24 months. Currently, this test is not required in the CTS and is only performed during refueling outages, as required by plant procedures. Monthly full stroke testing causes wear of both the bypass valves and the condenser intervals, leading to leaks and reduced efficiency. In addition, a review of historical maintenance and testing data for approximately the past 10 years has shown that this test normally passes its Surveillance at the refueling outage Frequency. The above data reviewed determined that there have been no failures of a bypass valve to cycle during this test.

E

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.7 PLANT SYSTEMS

B 3.7.9 Main Turbine Bypass System

PA1 6

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during ~~unit~~ plant (PA2) 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. (DB1) The Main Turbine Bypass System consists of ~~three~~ four (DB1) 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~~ four (DB1) valves is operated by ~~hydraulic cylinders~~ porting hydraulic fluid to the operating pistons through an electrically positioned servo valve (DB1). The bypass valves are controlled by the pressure regulation function of the Turbine Electro-Hydraulic Control (EHC) (PA2) System, as discussed in the ~~F~~SAR, (1) (PA2) Section ~~7.7.4~~ 11 (DB2) (Ref. 1). The bypass valves are normally closed, and the ~~pressure regulator~~ EHC (PA2) 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~~ manifold (DB1) ~~through~~ through each bypass valve and associated (DB1) connecting piping, to ~~the~~ pressure ~~breakdown assemblies~~ reducer (DB1), where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during ~~the turbine generator load rejection~~ some (DB3) abnormal operational (3) transients, as discussed in the ~~F~~SAR, (1) (PA2) Section ~~15.1.1~~ 14.5 (DB2) (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 ~~an~~ MCPR or LHGR penalties. (TA1) TSTF-318

With an inoperable Main Turbine Bypass System, the feedwater controller failure event may become the limiting event. (DA3) TSTF-319

The Main Turbine Bypass System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) (Ref. 3) (X1).

(continued)

PA1

and the LHGR limit (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)" TA1

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. ~~[~~With the Main Turbine Bypass System inoperable, modifications to the operating MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met.~~]~~ The MCPR limit~~s~~ for the inoperable Main Turbine Bypass System ~~is~~ are specified in the COLR, if applicable. An OPERABLE Main Turbine Bypass System requires ~~the~~ three of the four bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. ~~2~~ 4).

PA2 TA1 LHGR limit and PA2 DB3 PA2 DB3 E

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~~ applicable safety analyses. As discussed in the Bases for LCO 3.2.~~1~~, ~~"AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and~~ LCO 3.2.~~2~~ 3, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

DB3 TA1 2 E TSTF-319

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), ~~or~~ and the operating MCPR limit~~s~~ 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 LHGR limit and operating MCPR limit~~s~~ 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.

PA2 TA1 PA2 DB4 TSTF-319 E

(continued)

BASES

ACTIONS (continued)

PA2 TA1

LAGR limit and

operating PA2

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the ~~turbine generator load rejection~~ abnormal operational 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 plant systems.

DB3 5 PA2 TSTF-319

SURVEILLANCE REQUIREMENTS

SR 3.7.7 6 .1 PA1

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~~ X4 24 month 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.~~

E E

SR 3.7.7 6 .2 PA1

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required required system initiation signals, the valves will actuate to their required position. DB3 The (18) 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit plant 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) 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint. X2 PA2 X2

(continued)

BASES

6 PA1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.~~7~~.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in ~~[unit specific documentation]~~ the COLR DB2. The ~~[18]~~ 24 X2 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]~~ 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

PA2

1. U FSAR, Section [7.~~[7.4]~~ 11]. DB2
2. U FSAR, Section ~~[15.1.1]~~ 14.5. DB2

Insert REF DB3 X1

DB3

INSERT REF

3. 10 CFR 50.36(c)(2)(ii).

4. J11-03757SRL, Revision 0, Supplemental Reload Licensing Report for James A. Fitzpatrick Reload 14 Cycle 15, August 2000.

TSTF-319

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.6 - MAIN TURBINE BYPASS SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 ISTS 3.7.7 has been renumbered as ITS 3.7.6 to reflect deletion of ISTS 3.7.3. The Surveillances have been renumbered to reflect this change.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design of the Main Turbine Bypass System.

DB2 The brackets have been removed and the proper plant specific references provided.

DB3 Changes have been made to reflect the plant specific analysis. The References have been renumbered to reflect this change, as applicable.

DB4 The brackets have been removed and changes made to reflect the plant specific analyses.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 319, Revision 0, have been incorporated into the revised Improved Technical Specifications. However, some changes were made to be consistent with changes to the Specification. TSTF-319

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.7.6 - MAIN TURBINE BYPASS SYSTEM

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X2 The bracketed Surveillances Frequencies in ITS SRs 3.7.6.2 and 3.7.6.3 have been changed from 18 months to 24 months consistent with the length of the current operating cycle. The proposed Frequency is consistent with the bases justification for these surveillances.

X3 Not Used.

X4 The Bases have been changed to reflect a change made to the Specification.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.6

Main Turbine Bypass System

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.7 PLANT SYSTEMS

3.7.6 The Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

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

b. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify one complete cycle of each main turbine bypass valve.	24 months
SR 3.7.6.2	Perform a system functional test.	24 months
SR 3.7.6.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

B 3.7 PLANT SYSTEMS

B 3.7.6 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 plant 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 four valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve chest. Each of these four valves is operated by porting hydraulic fluid to the operating pistons through an electrically positioned servo valve. The bypass valves are controlled by the pressure regulation function of the Turbine Electro-Hydraulic Control (EHC) System, as discussed in the UFSAR, Section 7.11 (Ref. 1). The bypass valves are normally closed, and the EHC 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 EHC controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass manifold, through each bypass valve and associated connecting piping, to a pressure reducer, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during some transients, as discussed in the UFSAR, Section 14.5 (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 MCPR or LHGR penalties. With an inoperable Main Turbine Bypass System, the feedwater controller failure event may become the limiting event. TSTF-39

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 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 MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)") may be applied to allow this LCO to be met. The LHGR limit and MCPR operating limit for the inoperable Main Turbine Bypass System are specified in the COLR, if applicable. An OPERABLE Main Turbine Bypass System requires three of the four bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 4).
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 applicable safety analysis. As discussed in the Bases for LCO 3.2.2 and LCO 3.2.3, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.
ACTIONS	<u>A.1</u> If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), and the LHGR limit and MCPR operating limit 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 LHGR limit and MCPR operating limit

TSTF-319

TSTF-319 (E)

TSTF-319

TSTF-319

TSTF-319

(continued)

BASES

ACTIONS

A.1 (continued)

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. 1E

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the LHGR limit and MCPR operating limit 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 applicable safety analyses transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. TSTF-319

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

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 24 month 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 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. 1E 1E

SR 3.7.6.2

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 required valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and because of the potential for an

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.2 (continued)

unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

SR 3.7.6.3

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 the COLR. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant 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 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. USFAR, Section 7.11.
2. UFSAR, Section 14.5.
3. 10 CFR 50.36(c)(2)(ii).
4. J11-03757SRL, Revision 0, Supplemental Reload Licensing Report for James A. FitzPatrick Reload 14 Cycle 15, August 2000.

TSTF-319

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.7.7 (A1)

3.10 (cont'd)

[Applicability] (L1)

[3.7.7] C. Spent Fuel Storage Pool Water Level

[LCO 3.7.7] Whenever irradiated fuel is stored in the spent fuel storage pool, the pool water level shall be maintained at a minimum level of ~~33 ft.~~ (M1) (LA1)

D. Control Rod and Control Rod Drive Maintenance

1. Two control rods may be withdrawn from the reactor core to perform maintenance provided:

   a. The Reactor Mode Switch is locked in the Refuel position and all refueling interlocks are operable except for those necessary to perform the demonstration and maintenance described in Specification 4.10.D.1.

   b. Control rods immediately face and diagonally adjacent to the control rods to be withdrawn are fully inserted, electrically disarmed and sufficient margin to criticality demonstrated.

   c. Control rods to be withdrawn are separated by three or more cells in any direction. (This specification does not apply to the control rods used to perform the demonstration required by Specification 3.10.D.1.b.)

See ITS: 3.10.5

4.10 (cont'd)

[3.7.7] C. Spent Fuel Storage Pool Water Level (L1)

[SR 3.7.7.1] Whenever irradiated fuel is stored in the spent fuel storage pool, the pool water level shall be recorded ~~daily~~. every 7 days (L2)

D. Control Rod and Control Rod Drive Maintenance

1. When two control rods are withdrawn from the reactor core for maintenance, the following surveillance shall be performed:

   a. If the reactor vessel head is removed, specification 4.10.A.1 shall be satisfied.

   b. Demonstrate that the reactor core can be maintained subcritical with a margin of 0.38 percent Δ k at any time during the maintenance with the analytically determined strongest worth operable control rod fully withdrawn. This margin shall be demonstrated after Specification 3.10.D.1 has been satisfied.

See ITS: 3.10.5

add proposed Action A (M2)

add ACTION A Note (L3)

Page 1 of 1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.10.C requires the spent fuel storage pool level be maintained at a minimum level of 33 ft (equivalent to 17 feet above the top of irradiated fuel assemblies seated in the spent fuel storage racks). ITS 3.7.7 requires this level to be maintained at 21 ft 7 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool. This will ensure that the consequences of a refueling accident over the spent fuel pool storage pool will remain within the bounds of the refueling accident described in UFSAR, Section 14.6.1.4. Since the limit has been raised 4 ft 7 inches, this change is considered more restrictive but safer for plant operation since an additional 4 ft of water will provide for absorption of water soluble fission product gases and transport delays of soluble fission product gases.

M2 CTS 3.10.C does not provide any specific actions when the LCO is not met. Therefore when the LCO is not met CTS 3.0.C must be entered and the reactor placed in COLD SHUTDOWN within 24 hours. This shutdown requirement has been deleted as discussed in Discussion of Change L3 for this Specification and a more appropriate ACTION is provided. ITS 3.7.7 ACTION A will immediately require the suspension of movement of irradiated fuel in the spent fuel pool; thus, an accident cannot occur. This change is consistent with NUREG-1433, Revision 1, and represents an additional restriction on plant operation but necessary to ensure operation is maintained within the bounds of the safety analysis.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The requirement in CTS 3.10.C that whenever irradiated fuel is stored in the spent fuel storage pool, the pool water level shall be maintained at a minimum water level of 33 ft is proposed to be relocated to the UFSAR. The current requirement of 33 ft is to ensure both adequate cooling and shielding requirements of the fuel is met. The current limit is not provided to satisfy the requirements of any design basis events. The

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

proposed limit in ITS 3.7.7 will ensure the consequences of a refueling accident over the spent fuel storage pool will be bounded by the analysis in UFSAR, Section 14.6.1.4. As such, the current spent fuel pool level in CTS 3.10.C is not required to be in ITS to provide adequate protection of public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.10.C and CTS 4.10.C require the spent fuel storage pool level to be greater than or equal to the minimum level whenever irradiated fuel is stored in the spent fuel storage pool. The ITS 3.7.7 Applicability is during movement of irradiated fuel assemblies in the spent fuel storage pool. The requirement for a certain level in the spent fuel storage pool is only required when moving irradiated fuel. The current Technical Specifications specify that the Specification is applicable whenever irradiated fuel is stored in the pool. The refueling accident assumes a minimum water level above the irradiated fuel assemblies and that an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies. This proposed change, while relaxing the current Applicability, maintains the assumptions of the bounding design basis refueling accident described in UFSAR, Section 12.2A.6 Event 10 and UFSAR, Section 14.6.1.4. At other times, the spent fuel storage pool level will be maintained in accordance with the current requirements in CTS 3.10.C. This requirement has been relocated to the UFSAR as discussed in LA1. In addition, the design of the spent fuel storage pool prevents the water level from uncovering the fuel during an inadvertent drain down as long as the fuel is seated in the pool. Since the current requirement in CTS 3.10.C does not satisfy the requirements of 10 CFR 50.36(c)(2)(ii), this change is considered acceptable.

L2 CTS 4.10.C requires the spent fuel storage pool level to be recorded daily. ITS SR 3.7.7.1 requires the verification of the spent fuel storage level every 7 days. This change relaxes the Surveillance Requirement frequency in CTS 4.10.C to verify spent fuel storage pool water level from daily to once every 7 days. The 7 day Frequency is acceptable, based on operating experience considering that the water volume in the spent fuel storage pool is typically required (per plant procedures) to be maintained constant and any significant decrease in level below the normal level results in activation of an alarm in the Control Room well in advance of the minimum level of 21 ft 7 inches above the top irradiated fuel assemblies seated in the spent fuel

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

storage pool racks.

The SFP is maintained at a normal operating level of elev. 368' 6" (+3,-2 in.), equivalent to 22' 2" (+3, -2 in.) above fuel stored in the pool. A SFP low-level alarm is received at elev. 368' 2", requiring entry into an Abnormal Operating Procedure, which requires restoration of SFP level to the normal control band. Further lowering of level would result in additional alarms from the SFP Cooling System, whose suction invert is at elev. 368' 1". With the refueling gates in place, lowering SFP level below that specified in ITS 4.3.2 (elev. 367' 3") would require the installation and use of temporary pumps. Thus, the SFP would not be maintained at a level approaching the CTS limit, except during maintenance activities with attendant special administrative controls. RAI 3.7.7-1

L3 CTS 3.10.C does not provide any specific actions when the LCO is not met. Therefore, when the LCO is not met CTS 3.0.C must be entered and the reactor placed in COLD SHUTDOWN within 24 hours. The requirement to shutdown is not maintained in ITS 3.7.7. A Note has been added to ITS 3.7.7 Required Action A.1 (the requirement to suspend movement of irradiated fuel assemblies if the spent fuel storage pool water level is not within limits). The Note states that LCO 3.0.3 is not applicable. LCO 3.0.3 requires the reactor to be brought to a non-applicable MODE if the Required Actions cannot be met or no Required Actions exist for a particular Condition. Moving irradiated fuel assemblies while in MODE 1, 2, or 3 is independent of reactor operations and a reactor shutdown would not improve the situation when the water level in the spent fuel storage pool is not within limits. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown. This Note is also necessary because defaulting to ITS LCO 3.0.3 would require the reactor to be shutdown but would not require suspension of movement of irradiated fuel which would preclude a refueling accident from occurring over the spent fuel pool. Therefore, not allowing ITS 3.7.7 Required Action A.1 to be bypassed by entering LCO 3.0.3 requires the plant to be placed in a condition of minimum risk by ensuring actions are taken to preclude a refueling accident from occurring over the spent fuel pool.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change alters the Applicability to require the LCO to be applicable only during movement of irradiated fuel assemblies in the spent fuel storage pool which will ensure the assumptions of the refueling accident is met. The proposed change does not affect the probability of an accident. The spent fuel pool water level is not assumed to be an initiator of any analyzed event. The consequences of an accident are not affected by changing the Applicability to only when moving fuel assemblies in the spent fuel storage pool. The refueling accident assumes a minimum water level above the irradiated fuel assemblies and that an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies. This proposed change, while relaxing the current Applicability, maintains the assumptions of the bounding design basis refueling accident. At other times, the spent fuel storage pool level will be maintained in accordance with the current requirements. In addition, the design of the spent fuel pool storage pool prevents the water level from uncovering the fuel during an inadvertent fuel draindown as long as the fuel is seated in the pool. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change alters the Applicability to require the LCO to be applicable only during movement of irradiated fuel assemblies in the spent fuel storage pool. The proposed changes to the Applicability will not create the possibility of an accident. This Applicability is being changed to reflect the assumptions of the safety analysis. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

2. (continued)

   assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change alters the Applicability to require the LCO to be applicable only during movement of irradiated fuel assemblies in the spent fuel storage pool. The margin of safety is not significantly reduced because the refueling accident assumes a minimum water level above the irradiated fuel assemblies and that an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies. This proposed change, while relaxing the current Applicability, maintains the assumptions of the bounding design basis refueling accident. The safety analysis assumptions will still be maintained, thus no question of safety exist. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change proposes to relax the Surveillance Requirement Frequency to verify spent fuel storage pool water level from daily to once every 7 days. The proposed change does not affect the probability of an accident. The spent fuel pool water level is not assumed to be an initiator of any analyzed event. The proposed change still provides assurance spent fuel pool water level is maintained consistent with analysis assumptions. Low water level and flow alarms are available to alert the operator to ensure water loss from the spent fuel pool does not go undetected between surveillances. The consequences of an accident are not affected by decreasing the frequency of the Surveillance to verify the spent fuel storage pool water level since the proposed Frequency is considered acceptable, based on operating experience, considering that the volume in the pool is normally stable and all reactor level changes are controlled by plant procedures. Additionally, the most common outcome of the performance of a surveillance is the successful demonstration that the acceptance criteria are satisfied. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   This change relaxes the Surveillance Requirement frequency to verify spent fuel storage pool water level from daily to once every 7 days. The proposed changes to the Frequency will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

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

This change relaxes the Surveillance Requirement frequency to verify spent fuel storage pool water level from daily to once every 7 days. The increased interval for the verification of water level is acceptable since the 7 day frequency has been shown, based on industry operating experience, to be adequate for maintaining the spent fuel storage pool water level within limits. In addition, low water level and flow alarms are available to alert the operator to ensure water loss from the spent fuel pool does not go undetected between surveillances. Therefore, the margin of safety is not significantly reduced because the proposed changes to the Surveillance Frequency will continue to provide the necessary assurance that spent fuel storage pool water level is being maintained within limits. Also, this change is considered acceptable since the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change provides a Note to the requirement to suspend movement of irradiated fuel assemblies if the spent fuel storage pool water level is not within limits. The Note states that ITS LCO 3.0.3 is not applicable. ITS LCO 3.0.3 requires the reactor to be brought to a non-applicable mode if the Required Actions cannot be met or no actions exist for a particular condition. The proposed change does not affect the probability of an accident. The spent fuel pool water level is not assumed to be an initiator of any analyzed event. The consequences of an accident are not affected by adding a note exempting the actions from ITS LCO 3.0.3. Irradiated fuel movement within the spent fuel pool is independent of reactor operation. It does not affect any systems required for safe operation or safe shutdown of the reactor. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change provides a Note to the requirement to suspend movement of irradiated fuel assemblies if the spent fuel storage pool water level is not within limits. The Note states that ITS LCO 3.0.3 is

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

3. (continued)

not applicable. ITS LCO 3.0.3 requires the reactor to be brought to a non-applicable mode if the Required Actions cannot be met or no actions exist for a particular condition. The margin of safety is not reduced because the movement of irradiated fuel within the spent fuel storage pool is independent of reactor operations and a reactor shutdown would not improve the situation where the water level in the spent fuel storage pool is not within limits. The safety analysis assumptions will still be maintained, thus no question of safety exist. Therefore, this change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

MARKUP OF NUREG-1433, REVISION 1 SPECIFICATION

3.7 PLANT SYSTEMS

PA1 3.7.8 7 Spent Fuel Storage Pool Water Level

DB1 21 7 inches

[3.16.C] LCO 3.7.8 7 — The spent fuel storage pool water level shall be ≥ [23] 21 ft 7 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

[3.10.C] APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
[L3] [M2] 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 7 PA1	Verify the spent fuel storage pool water level is ≥ [23] 21 ft 7 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days DB1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.7.7 - SPENT FUEL STORAGE POOL WATER LEVEL

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 ISTS 3.7.8 has been renumbered as ITS 3.7.7 to reflect deletion of ISTS 3.7.3. The Surveillance has also been renumbered as a result of this change.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level 7 PA1

BASES

BACKGROUND

PA3 ensures that are met

The minimum water level in the spent fuel storage pool ~~meets~~ the assumptions of iodine decontamination factors following a ~~fuel handling~~ accident. refueling PA2 PA2

A general description of the spent fuel storage pool design is found in the FSAR, Section ~~9.1~~ (Ref. 1). The assumptions of the ~~fuel handling~~ accident are found in the FSAR, Section ~~15.1.A~~ (Ref. 2). 9.3 DB1 U DB1 14.6.1.4 refueling PA2 PA2

APPLICABLE SAFETY ANALYSES

refueling refueling implicit PA3

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). refueling PA2

refueling PA2

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 ~~9.1.2/22.2~~ (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. refueling refueling PA2 U PA2 14.6.1.1 DB1 refueling PA2

The spent fuel storage pool water level satisfies Criterion 2 and 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) (Ref. 7). X1 TA1

(continued)

BASES (continued)

LCO

PA2 refueling

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.

ACTIONS

A.1

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

PA3

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 7 PA1

PA2 refueling

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.

plant PA2

(continued)

BASES (continued)

PA2 9.3 DB1 DB1

REFERENCES

1. ~~FSAR~~ UFSAR, Section ~~[7.4]~~ 9.3.
2. ~~FSAR~~ UFSAR, Section ~~15.2.4~~ 14.6.1.4.
3. ~~2.~~ ~~NUREG-0800,~~ Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 15.7.4, Radiological Consequences of Fuel Handling Accident, Revision 1, July 1981.
4. ~~3.~~ 10 CFR 100.
5. Regulatory Guide 1.25, Assumptions Used for Evaluating The Potential Radiological Consequences Of A Fuel Handling Accident In The Fuel Handling And Storage Facility For Boiling And Pressurized Water Reactors, March 1972.
6. ~~FSAR~~ UFSAR, Section ~~9.1.2.2~~ 14.6.1.1.
7. 10 CFR 50.56 (c)(2)(ii).

PA3 PA4 PA3 DB1 X1 PA3

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 ISTS 3.7.8 has been renumbered as ITS 3.7.7 to reflect deletion of ISTS 3.7.3. The surveillance has been renumbered as a result of this change.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

PA3 Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

PA4 The references have been renumbered to reflect the sequence in the Bases description.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific references have been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 139, Revision 1, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference the "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.7.7

Spent Fuel Storage Pool Water Level

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.7 PLANT SYSTEMS

3.7.7 Spent Fuel Storage Pool Water Level

LCO 3.7.7 The spent fuel storage pool water level shall be ≥ 21 ft 7 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify the spent fuel storage pool water level is ≥ 21 ft 7 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days

B 3.7 PLANT SYSTEMS

B 3.7.7 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 refueling accident. A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.3 (Ref. 1). The assumptions of the refueling accident are found in the UFSAR, Section 14.6.1.4 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The water level above the irradiated fuel assemblies is an implicit assumption of the refueling accident. A refueling 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 refueling 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 refueling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a refueling accident over the spent fuel storage pool are no more severe than those of the refueling accident over the reactor core, as discussed in the UFSAR, Section 14.6.1.1 (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 refueling accident. The spent fuel storage pool water level satisfies Criterion 2 and 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 7).

(continued)

BASES (continued)

LCO	The specified water level preserves the assumptions of the refueling 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.
ACTIONS	<u>A.1</u> LCO 3.0.3 is not applicable while in MODE 4 and 5. However, because irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving 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	<u>SR 3.7.7.1</u> This SR verifies that sufficient water is available in the event of a refueling 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 plant procedures.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.3.
2. UFSAR, Section 14.6.1.4.
3. 10 CFR 100.
4. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 15.7.4, Revision 1, Radiological Consequences of Fuel Handling Accident, July 1981.
5. Regulatory Guide 1.25, Assumptions Used for Evaluating The Potential Radiological Consequences Of A Fuel Handling Accident In The Fuel Handling And Storage Facility For Boiling And Pressurized Water Reactors, March 1972.
6. UFSAR, Section 14.6.1.1.
7. 10 CFR 50.36(c)(2)(ii).

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.7.3

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

THIS SPECIFICATION IS DELETED.

THERE ARE NO REQUIREMENTS FOR THIS SPECIFICATION AT JAFNPP; THEREFORE THIS MARKUP PACKAGE CONTAINS ONLY THE FOLLOWING SECTIONS:

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.7.3

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

MARKUP OF NUREG-1433, REVISION 1 SPECIFICATION

DB1

3.7 PLANT SYSTEMS

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

LCO 3.7.3 The DG [1B] SSW System shall be OPERABLE.

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

ACTIONS

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

DB1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify each DG [1B] SSW System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.3.2	Verify the DG [1B] SSW System pump starts automatically when DG [1B] starts and energizes the respective bus.	[18] months

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.7.3

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

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 ISTS 3.7.3, "Diesel Generator (DG)[1B] Standby Service Water (SSW) System is not being included in the JAFNPP ITS since the Emergency Service Water System provides the cooling water requirements to the Emergency Diesels Generators (EDGs). The requirements of the Emergency Service Water System are included in ITS 3.7.2, "Emergency Service Water (ESW) and Ultimate Heat Sink (UHS)" therefore this Specification is not required to be included in the ITS.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.7.3

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

MARKUP OF NUREG-1433, REVISION 1, BASES

DB1

B 3.7 PLANT SYSTEMS

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

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. 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).
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]. The DG [1B] SSW System satisfies Criterion 3 of the NRC Policy Statement.
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. 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)]").

(continued)

DB1

BASES (continued)

APPLICABILITY

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

ACTIONS

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

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

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

B.1

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. 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. Additionally, if the DG [1B] SSW System is

(continued)

(DB1)

BASES

ACTIONS

B.1 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

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

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

SR 3.7.3.2

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.

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.

(continued)

BASES (continued)

REFERENCES

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

DB1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.7.3

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

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
NUREG BASES: 3.7.3 - DIESEL GENERATOR (DG) [1B] STANDBY SERVICE WATER (SSW) SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 ISTS 3.7.3, "Diesel Generator (DG)[1B] Standby Service Water (SSW) System is not being included in the JAFNPP ITS since the Emergency Service Water System provides the cooling water requirements to the Emergency Diesels Generators (EDGs). The requirements of the Emergency Service Water System are included in ITS 3.7.2, "Emergency Service Water (ESW) and Ultimate Heat Sink (UHS)" therefore this Specification is not required to be included in the ITS.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.8

Miscellaneous Radioactive Materials Sources

THIS SPECIFICATION IS Relocated.

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTICTIVE CHANGES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.8

Miscellaneous Radioactive Materials Sources

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

CTS 3/4.8 1

3.8 LIMITING CONDITIONS FOR OPERATION

3.8 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability:

Applies to the handling and use of sealed special nuclear, source and by-product material at all times.

Objective:

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

Specification:

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

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

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with applicable regulations.

R1

4.8 SURVEILLANCE REQUIREMENTS

4.8 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability:

Applies to the surveillance requirements of sealed special nuclear, source and by-product materials.

Objective:

To specify the surveillances to be applied to sealed special nuclear, source and by-product materials.

Specification:

Tests for leakage and/or contamination shall be conducted as follows:

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

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

C. Startup sources shall be leak tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

D. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample. Testing shall be performed by the licensee or by other persons specifically authorized by the NRC or an agreement State.

Page 1 of 1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.8

Miscellaneous Radioactive Materials Sources

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
CTS: 3/4.8 - MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

R1 The limitations on miscellaneous radioactive materials sources are intended to ensure that the whole body or individual organ irradiation doses does not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of removable contamination on each sealed source. This requirement and the associated Surveillance Requirements bear no relation to the conditions or limitations which are necesary to ensure safe reactor operation. Therefore, the requirements specified in CTS 3/4.8 did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the JAFNPP Technical Specifications and are proposed to be relocated to the Technical Requirements Manual (TRM). At ITS implementation, the TRM will be incorporated by reference into the UFSAR. Changes to the relocated requirements in the TRM will be controlled by the provisions of 10 CFR 50.59.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.8

Miscellaneous Radioactive Materials Sources

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3/4.8 - MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

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

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.11.C

Battery Room Ventilation

THIS SPECIFICATION IS Relocated.

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTICTIVE CHANGES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.11.C

Battery Room Ventilation

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

CTS 3/4.11.C

3.11 (cont'd)

B. DELETED

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

4.11 (cont'd)

B. DELETED

C. Battery Room Ventilation

Battery room ventilation equipment shall be demonstrated operable once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be verified operable and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

LA1

Page 1 of 1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.11.C

Battery Room Ventilation

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The requirements for the Battery Room Ventilation System in CTS 3/4.11.C are proposed to be relocated to the Technical Requirements Manual (TRM). The system was designed to control the temperature rise and hydrogen buildup in the battery room compartments during normal and accident condition to support the OPERABILITY of the 125 VDC batteries and chargers. However, it has been determined the system is no longer required to ensure the removal of hydrogen generated by the station batteries, since the plant is currently using lead-calcium cells which do not generate significant amounts of hydrogen. ITS 3.8.4 and 3.8.5 specifies the requirements for the 125 VDC batteries and chargers during Operating and Shutdown conditions, respectively. In addition, the requirements for battery cell parameters are included in ITS 3.8.6. ITS SR 3.8.6.3 requires the average electrolyte temperature of representative cells for the 125 VDC batteries to be ≥ 60°F. If this is not met the associated battery must be declared inoperable immediately. The definition of OPERABLE-OPERABILITY requires all necessary attendant instrumentation, controls, normal or emergency electrical power sources, cooling and seal water, lubrication and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). Therefore, since requirements for the 125 VDC batteries and chargers are retained in the Improved Technical Specifications, the Operability of the Battery Room Ventilation System will be required by the definition of OPERABLE-OPERABILITY to support the 125 VDC batteries and chargers. Therefore, the operability requirements contained in CTS 3/4.11.C are not required to be included in the ITS to provide adequate protection of the public health and safety. At ITS implementation, the relocated requirements will be incorporated by reference into the UFSAR. As such, changes to the relocated requirements in the Technical Requirements Manual will be controlled by the provisions of 10 CFR 50.59.

RAI CTS 3/4.11.C-1

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: **3/4.11.C**

Battery Room Ventilation

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3/4.11.C - BATTERY ROOM VENTILATION

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

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