ENCLOSURE 2

VOLUME 12

TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.7 PLANT SYSTEMS

Revision 0

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 3.7.1, MAIN STEAM SAFETY VALVES (MSSVs)

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

3/4.7 PLANT SYSTEMS

3/4.7.1 ~~TURBINE CYCLE~~ A01

SAFETY VALVES

Main Steam MSSVs

LIMITING CONDITION FOR OPERATION

Four MSSVs per A02

LCO 3.7.1
SR 3.7.1.1 3.7.1.1 ~~All main steam line Code safety valves associated with each~~ steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

Applicability APPLICABILITY: MODES 1, 2, and 3.

Add proposed ACTIONS Note A03

ACTION:

Add proposed Required Action B.2 Note L01

With (3) reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, and

Add proposed Required Action B.1 A04

ACTION B a. in MODES 1 and 2, with a positive Moderator Temperature Coefficient, operation may continue provided that, within 4 hours, either the inoperable valve(s) ~~are restored to OPERABLE status or~~ the Power Range Neutron Flux High Trip Setpoint is reduced to the maximum allowable percent of RATED THERMAL POWER listed In Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours, or A05 L02

ACTION C within 36 hours

ACTION A b. in MODES 1 and 2, with a negative or zero Moderator Temperature Coefficient; or in Mode 3, with a positive, negative or zero Moderator Temperature Coefficient, operation may continue provided that, within 4 hours, either the inoperable valve(s~~) are restored to OPERABLE status~~ or reactor power is reduced to less than or equal to the maximum allowable percent of RATED THERMAL POWER listed in Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. A05

ACTION C

SURVEILLANCE REQUIREMENTS

Add proposed SR Note L03

SR 3.7.1.1 4.7.1.1 No additional requirements other than those required by the INSERVICE TESTING PROGRAM. ~~The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.~~ A06

TABLE 3.7-1

~~MAXIMUM ALLOWABLE POWER LEVEL WITH~~
~~IN~~OPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION

MSSVs ~~MAXIMUM~~ NUMBER OF ~~IN~~OPERABLE ~~SAFETY VALVES ON ANY~~ PER ~~OPERATING~~ STEAM GENERATOR	MAXIMUM ALLOWABLE POWER LEVEL (PERCENT OF RATED THERMAL POWER)
3 ~~1~~	44
2	27
1 ~~3~~	10

A01

TABLE 3.7-2

Main Steam STEAM ~~LINE~~ SAFETY VALVE~~S~~ ~~PER LOOP~~ Lift Settings

A01 LA01 LA02

VALVE NUMBER	Loop A	Loop B	Loop C	LIFT SETTING (±3%)~~*~~ **	~~ORIFICE SIZE SQUARE INCHES~~
1.	RV1400	RV1405	RV1410	1085 psig	~~16~~
2.	RV1401	RV1406	RV1411	1100 psig	~~16~~
3.	RV1402	RV1407	RV1412	1105 psig	~~16~~
4.	RV1403	RV1408	RV1413	1105 psig	~~16~~

SR 3.7.1.1 Note *~~The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure~~. LA01

SR 3.7.1.1 **All valves tested must have “as left” lift setpoints that are within ±1% of the lift setting value listed in Table 3.7-2.

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

A02 CTS 3.7.1.1 states that all main steam safety valves (MSSVs) shall be OPERABLE with lift settings as specified in Table 3.7-2. CTS Table 3.7-2 lists lift setting pressures for four MSSVs in each of the three loops. ITS Limiting Condition for Operation (LCO) 3.7.1 requires four MSSVs per steam generator (SG) to be OPERABLE. This changes the CTS by combining the current LCO requirement and portions of CTS Table 3.7-2 into a single ITS LCO requirement.

This change is acceptable because the number of MSSVs required OPERABLE under the various conditions has not changed. This change results in a format change only to follow the manner in which the ISTS presents the MSSV requirements. This change is designated as an administrative change because it does not result is any technical changes to the CTS.

A03 CTS 3.7.1.1 ACTIONS a and b provide compensatory actions for one or more inoperable MSSVs. CTS 3.7.1.1 ACTION a requires that within 4 hours the MSSV(s) be restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip be reduced in accordance with the requirements of CTS Table 3.7-1. CTS 3.7.1.1 ACTION b requires that within 4 hours the MSSV(s) be restored to OPERABLE status or reactor power is reduced to less than or equal to the maximum allowable percent of RATED THERMAL POWER (RTP) listed in Table 3.7-1. ITS 3.7.1 ACTIONS Note states "Separate Condition entry is allowed for each MSSV." This changes the CTS by explicitly specifying separate condition entry for each inoperable MSSV.

The purpose of the CTS ACTIONS is to allow separate condition entry for each inoperable MSSV. Each time it is discovered that an MSSV is inoperable, entry is required and the specified Completion Time is applied to complete the compensatory actions. The ITS 3.7.1 ACTIONS Note allows a separate Completion Time period for each MSSV that is inoperable. This change is acceptable, because it only provides clarification of the Completion Time when one valve is inoperable and, subsequently, a second valve becomes inoperable. This change is designated as administrative, because it does not result in a technical change to the Specifications.

A04 CTS 3.7.1.1 ACTION a states that the Power Range Neutron Flux - High Setpoint trip must be reduced per CTS Table 3.7-1 when one or more MSSVs are found to be inoperable. CTS Table 3.7-1 provides the maximum allowable Power Range Neutron Flux - High Setpoint corresponding to the maximum number of inoperable MSSVs on any operating SG. ITS 3.7.1 ACTION B requires a reduction in THERMAL POWER and a reduction in the Power Range Neutron Flux - High reactor trip setpoint consistent with the requirements of ITS Table 3.7.1-1. The Table has been revised slightly to provide the associated maximum allowable power for the number of OPERABLE MSSVs. This changes the CTS by adding an additional explicit statement to reduce THERMAL POWER consistent with ITS Table 3.7.1-1 and by stating the maximum allowable power as a function of OPERABLE, instead of inoperable, MSSVs.

The purpose of CTS 3.7.1.1 ACTION a is to reduce the Power Range Neutron Flux – High Setpoint to within the limits of the safety analyses. Current plant operation dictates that THERMAL POWER is reduced before reducing the setpoints to prevent a reactor trip. Explicitly stating this practice in ITS and stating the maximum power level in terms of OPERABLE instead of inoperable MSSVs does not change how the plant is operated. This change is considered administrative, because it does not result in technical changes to the CTS.

A05 CTS 3.7.1.1 ACTION a states that with one or more MSSVs inoperable, either restore the inoperable valves to OPERABLE status or reduce the Power Range Neutron Flux – High Setpoints. ITS 3.7.1 ACTION A does not include the restoration requirement, only the alternate compensatory measure. This changes the CTS by eliminating the explicit statement to restore the MSSV(s) to OPERABLE status.

This change is acceptable, because it does not result in a technical change to the Technical Specifications. Restoration of compliance with the LCO is always an option in an ACTION, so eliminating the restoration ACTION from the CTS has no effect. In both the CTS and the ITS, if the inoperable MSSV(s) are not restored, actions are taken that result in reducing reactor power to within the relief capability of the OPERABLE MSSVs within 4 hours. This change is designated as administrative, because it does not result in a technical change to the CTS.

A06 CTS 4.7.1.1 requires verification of each required MSSV lift setpoint, and states "The provisions of Specification 4.0.4 are not applicable for entry into MODE 3." ITS Surveillance Requirement (SR) 3.7.1.1 does not contain this statement. However, ITS SR 3.7.1.1 does contain a Note that states, "Only required to be performed in MODES 1 and 2." This changes the CTS by not adding the CTS 4.0.4 exception.

The CTS 4.0.4 exception allows entry into MODE 3 to perform CTS Surveillance 4.7.1.1. This exception is not required in ITS SR 3.7.1.1 because the SR Note in the ITS only requires the SR to be performed in MODES 1 and 2. This change is designated as administrative as it results in no technical change to the CTS.

<u>MORE RESTRICTIVE CHANGES</u>

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 3.7-2 is modified by a footnote (footnote*) that states, "The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure." ITS 3.7.1 does not contain this information. This changes the CTS by moving details on setting the lift pressure to the ITS Bases.

The removal of these details for performing SRs from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the lift settings and the definition of OPERABLE states that the components must be capable of performing the specified safety function. It is understood that the MSSVs must be adjusted to lift at the settings given under the conditions that the safety analysis assumes the MSSVs will operate. This change is acceptable, because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because details for meeting Technical Specification requirements are being removed from the Technical Specifications to the ITS Bases.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.7-2 specifies the MSSV number and associated lift settings and nozzle size for each MSSV. ITS Table 3.7.1-2 only provides the MSSV number and associated lift setting. This changes the CTS by deleting the required nozzle size and relocating this detail to the Updated Final Safety Analysis Report (UFSAR).

The removal of details related to system design from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the valve numbers and corresponding lift setting. The nozzle size does not normally vary, because it is a function of the design of the valve. The lift settings can vary and are adjustable. Therefore, this information is important to be retained in the Technical Specification. Also, this change is acceptable, because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change, because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (*Category 4 – Relaxation of Required Action)* CTS 3.7.1.1 ACTION a states, in part, that with one or more MSSVs inoperable, reduce the Power Range Neutron Flux - High Setpoint trip within 4 hours. ITS 3.7.1 Required Action B.2 also requires the Power Range Neutron Flux - High trip setpoint to be reduced but is modified by a Note stating that this action is only required in MODE 1. This changes the CTS by only requiring the Power Range Neutron Flux - High Setpoint trip be reduced when in MODE 1.

The purpose of CTS 3.7.1.1 is to ensure that the MSSVs are capable of relieving Main Steam System pressure. In MODES 2 and 3, the Reactor Trip System trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a Design Basis Accident (DBA) occurring during the repair period. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 (*Category 3 – Relaxation of Completion Time)* CTS 3.7.1.1 ACTION a specifies the compensatory actions when one or more MSSVs are inoperable. The action allows operation to continue provided that within 4 hours, either the inoperable MSSV(s) are restored to OPERABLE status or the Power Range Neutron Flux - High Setpoint trip is reduced per Table 3.7-1. ITS 3.7.1 Required Action B.2 requires the reduction of the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable percent RTP specified in Table 3.7.1-1 within 36 hours. This changes the CTS by extending the time allowed to reduce the Power Range Neutron Flux - High reactor trip setpoints. The discussion of the change that deletes the restoration option is discussed in DOC A05.

The purpose of CTS 3.7.1.1 ACTION a is to limit the time the unit can operate with inoperable MSSVs without reducing the Power Range Neutron Flux – High reactor trip setpoints. This change is acceptable, because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs, and the low probability of a DBA occurring during the allowed Completion Time. This change extends the time allowed to reduce the Power Range Neutron Flux - High reactor trip setpoints when the MSSVs are inoperable. The time extension is from 4 hours to 36 hours. However, the time to reduce THERMAL POWER to within the same limits (four hours) is maintained in ITS 3.7.1 as Required Action B.1, as described in DOC A04. This change is acceptable because the Completion Time of 36 hours is a reasonable time to reset the channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. In addition, the actual reactor power level continues to be required to be reduced to within the same limits within 4 hours. Thus, operation of the unit at a RTP above limits with one or more inoperable MSSV(s) is still only allowed for 4 hours, consistent with the current allowance. This change is designated as less restrictive, because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L03 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.7.1.1 requires verification of each MSSV lift setpoint pursuant to the INSERVICE TESTING PROGRAM. CTS Table 3.7-2 Footnote * states that the lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. ITS SR 3.7.1.1 requires the same testing; however, a Note has been included that requires the performance of the lift setpoint verification only in MODES 1 and 2, which corresponds to the valve's ambient conditions at normal operating temperature and pressure. This changes the CTS by adding a Note that requires performance of the MSSV lift setpoint verification only in MODES 1 and 2.

The purpose of CTS 4.7.1.1 is to perform the MSSV lift setpoint verification in accordance with the Frequency of the INSERVICE TEST PROGRAM. This change is acceptable, because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The SR is modified by a Note that states the Surveillance is only required to be performed in MODES 1 and 2. The Note allows entry into and operation in MODE 3 prior to performing the SR. By allowing entry into MODE 3 prior to performing the SR, testing can be performed at ambient conditions of normal operating temperature and pressure. Otherwise, if the MSSVs are not tested at hot conditions, the lift setting pressure is corrected to ambient conditions of the valve at operating temperature and pressure. This change is designated as less restrictive because Surveillances will be performed in fewer operating Conditions than in the CTS.

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

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

3.7.1.1 LCO 3.7.1 Four ~~[Five]~~ MSSVs per steam generator shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTE--

DOC A03 Separate Condition entry is allowed for each MSSV.

~~---REVIEWER'S NOTE---~~
~~The * noted text is required for units that are licensed to operate at partial power with a positive Moderator Temperature Coefficient (MTC).~~
~~---~~ 3

CONDITION	REQUIRED ACTION	COMPLETION TIME
INSERT 1 ~~A. One or more steam generators with one MSSV inoperable [and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels]*.~~	~~A. 1 Reduce THERMAL POWER to ≤ [72] % RTP~~.	~~4 hours~~
~~B. One or more steam generators with two or more MSSVs inoperable.~~ ~~[OR~~ ~~One or more steam generators with one MSSV inoperable and the MTC positive at any power level.]*~~	~~B.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.~~ ~~AND~~	~~4 hours~~

1

 1

2

INSERT 1

Action b. Action b. Action b.	A. One or more steam generators with one or more MSSV inoperable. AND In MODE 1 or 2 with the Moderator Temperature Coefficient (MTC) zero or negative. OR In MODE 3 with the MTC positive, negative, or zero.	A.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours
Action a. Action a.	B. One or more steam generators with one or more MSSV inoperable AND In MODE 1 or 2 with a positive MTC.	B.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND B.2 --------------NOTE-------------- Only required in MODE 1. ------------------------------------- Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~B.2~~ ~~-------------NOTE-------------~~ ~~Only required in MODE 1.~~ ~~---------------------------------~~ ~~Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.~~	~~36 hours~~
ACTION a ACTION b — C. Required Action and associated Completion Time not met. <u>OR</u> One or more steam generators with all ~~≥ [4]~~ MSSVs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 18 ~~12~~ hours

1

1

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.1.1 DOC L03 — SR 3.7.1.1	------------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. ---	
4.7.1.1	Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift setting shall be within ±1%.	In accordance with the INSERVICE TESTING PROGRAM

CTS

Table 3.7-1

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
3 ~~[4]~~	~~[65]~~ 44
2 ~~3~~	~~[46]~~ 27
1 ~~2~~	~~[28]~~ 10

2

1

1

CTS

Table 3.7-2

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFT SETTING (psig ± ~~[~~3~~]~~%)
STEAM GENERATOR				
A ~~#1~~	B ~~#2~~	C ~~[#3]~~	~~[#4]~~	
~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~
~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~
~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~
~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~	~~[ ]~~

INSERT 2

2

1

2 **INSERT 2**

Table 3.7-2

RV1400	RV1405	RV1410	1085
RV1401	RV1406	RV1411	1100
RV1402	RV1407	RV1412	1105
RV1403	RV1408	RV1413	1105

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.1, MAIN STEAM SAFETY VALVES (MSSVs)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. Editorial changes made for consistency with Specification.

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

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Four ~~[Five]~~ MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the U FSAR, Section 10.2 ~~[10.3.1]~~ (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip. 1 2 1

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the U FSAR, Chapter 14 ~~Section [15.2]~~ (Ref. 3). Of these, the full power turbine trip without steam dump is typically the limiting AOO. This event also terminates normal feedwater flow to the steam generators. 2 1

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity

BASES

APPLICABLE SAFETY ANALYSES (continued)

is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The U FSAR ~~Section [15.4]~~ Chapter 14 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO. 2 1

The U FSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. ~~For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. [When the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during~~ 2 3 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs, it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions, unless it is demonstrated by analysis that a specified reactor power reduction alone is sufficient to prevent overpressurization of the steam system.]~~ 2

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires that ~~[five]~~ four MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that ~~[five]~~ four MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis. 1 1

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the INSERVICE TESTING PROGRAM.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, ~~[five]~~ four MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization. 1

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements. Operation with less than all four ~~[five]~~ MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. 1

A.1

In the case of only a single inoperable MSSV on one or more steam generators ~~[~~when the Moderator Temperature Coefficient is not positive~~]~~, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1 requires an appropriate reduction in reactor power within 4 hours. 1

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for calorimetric power uncertainty.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~To determine the maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs, the governing heat transfer relationship is the equation q =~~ $\dot{m}$ ~~Δh, where q is the heat input from the primary side,~~ $\dot{m}$ ~~is the mass flow rate of the steam, and Δh is the increase in enthalpy that occurs in converting the secondary side water to steam. If it is conservatively assumed that the secondary side water is all saturated liquid (i.e., no subcooled feedwater), then the Δh is the heat of vaporization (h_{fg}) at the steam relief pressure.~~ 4
The following equation is used to determine the maximum allowable power level for continued operation with inoperable MSSV(s):

BASES

ACTIONS (continued)

Nuclear Steam Supply System (NSSS)

$$\text{Maximum } \cancel{\text{NSSS}} \text{ Power} \leq (100/Q)\,(w_s h_{fg} N) / K$$ 5

where:

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt

K = Conversion factor, 947.82 (Btu/sec)/MWt

w_s = Minimum total steam flow rate capability of the OPERABLE MSSVs on any one steam generator at the highest OPERABLE MSSV opening pressure, including tolerance and accumulation, as appropriate, lbm/sec.

h_{fg} = Heat of vaporization at the highest MSSV opening pressure, including tolerance and accumulation as appropriate, Btu/lbm.

N = Number of steam generators in the plant.

For use in determining the %RTP in the Required Action statement A.1, the Maximum NSSS Power calculated above is reduced ~~by [2]% RTP~~ to account for ~~calorimetric power~~ uncertainty. 2 4

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. ~~[Furthermore, for a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is positive the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient.]~~ The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. 1

BASES

ACTIONS (continued)

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

~~------------------------------REVIEWER'S NOTE------------------------------~~
~~To determine the Table 3.7.1-1 Maximum Allowable Power for Required Actions B.1 and B.2 (%RTP), the Maximum NSSS Power calculated using the equation in the Reviewer's Note above is reduced by [9]% RTP to account for Nuclear Instrumentation System trip channel uncertainties.~~
~~--~~ 4

2

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ ~~[~~4~~]~~ inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~12~~ 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 1

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the INSERVICE TESTING PROGRAM. The ASME Code (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM~~-1-1987~~ (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination,

BASES

SURVEILLANCE REQUIREMENTS (continued)

b. Seat tightness determination,

c. Setpoint pressure determination (lift setting),

d. Compliance with owner's seat tightness criteria, and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± ~~[~~3~~]~~% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure. 1

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. U FSAR, Section ~~[10.3.1]~~ 10.2. 1 2

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. U FSAR, ~~Section [15.2]~~ Chapter 14. 1 2

4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

5. ANSI/ASME OM~~-1-1987~~ 2004 Edition through 2006 Addenda. 2

6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.1 BASES, MAIN STEAM SAFETY VALVES (MSSVs)

1. The Improved Standard Technical Specification (ISTS) contain bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. This redundant example has been deleted.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Editorial/grammatical changes made.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.1, MAIN STEAM SAFETY VALVES (MSSVS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

ITS 3.7.2, MAIN STEAM ISOLATION VALVES (MSIVS)

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

A01

ITS ITS 3.7.2

PLANT SYSTEMS

(MSIVs)

MAIN STEAM LINE ISOLATION VALVES A01

LIMITING CONDITION FOR OPERATION

Three s

LCO 3.7.2 3.7.1.5 ~~Each main steam line isolation valve (~~MSIV~~)~~ shall be OPERABLE. A01

, excepted when all MSIVs are closed

Applicability APPLICABILITY: MODES 1, 2, and 3. L01

ACTION:

ACTION A MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in MODE 2 within the next 6 hours.

ACTION C MODES 2 and 3:

Add Action C Note A02

With one or more MSIVs inoperable, subsequent operation in MODE 2 or 3 may continue provided:

ACTION C 1. The inoperable MSIVs are closed within 8 hours, and

ACTION C 2. The inoperable MSIVs are verified closed once per 7 days.

ACTION D Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

the isolation time of each MSIV is within limits LA01

SR 3.7.2.1 4.7.1.5 ~~Each MSIV shall be demonstrated OPERABLE by~~ verify~~ing full closure within 5 seconds when tested~~ in accordance with the INSERVICE TESTING PROGRAM. ~~The provisions of Specification 4.0.4 are not applicable for entry into MODE 3~~. A03

Add proposed SR 3.7.2.2 M01

~~PLANT SYSTEMS~~

~~3/4 3.7.1.6 DELETED~~

~~PLANT SYSTEMS~~

A01

~~4.7.1.6.4 DELETED~~

DISCUSSION OF CHANGES
ITS 3.7.2, MAIN STEAM ISOLATION VALVES (MSIVs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

A02 CTS 3.7.1.5 ACTIONS (MODES 2 and 3) 1 and 2 provide compensatory actions for one or more inoperable Main Steam Isolation Valves (MSIVs). CTS 3.7.1.5 ACTION 1 (MODES 2 and 3) requires that within 8 hours, inoperable MSIV(s) are closed. CTS 3.7.1.2 ACTION 2 (MODES 2 and 3) requires that once per 7 days the inoperable MSIV(s) be verified closed. ITS 3.7.1 ACTION C Note states "Separate Condition entry is allowed for each MSIV." This changes the CTS by explicitly specifying separate condition entry for each inoperable MSIV.

The purpose of the CTS ACTIONS is to allow separate condition entry for each inoperable MSIV. Each time it is discovered that an MSIV is inoperable, entry is required and the specified Completion Time is allowed to complete the compensatory actions. The ITS 3.7.2 ACTIONS Note allows a separate Completion Time clock for each MSIV that is inoperable. This change is acceptable, because it only provides clarification of the Completion Time when one valve is inoperable and, subsequently, a second valve becomes inoperable. This change is designated as administrative, because it does not result in a technical change to the Specifications.

A03 CTS 4.7.1.5 requires verification of each required MSIV lift setpoint, and states "The provisions of Specification 4.0.4 are not applicable for entry into MODE 3." ITS Surveillance Requirement (SR) 3.7.2.1 does not contain this statement. However, ITS SR 3.7.2.1 does contain a Note that states, "Only required to be performed in MODES 1 and 2." This changes the CTS by not adding the CTS 4.0.4 exception.

The CTS 4.0.4 exception allows entry into MODE 3 to perform CTS Surveillance 4.7.1.5. This exception is not required in ITS SR 3.7.2.1 because the SR Note in the ITS only requires the SR to be performed in MODES 1 and 2. This change is designated as administrative as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS SR 3.7.2.2 states, in part, "Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal." The CTS does not contain this Surveillance Requirement. This changes the CTS by adding the ITS requirement of SR 3.7.2.2.

The purpose of ITS SR 3.7.2.2 is to verify the actuation system of the MSIVs. This SR is normally performed upon returning the plant to operation following a refueling outage. This change is acceptable because it provides additional assurance that the actuation system will be capable of performing its specified safety function. This change is designated as more restrictive because it adds a SR to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 3.7.1.5 requirement to verify the closure time of the MSIV contains the actual closure time of 5 seconds. ITS 3.7.2 requires verification of the isolation time of the MSIV but does not include the isolation time. This changes the CTS moving the actual closure time acceptance criteria from the Technical Specification to the TS Bases.

The purpose of the SR is to verify the closure time of the MSIVs. The removal of MSIV closure time from the Technical Specifications and moving it to the Technical Specification Bases is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the requirement to verify that the isolation time of each MSIV is within limits. Also, this change is acceptable because this detail will be adequately controlled in the Technical Specification Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because MSIV closure times is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (Category 2 – Relaxation of Applicability) CTS 3.7.1.5 is applicable in MODES 1, 2, and 3. ITS Limiting Condition for Operation (LCO) 3.7.2 is applicable in MODE 1, and in MODES 2 and 3 except when all MSIVs are closed. This changes the CTS by making the Specification not applicable in MODES 2 and 3 when all MSIVs are closed.

The purpose of the ITS 3.7.2 Applicability exception is to clarify that an MSIV is not required to be OPERABLE when the valve(s) is in a position that supports the safety analyses. This change is acceptable, because the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses. When in the closed position, the valves are in the assumed accident position. This change is designated as less restrictive, because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS.

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

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

Three

3.7.1.5 LCO 3.7.2 ~~[Four]~~ MSIVs shall be OPERABLE. 2

Applicability APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed ~~[and de-activated]~~. 2

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION MODE 1	A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	24 ~~[8]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~	2
ACTION - MODE 1	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours	
ACTION - MODE 2 and 3, 1 and 2	C. ------------NOTE------------ Separate Condition entry is allowed for each MSIV. --------------------------------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	~~[~~8~~]~~ hours Once per 7 days	2
ACTION – MODE 2 and 3	D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours	

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.7.1.5 DOC L02	SR 3.7.2.1	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify the isolation time of each MSIV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
DOC M01	SR 3.7.2.2	-------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.2, MAIN STEAM ISOLATION VALVES (MSIVs)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

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

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by either low steam generator pressure or high containment pressure. The MSIVs fail closed on loss of control or actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

U A description of the MSIVs is found in the FSAR, Section 10.2 ~~[10.3]~~ (Ref. 1). 1 2

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the U FSAR, Section 14.3.4 ~~[6.2]~~ (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the U FSAR, Section 14.2.5 ~~[15.1.5]~~ (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). 1 2

The limiting case for the containment analysis is the SLB inside containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. ~~The large SLB inside containment at hot zero power is the limiting case for a post trip return to power.~~ The analysis includes scenarios with offsite power available~~, and with a loss of offsite power following turbine trip~~. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. ~~With a loss of offsite power, the response of mitigating systems is delayed~~. Significant single failures considered include failure of ~~an MSIV to close~~. 1

a main steam check valve (MSCV), a main feedwater control valve (FCV), an auxiliary feedwater pump flow control valve, and loss of a containment safeguards train due to EDG sequencer failure

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the ~~MSIV~~ main steam check valve (MSCV). in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators. 1

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

three

This LCO requires that ~~[four]~~ MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. 1

While the Main Steam Bypass Valves (MSBV) and Main Steam Check Valves (MSCV) support the Main Steam Isolation function, no Technical Specification Limiting Condition for Operation or Action applies to them.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the ~~10 CFR 100~~ (Ref. 4) limits or the NRC staff approved licensing basis. 1

10 CFR 50.67

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally most of the MSIVs are closed, and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within ~~[~~8~~]~~ hours ~~[or in accordance with the Risk Informed Completion Time Program]~~. Some repairs to the MSIV can be made with the unit hot. The ~~[~~8~~]~~ hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs. 2

The ~~[~~8~~]~~ hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation. 2

B.1

If the MSIV cannot be restored to OPERABLE status within ~~[~~8~~]~~ hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems. 2

BASES

ACTIONS (continued)

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The ~~[~~8~~]~~ hour Completion Time is consistent with that allowed in Condition A. 2

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is within the limit given in Reference 5 and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 6), requirements during operation in MODE 1 or 2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is in accordance with the INSERVICE TESTING PROGRAM.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. ~~[The Frequency of MSIV testing is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3
~~---~~]

REFERENCES

1. U FSAR, Section ~~[10.3]~~ 10.2.
2. U FSAR, Section ~~[6.2]~~ 14.3.4. 1 2
3. U FSAR, Section ~~[15.1.5]~~ 14.2.5.
4. ~~10 CFR 100.11~~ 10 CFR 50.67. 1
5. ~~[Technical Requirements Manual~~ Inservice Testing Program.] 2
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.2 BASES, MAIN STEAM ISOLATION VALVES (MSIVs)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS Bases contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.2, MAIN STEAM ISOLATION VALVES (MSIVs)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 3.7.3, FEEDWATER ISOLATION VALVES (FIVs) AND FEEDWATER CONTROL VALVES (FCVs)

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

PLANT SYSTEMS

3/4.7.1.7 FEEDWATER ISOLATION

LIMITING CONDITION FOR OPERATION

LCO 3.7.3 3.7.1.7 Six Feedwater Control Valves (FCVs) ~~both main and bypass~~ and six Feedwater Isolation Valves (FIVs) ~~both main and bypass~~ shall be OPERABLE.* A04

Applicability APPLICABILITY: MODES 1, 2 and 3** except when all FIVs, and FCVs are closed or isolated by a manual valve. A03 L01

ACTION:

ACTION B / ACTION D a. With one or more FCVs inoperable, ~~restore operability, or~~ close or isolate the inoperable FCVs within 72 hours, and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. A02

ACTION A / ACTION D b. With one or more FIVs inoperable, ~~restore operability, or~~ close or isolate the inoperable FIV(s) within 72 hours, and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. A02

c. With one or more bypass valves in different steam generator flow paths inoperable, ~~restore operability, or~~ close or isolate the inoperable bypass valve(s) within 72 hours, and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. A04

ACTION C / ACTION D d. With two valves in the same steam generator flow paths inoperable, ~~restore operability, or~~ isolate the affected flowpath within 8 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. A02

SURVEILLANCE REQUIREMENTS

4.7.1.7 Each FCV, FIV and bypass valve shall be demonstrated OPERABLE:

SR 3.7.3.2 a. In accordance with the Surveillance Frequency Control Program by:

1) Verifying that each FCV, FIV ~~and bypass valve~~ actuates to the isolation position on an actual or simulated actuation signal. A04

SR 3.7.3.1 b. In accordance with the INSERVICE TESTING PROGRAM by:

1) Verifying that each FCV, FIV ~~and bypass valve~~ isolation time is within limits. A04

ACTIONS Note *Separate Condition entry is allowed for each valve.

SR 3.7.3.1 and SR 3.7.3.2 Note ~~**The provisions of specification 4.0.4 are not applicable~~. Not required to be met in MODE 3. A03

DISCUSSION OF CHANGES
ITS 3.7.3, FEEDWATER ISOLATION VALVES (FIVS) AND FEEDWATER CONTROL VALVES (FCVS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

A02 CTS 3.7.1.7 ACTION a states, in part, to either restore OPERABILITY, or close or isolate the inoperable Feedwater Control Valve (FCV) within 72 hours. CTS 3.7.1.7 ACTION b states, in part, to either restore operability, or close or isolate the inoperable Feedwater Isolation Valve (FIV) within 72 hours. CTS 3.7.1.7 ACTION c states, in part, to either restore operability, or close or isolate the inoperable bypass valve(s) within 72 hours. CTS 3.7.1.7 ACTION d states, in part, to either restore operability, or isolate the affected flow path within 8 hours. ITS 3.7.3 ACTIONS A, B, C and D do not contain the statement to restore an inoperable FIV, FCV, bypass valve, or affected flow path to OPERABLE status. This changes the CTS by not including the statement to restore an inoperable FIV, FCV, bypass valve, or affected flow path to OPERABLE status.

This change is acceptable because the technical requirements have not changed. Restoration to compliance with the Limiting Condition for Operation (LCO) is always an available Required Action and it is the convention in the ITS to not state "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.7.1.7 Applicability Footnote ** contains the provision that Specification 4.0.4 is not applicable for entry into MODE 3. CTS 4.0.4 states that entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified frequency, except as provided by Specification 4.0.3. ITS 3.7.3 contains a similar Note that is associated with SR 3.7.3.1 and SR 3.7.3.2 stating these surveillances are not required to be met in MODE 3. This changes the CTS by providing similar wording that the surveillances associated with ITS LCO 3.7.3 are not required to be met for entry into MODE 3.

The CTS Applicability footnote allows entry into MODE 3 without meeting the Surveillance Requirement (SR). This portion of the note allows continuous operation in MODE 3. Allowances provided by the CTS and ITS accomplish equivalent results; both allow entry into and operation in MODE 3. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 3.7.1.7 states that six FCVs (both main and bypass) and six FIVs (both main and bypass) shall be OPERABLE. CTS 3.7.1.7 ACTION c states "With one or more bypass valves in different steam generator flow paths inoperable, restore operability, or close or isolate the inoperable bypass valve(s) within 72 hours, and verify that the inoperable valve(s) is closed or isolated at least once per 7 days." CTS 4.7.1.7 requires verification of each FCV, FIV, and bypass valve's actuation and isolation time. ITS 3.7.3 requires three FIVs, three FCV, and their associated bypass valves to be OPERABLE. This changes the CTS by grouping the bypass valves (both FIV and FCV) in a separate group.

This change is acceptable because the reference to "bypass valves" is included in the requirement to have six FCVs and six FIVs OPERABLE. PTN is a three loop Westinghouse plant with three steam generators (SGs). Each SG has a main feedwater line that has a flow control valve and an isolation valve. In addition, the main flow control valve and isolation valve have a bypass line around each FIV and FCV. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

<u>MORE RESTRICTIVE CHANGES</u>

None

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

None

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.7.1.7 requires three FIVs, three FCVs, three FIV bypass valves, or three FCV bypass valves to be OPERABLE in MODES 1, 2 and 3. ITS LCO 3.7.3 requires three FIVs, three FCVs, three FIV bypass valves, and three FCV bypass valves to be OPERABLE in MODES 1, 2 and 3 except when all FIVs, FCVs, and associated bypass valves are closed or isolated by a closed manual valve. This changes the CTS by making the Specification not applicable in MODES 1, 2, and 3 when all FIVs, FCVs, and associated bypass valves are closed or isolated by a closed manual valve.

The purpose of CTS 3.7.1.7 Applicability is to ensure that the FIVs, FCVs and associated bypass valves are OPERABLE and capable of closing when required to support the safety analysis. This change is acceptable because the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. When the FIVs, FCVs, and associated bypass valves are in the closed position or isolated, the valves are in the assumed accident position. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

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

3.7 PLANT SYSTEMS

3.7.3 ~~Main~~ Feedwater Isolation Valves (~~M~~FIVs) and ~~Main~~ Feedwater Control Valves (~~M~~F~~R~~Vs) and ~~[~~Associated Bypass Valves~~]~~ 2 1

C

Three three

3.7.1.7 LCO 3.7.3 ~~[Four]~~ ~~M~~FIVs~~,~~ ~~[four]~~ ~~M~~F~~R~~Vs~~,~~ ~~[~~and associated bypass valves~~]~~ shall be OPERABLE. 1 2

C

C

Applicability APPLICABILITY: MODES 1, ~~[and 2]~~ ~~[~~2, and 3~~]~~ except when ~~M~~FIV, ~~M~~F~~R~~V~~,~~ ~~[~~or associated bypass valve~~]~~ is closed ~~and [de-activated]~~ ~~[~~or isolated by a closed manual valve~~]~~. 1

ACTIONS

---NOTE---

LCO 3.7.1.7 Note * Separate Condition entry is allowed for each valve.

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
ACTION b	A. One or more ~~M~~FIVs inoperable.	A.1 Close or isolate ~~M~~FIV. AND A.2 Verify ~~M~~FIV is closed or isolated.	~~[~~72~~]~~ hours Once per 7 days	2 1
ACTION a	B. One or more ~~M~~F~~R~~Vs inoperable. (C)	B.1 Close or isolate ~~M~~F~~R~~V. (C) AND B.2 Verify ~~M~~F~~R~~V is closed or isolated. (C)	~~[~~72~~]~~ hours Once per 7 days	2 1 2
ACTION c	C. ~~[~~One or more ~~[MFRV or preheater]~~ bypass valves inoperable. in different steam generator flow paths	C.1 Close or isolate bypass valve. AND C.2 Verify bypass valve is closed or isolated.	~~[~~72~~]~~ hours Once per 7 days ~~]~~	

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
ACTION d	D. Two valves in the same flow path inoperable.	D. 1	Isolate affected flow path.	8 hours	
ACTION a, ACTION b, ACTION c	E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours	
		[AND			1
		E.2	Be in MODE 4.	12 hours]	

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.7.1.7 b.1	SR 3.7.3.1	Verify the isolation time of each ~~M~~FIV, ~~M~~F~~R~~V[, and associated bypass valve] is within limits. C -----NOTE----- Not required to be met in MODE 3. ----------	In accordance with the INSERVICE TESTING PROGRAM	2 1 3
4.7.1.7 a.1	SR 3.7.3.2	Verify each ~~M~~FIV, ~~M~~F~~R~~V[, and associated bypass valves] actuates to the isolation position on an actual or simulated actuation signal. C -----NOTE----- Not required to be met in MODE 3. ----------	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]	2 1 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.3, FEEDWATER ISOLATION VALVES (FIVs), FEEDWATER CONTROL VALVES (FCVs)

1. The Improved Standard Technical Specification (ISTS) contain bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

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

3. The Improved Technical Specifications (ITS) is adopting the Turkey Point Nuclear Generating Station (PTN) current licensing basis allowance to enter MODE 3 without having to perform the applicable Surveillance Requirements (SRs). Instead of incorporating an exception in the Applicability as with the PTN Current Technical Specifications (CTS), a Note will be added to the SRs that would allow entry into MODE 3 with the SR unperformed. This exception is required to ensure the proper test environment is available to perform the test.

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

B 3.7 PLANT SYSTEMS

B 3.7.3 ~~Main~~ Feedwater Isolation Valves (~~M~~FIVs) and ~~Main~~ Feedwater ~~Regulation~~ Control Valves (~~M~~F~~R~~Vs) C ~~[~~and Associated Bypass Valves~~]~~ 1 2

BASES

BACKGROUND

C The ~~M~~F~~I~~Vs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The ~~safety related~~ function of the ~~M~~F~~R~~Vs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. INSERT 1 I C Closure of the ~~M~~FIVs and associated bypass valves or ~~M~~F~~R~~Vs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the ~~M~~FIVs or ~~M~~F~~R~~Vs. C The consequences of events occurring in the main steam lines or in the MFW lines downstream from the ~~M~~FIVs will be mitigated by their closure. Closure of the ~~M~~FIVs and associated bypass valves, or ~~M~~F~~R~~Vs and associated bypass valves, C effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) ~~or FWLBs~~ inside containment, and reducing the cooldown effects for SLBs.

C The ~~M~~FIVs and associated bypass valves, or MF~~R~~Vs and associated bypass valves, isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops. 1

C One ~~M~~FIV and associated bypass valve, and one ~~M~~F~~R~~V and its associated bypass valve, are located on each MFW line, outside but close to containment. C The ~~M~~FIVs and ~~M~~F~~R~~Vs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following ~~M~~FIV or ~~M~~F~~R~~V closure. C The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

C The MFIVs and associated bypass valves, and MF~~R~~Vs and associated bypass valves~~,~~ , safety injection, close on receipt ~~of a Tavg - Low coincident with reactor trip (P-4)~~ or steam generator water level - high high signal. ~~They may also be actuated manually.~~ The FCVs also close on receipt of a Tavg - Low coincident with reactor trip. In addition to the ~~M~~FIVs and associated bypass valves, and the ~~M~~F~~R~~Vs and associated bypass valves, C a check valve inside containment is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems.

1 **INSERT 1**

The LCO requires three FCVs, three FIVs, three bypass line control valves, and three bypass line isolation valves to be OPERABLE.

BASES

BACKGROUND (continued)

U
C A description of the ~~M~~FIVs and ~~M~~F~~R~~Vs is found in the FSAR, 2
10.2 Section ~~[10.4.7]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES

C The design basis of the ~~M~~FIVs and ~~M~~F~~R~~Vs is established by the analyses for the large SLB. ~~It is also influenced by the accident analysis for the large FWLB~~. Closure of the ~~M~~FIVs and associated bypass valves, or ~~M~~F~~R~~Vs and associated bypass valves, may also be relied on to C terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level - high high signal ~~or a feedwater isolation signal on high steam generator level~~.

Feedwater isolation valves do not have a credited function in the Feedwater Line Break analysis.

C Failure of an ~~M~~FIV, ~~M~~F~~R~~V, or the associated bypass valves to close following an SLB ~~or FWLB~~ can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB ~~or FWLB~~ event.

C The ~~M~~FIVs and ~~M~~F~~R~~Vs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the ~~M~~FIVs, ~~M~~F~~R~~Vs, and their associated bypass valves will isolate MFW flow to the steam generators, following a~~n FWLB or~~ main steam line break. These valves will also isolate the nonsafety related portions from the safety related portions of the system. 1

three three C C This LCO requires that ~~[four]~~ MFIVs and associated bypass valves~~-and [four] M~~F~~R~~Vs ~~[~~and associated bypass valves~~]~~ be OPERABLE. The ~~M~~FIVs and ~~M~~F~~R~~Vs and the associated bypass valves~~-~~are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal. 2

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB ~~or FWLB~~ inside containment. If a feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

C The ~~M~~FIVs and ~~M~~F~~R~~Vs and the associated bypass valves~~-~~must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, ~~[~~and 3~~]~~, the ~~M~~FIVs and C ~~M~~F~~R~~Vs and the associated bypass valves~~-~~are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed ~~and de-activated~~ or isolated by a closed manual valve, they are already performing their safety function. 2

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the ~~M~~FIVs~~,~~ ~~M~~F~~R~~CVs, and the associated bypass valves are normally closed since MFW is not required. 1

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one ~~M~~FIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within ~~[~~72~~]~~ hours. When these valves are closed or isolated, they are performing their required safety function. 1 2

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The ~~[~~72~~]~~ hour Completion Time is reasonable, based on operating experience. 2 2

Inoperable ~~M~~FIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated. 1

B.1 and B.2

With one ~~M~~F~~R~~CV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within ~~[~~72~~]~~ hours. When these valves are closed or isolated, they are performing their required safety function. 1 2

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The ~~[~~72~~]~~ hour Completion Time is reasonable, based on operating experience. 2 2

BASES

ACTIONS (continued)

C Inoperable ~~M~~F~~R~~Vs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated. 1

C.1 and C.2

With one associated bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within ~~[~~72~~]~~ hours. When these valves are closed or isolated, they are performing their required safety function.

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The ~~[~~72~~]~~ hour Completion Time is reasonable, based on operating experience. 1

Inoperable associated bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1 1

With two inoperable valves in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the ~~M~~FIV or ~~M~~F~~R~~V, or otherwise isolate the affected flow path. C 1

BASES

ACTIONS (continued)

E.1 and E.2 1

C If the ~~M~~FIV(s) and ~~M~~F~~R~~V(s) and the associated bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours~~[~~, and in MODE 4 within 12 hours~~]~~. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 1 2

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

C This SR verifies that the closure time of each ~~M~~FIV, ~~M~~F~~R~~V, and ~~[~~associated bypass valve~~]~~ is within the limit given in Reference 2 and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2. 2 1

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to the surveillance being met. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated. 4

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.7.3.2

C This SR verifies that each ~~M~~FIV~~,~~ MF~~R~~V, and ~~[~~associated bypass valves~~]~~ can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. 1 2

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to the surveillance being met. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

~~[The Frequency for this SR is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3
~~--]~~

REFERENCES

1. FSAR, Section ~~[10.4.7]~~ 10.2. 2
2. ~~[Technical Requirements Manual.]~~ In Service Testing Program 2
3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.3 BASES, FEEDWATER ISOLATION VALVES (FIVs) AND FEEDWATER CONTROL VALVES (FCVs)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS Bases contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. Changes are being made to the Improved Technical Specifications (ITS) Bases to be consistent with changes made to the ITS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.3, FEEDWATER ISOLATION VALVES (FIVS) AND FEEDWATER CONTROL VALVES (FCVS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

ITS 3.7.4, SECONDARY SPECIFIC ACTIVITY

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

ITS

PLANT SYSTEMS

SECONDARY

SPECIFIC ACTIVITY

A01

LIMITING CONDITION FOR OPERATION

LCO 3.7.4 3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.10 microCurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

ACTION A With the specific activity of the Secondary Coolant System greater than 0.10 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1 4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performing the sampling and analysis described in Table ~~4.7-1~~ B 3.7.4-1.

LA01

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	SFCP
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

LA01

DISCUSSION OF CHANGES
ITS 3.7.4, SECONDARY SPECIFIC ACTIVITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 LA0_ *(Type 4 – – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS SR 4.7.1.4 requires the specific activity of the secondary coolant system to be within the limit by performing the sampling and analysis described in Table 4.7-1 of the Technical Specifications. ITS Surveillance Requirement (SR) 3.7.4.1 contains the same SR but refers to performing the sampling and analysis described in Table B 3.7.4-1. This changes the CTS by moving the sampling and analysis Table for determining specific activity to the Technical Specification Bases.

The removal of the sampling and analysis Table for determining specific activity is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the requirement to perform the SR. Also, this change is acceptable because sampling and analysis criteria will be adequately controlled in the Technical Specification Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because the SR sampling and analysis criteria is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

3.7 PLANT SYSTEMS

4 3.7.~~18~~ Secondary Specific Activity 1

3.7.4.1 LCO 4 3.7.~~18~~ The specific activity of the secondary coolant shall be ≤ ~~[~~0.10~~]~~ µCi/gm DOSE EQUIVALENT I-131. 1 2

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION	A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
		AND	
		A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.7.1.4	SR 3.7.~~18~~.1 4 Verify the specific activity of the secondary coolant is ~~≤ [0.10] µCi/gm DOSE EQUIVALENT I-131~~. within limits.	~~[31 days~~ ~~OR~~ In accordance with ~~the Surveillance Frequency Control Program]~~ Table B 3.7.4-1.	1 3 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.4, Secondary Specific Activity

1. Improved Standard Technical Specification (ISTS) 3.7.18, "Secondary Specific Activity" has been renumbered as ITS 3.7.4, "Secondary Specific Activity."

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

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

B 3.7 PLANT SYSTEMS

B 3.7.~~18~~ 4 Secondary Specific Activity 1

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of ~~[1.0]~~ 0.25 μCi/gm DOSE EQUIVALENT I-131 (LCO 3.4.16, "RCS Specific Activity"). 2
The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

~~With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.~~ 3

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the ~~10 CFR 100~~ 10 CFR 50.67 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis. 3

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), INSERT 1 as discussed in the 14 U FSAR, Chapter ~~[15]~~ (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of ~~[~~0.10~~]~~ μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB , SGTR, or REA, do not exceed a small fraction of the unit ~~EAB~~ exclusion area boundary (EAB) limits (Ref. 1) for whole body and thyroid dose rates. 3 2 2 3 5

~~With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary~~ 3

3

INSERT 1

Steam Generator Tube Rupture (SGTR), and Rod Ejection Accident (REA),

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown~~. 3

~~In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event~~. ~~Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.~~ 3

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ ~~[~~0.10~~]~~ μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1). 2

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~18~~ 4 .1 1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. ~~[The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.~~ 2

INSERT 2

~~OR~~ 3

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~-----]~~

REFERENCES

1. ~~10 CFR 100.11~~ 10 CFR 50.67. 3

2. U FSAR, Chapter ~~[15]~~ 14. 1 2

INSERT 3 Table B 3.7.4-1 on new page B 3.7.4-4 3

3

INSERT 2

The specific activity will be determined to be within limit by performing the sampling and analysis described in Table B 3.7.4-1. The table describes two Frequencies when performing the isotopic analysis for DOSE EQUIVALENT I-131 concentration. The Frequency is required once per 31 days whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. The Frequency can be decreased to once per 6 months whenever the gross activity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines. The 31 day Frequency, when the concentrations are greater than10%, is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131 and allows for appropriate action to be taken to maintain level below the LCO limit. The 6 month Frequency, when the concentrations are less than or equal to 10%, indicates the level of DOSE EQUIVALENT I-131 are low enough such that the frequency of the sampling and analysis can be decreased while still maintaining a Frequency that can detect increasing trends in the concentrations.

3

INSERT 3

TABLE B 3.7.4-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	SFCP
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.4 BASES, SECONDARY SPECIFIC ACTIVITY

1. Improved Standard Technical Specification (ISTS) 3.7.18, "Secondary Specific Activity," has been renumbered as ITS 3.7.4, "Secondary Specific Activity."

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Editorial/grammatical changes made.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.4, SECONDARY SPECIFIC ACTIVITY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ITS 3.7.5, AUXILIARY FEEDWATER (AFW) SYSTEM

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

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.5 3.7.1.2 Two ~~independent~~ auxiliary feedwater trains including 3 steam supply flowpaths, 3 pumps ~~and associated discharge water flowpaths~~ shall be OPERABLE.~~(1)(2)~~ LA01 LA01

Applicability APPLICABILITY: MODES 1, 2 and 3

ACTION:

ACTIONS Note NOTE: LCO 3.0.4.b is not applicable to the required auxiliary feedwater trains when entering Mode 1.

ACTION C 1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or place the affected unit(s) in at least HOT STANDBY within the next
ACTION D 6 hours* and in HOT SHUTDOWN within the following 6 hours.

ACTION F 2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify that both Standby Feedwater Pumps are capable of providing makeup flow to the steam generators and place the affected unit(s) in at least HOT STANDBY within the next
ACTION D 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective
ACTION F action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

ACTION B 3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the
ACTION D operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours.

ACTION A 4) With a single steam supply flowpath inoperable, within 4 hours verify OPERABILITY of two independent steam supply flowpaths or follow ACTION statement 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent steam supply flowpaths, restore the inoperable steam supply flowpath to OPERABLE status within 7 days of discovery, or place the
ACTION D affected Unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours.

~~NOTES:~~

~~(1) One steam supply flowpath shall be OPERABLE in each AFW train and the third steam supply flowpath (via MOV-3-1404 for Unit 3 and MOV-4-1404 for Unit 4) shall be OPERABLE and aligned to either AFW train but not both simultaneously.~~

LA01

~~(2) During single and two unit operation, one pump shall be OPERABLE in each train and the third auxiliary feedwater pump shall be OPERABLE and capable of being powered from, and supplying water to either train, except as noted in ACTION 3 of Technical Specification 3.7.1.2. The third auxiliary feedwater pump (normally the "C" pump) can be aligned to either train to restore OPERABILITY in the event one of the required pumps is inoperable.~~

ACTION E *If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

LA02

SR 3.7.5.2 1) Verifying ~~by control panel indication and visual observation of equipment~~ that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or equal to 373 gpm to the entrance of the steam generators. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3;

SR 3.7.5.2 Note, SR 3.7.5.6 Note

SR 3.7.5.6 2) Verifying ~~by control panel indication and visual observation of equipment~~ that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance test above;

SR 3.7.5.1 3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

SR 3.7.5.7 4) Verifying that power is available to those components which require power for flow path operability.

b. In accordance with the Surveillance Frequency Control Program by:

SR 3.7.5.3 1) Verifying that each automatic valve in the flow path actuates to its correct position ~~upon receipt of each Auxiliary Feedwater Actuation test~~ on an actual or simulated actuation signal, and

L01

SR 3.7.5.4 2) Verifying that each auxiliary feedwater pump receives a start signal as designed automatically ~~upon receipt of each Auxiliary Feedwater Actuation test~~ on an actual or simulated actuation signal.

SR 3.7.5.5 4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 1 by verifying normal flow to each steam generator.

~~TABLE 3.7-3~~

~~AUXILIARY FEEDWATER SYSTEM OPERABILITY~~

~~DELETED~~

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS LCO 3.7.1.2 requires two independent auxiliary feedwater trains including 3 steam supply flowpaths, 3 pumps and associated discharge water flowpaths shall be OPERABLE. The CTS LCO also contains footnotes further describing specifics that are required in each AFW train. ITS LCO 3.7.5 states "Two AFW trains shall be OPERABLE including: a. three steam supply flowpaths, and b. three AFW pumps." The ITS does not include design details and associated flow paths that comprise an OPERABLE AFW train. This changes the CTS by moving the description of the AFW independence, trains, and components to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included to provide adequate protection of public health and safety. The ITS retains all necessary requirements in the LCO to ensure OPERABILITY for the AFW trains and specific details, such as, system composition, independence, etc., are located in the Bases. Also, this change is acceptable because the removed information will be controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program directs the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.7.1.2.1.a.1) and CTS 4.7.1.2.1.a.2) state, in part, to verify equipment performance, "by control panel indication and visual observation of equipment…". ITS 3.7.5 does not include this guidance on how to verify equipment performed as required. This changes the CTS by removing procedural details for meeting TS requirements.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements to verify that the steam turbine-driven pumps and the auxiliary feedwater valves operate as required. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program directs the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L01 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.7.1.2.b.1) and 4.7.1.2.b.2) require verification of the automatic actuation of auxiliary feedwater components upon receipt of each Auxiliary Feedwater Actuation test signal. ITS SR 3.7.5.3 and SR 3.7.5.4 specify that the signal may be from either an "actual" or simulated (i.e., test) signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the test.

The purpose of CTS 4.7.1.2.b.1) and 4.7.1.2.b.2) is to ensure that the auxiliary feedwater components operate correctly upon receipt of an actuation signal. This change is acceptable because using only a simulated signal to verify the correct actuation of components is not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

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

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

3.7.1.2 LCO 3.7.5 ~~[Three]~~ Two AFW trains shall be OPERABLE including:
a. three steam supply flowpaths, and
b. three AFW pumps (1) (2)

~~---NOTE---~~
~~[Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.]~~
~~--~~ (1)

Applicability APPLICABILITY: MODES 1, 2, and 3~~,~~
~~MODE 4 when steam generator is relied upon for heat removal~~. (2)

ACTIONS

---NOTE---
Actions Note LCO 3.0.4.b is not applicable ~~[~~when entering MODE 1.~~]~~ (1)

Action 4) (1) (3)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~[Turbine driven AFW train inoperable due to~~ one inoperable steam supply flowpath. ~~OR~~ ~~------------NOTE------------~~ ~~Only applicable if MODE 2 has not been entered following refueling.~~ ~~----------------------------~~ ~~One turbine driven AFW pump inoperable in MODE 3 following refueling.~~	~~A.1 Restore affected equipment to OPERABLE status.~~ A.1 Verify number of steam supply flowpaths OPERABLE. AND A.2 Restore inoperable steam supply flowpath to OPERABLE status.	~~7 days~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~ 4 hours 7 days

INSERT 1

3 INSERT 1

Action 3)	B. One AFW pump inoperable.	B.1.1 Verify OPERABILITY of two AFW trains. OR B.1.2 Enter Required Actions for inoperable AFW train(s). AND B.2 Restore inoperable AFW pump to OPERABLE status.	4 hours 4 hours 30 days

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action 1)	~~B~~ C. One AFW train inoperable ~~in MODE 1, 2, or 3 [for reasons other than Condition A]~~. OR Two inoperable steam supply flowpaths.	~~B~~ C.1 Restore AFW train to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~	3
	~~C. Turbine driven AFW train inoperable due to one inoperable steam supply.~~ ~~AND~~ ~~One motor driven AFW train inoperable.~~ INSERT 2	~~C.1 Restore the steam supply to the turbine driven train to OPERABLE status.~~ ~~OR~~ ~~C.2 Restore the motor driven AFW train to OPERABLE status.~~	~~[24 or 48] hours~~ ~~[24 or 48] hours~~	3
Action 1), Action 2), Action 3), Action 4)	INSERT 3 ~~D~~ E. Required Action and associated Completion Time of Condition A ~~[~~, B, ~~or C]~~ C, or D not met. ~~[OR~~ ~~Two AFW trains inoperable in MODE 1, 2, or 3 for reasons other than Condition C.]~~	~~D~~ E.1 Be in MODE 3. AND ~~D~~ E.2 ~~[~~Be in MODE 4.	6 hours ~~[18]~~ 12 hours ~~]~~	2

INSERT 4

3 **INSERT 2**

Action 2)	D. Two AFW Trains inoperable with both standby feedwater pumps capable of providing makeup flow to the steam generators.	D.1 Restore one AFW train to OPERABLE status.	2 hours

4 **INSERT 3**

------------------NOTE---------------
Condition E only applies to only one Unit during a dual Unit shutdown.
----

4 **INSERT 4**

Action 1), Action 2), Action 3), Action 4)	F. ------------NOTES------------ 1. Condition F only applies when a dual Unit shutdown is required. 2. Only one Unit can enter Condition F. --------------------------------- Required Action and Associated Completion Time of Condition A, B, C, or D not met.	F.1 Be in MODE 3 AND F.2 Be in MODE 4.	12 hours 18 hours

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
Action 2)	~~E~~ G. ~~[Three]~~ Two AFW trains inoperable ~~in MODE 1, 2, or 3~~ for reasons other than condition D.	~~E~~ G. 1	--------------NOTE-------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ------------------------------------- Initiate action to restore one AFW train to OPERABLE status.	Immediately ~~]~~	4 1 2 3 2 1
	~~F. Required AFW train inoperable in MODE 4.~~	~~F.1~~	~~Initiate action to restore AFW train to OPERABLE status.~~	~~Immediately~~	2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.7.1.2.1.a.3)	SR 3.7.5.1	~~----------NOTE---------- [AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.] ------------------------------~~ Verify each ~~AFW manual, power operated, and~~ non-automatic valve in ~~each water~~ the flow path, ~~[and in both steam supply flow paths to the steam turbine driven pump,]~~ that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 1 2 1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.7.1.2.1.a.1)	SR 3.7.5.2	INSERT 5 ~~-----NOTE-----~~ ~~[Not required to be performed for the turbine driven AFW pump until [24 hours] after ≥ [1000] psig in the steam generator.]~~ ~~----------~~ ~~Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.~~	~~In accordance with the INSERVICE TESTING PROGRAM~~	1 2
4.7.1.2.1.b.1)	SR 3.7.5.3	~~-----NOTE-----~~ ~~[AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.]~~ ~~----------~~ Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 1 1

2 INSERT 5

4.7.1.2.1.a.1)	SR 3.7.5.2	-------------------------------NOTE------------------------------ The provisions of SR 3.0.4 are not applicable for entry into MODES 2 and 3. --- Verify that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or equal to 373 gpm to the entrance of the steam generators.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.7.1.2.1.b.2)	SR 3.7.5.4	~~-----NOTES-----~~ ~~1. [Not required to be performed for the turbine driven AFW pump until [24 hours] after ≥ [1000] psig in the steam generator.]~~ ~~2. [AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.]~~ ~~----------~~ Verify each AFW pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 1 1
4.7.1.2.2	SR 3.7.5.5	~~[~~ Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE ~~2~~ 1 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days ~~]~~	1 1

INSERT 6 2

INSERT 7 2

2 **INSERT 6**

4.7.1.2.1.a.2)	SR 3.7.5.6	-------------------------------NOTE------------------------------ The provisions of SR 3.0.4 are not applicable for entry into MODES 2 and 3. --- Verify that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance of SR 3.7.5.2.	In accordance with the Surveillance Frequency Control Program

2 **INSERT 7**

4.7.1.2.1.a.4)	SR 3.7.5.7	Verify power is available to those components which require power for flow path OPERABILITY.	In accordance with the Surveillance Frequency Control Program

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.5, AUXILIARY FEEDWATER (AFW) SYSTEM

1. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The PTN AFW Systems contains three turbine-driven AFW pumps (and no motor-driven AFW pumps) that feed two trains. The PTN AFW System is different than the AFW System(s) that NUREG-1431 was based on, therefore changes were made to reflect the PTN current licensing basis.

4. PTN CTS includes an ACTION to perform a shut down of one of the units where the Completion Time is extended so that the units are shut down sequentially. An ACTION and ACTION Notes are added to allow for this sequential shut down. Later ACTIONs are relabeled.

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

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through ~~separate and independent~~ suction lines from the condensate storage tank (CST) (LCO 3.7.6, "Condensate Storage Tank (CST)") and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or atmospheric dump valves ~~(LCO 3.7.4, "Atmospheric Dump Valves~~ (ADVs)~~")~~. If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST. 1

INSERT 1

~~The AFW System consists of [two] motor driven AFW pumps and one steam turbine driven pump configured into [three] trains. Each motor driven pump provides [100]% of AFW flow capacity, and the turbine driven pump provides [200]% of the required capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds [two] steam generators, although each pump has the capability to be realigned from the control room to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.~~ 1

~~The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.~~

~~The turbine driven AFW pump supplies a common header capable of feeding all steam generators with DC powered control valves actuated to the appropriate steam generator by the Engineered Safety Feature Actuation System (ESFAS). One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.~~

1 **INSERT 1**

The AFW System is a shared system between Units 3 and 4. The AFW System consists of three steam turbine driven pumps configured into two trains. Each pump is capable of delivering 624.8 gpm to the steam generators between 1085 psig and 120 psig. The three pumps are installed such that each supply auxiliary feedwater to either Unit 3 or 4, with any single pump supplying the total feedwater requirement of either unit. Two pumps (B and C) are normally aligned to AFW Train 2 and the third (A) is normally aligned to AFW Train 1. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two DC motor operated valves and one AC motor operated valve on each unit isolate the three main steam lines from these headers. Both the DC and AC motor operated valves are powered from safety-related sources. Auxiliary feedwater can be supplied through redundant lines to the safety-related portions of the main feedwater lines to each of the steam generators. Each steam driven auxiliary feedwater pump has sufficient capacity for single- and two-unit operation to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal (RHR) System may be placed into operation.

The standby steam generator feedwater pumps are normally used to supply feedwater to the steam generators during normal start-up, shutdown, and hot standby conditions.

The turbine driven pumps are normally supplied with steam from the unit which has lost its normal feedwater supply. Steam can also be supplied from the unit having normal feedwater supply or from the unit's auxiliary steam supply. The supply valves will automatically open by any one of the following five signals.

a. Safety Injection.
b. Low-Low Level in any of the three steam generators.
c. Loss of both feedwater pumps under normal operating conditions.
d. Bus Stripping.
    1. Loss of voltage on either the A or B 4.16 KV bus.
    2. Degraded voltage on one 480V load center (instantaneous) coincident with safety injection and the emergency diesel generator breaker open.
    3. Degraded voltage on one 480V load center (delayed) coincident with the emergency diesel generator breaker open.
e. Anticipated Transient Without Scram (ATWS) Mitigating Actuation Circuitry (AMSAC) signal.

BASES

BACKGROUND (continued)

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

~~The AFW System actuates automatically on steam generator water level - low-low by the ESFAS (LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"). The system also actuates on loss of offsite power, safety injection, and trip of all MFW pumps.~~ 6

The AFW System is discussed in the U FSAR, Section ~~[10.4. 9]~~ 9.11 (Ref. 1). 1 2

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB) (limited scope analysis), ~~and~~ 1

b. Loss of MFW,

c. Main Steam Line Break (MSLB), and

d. Dual Unit Loss of Offsite Power (LOOP). 1

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident (LOCA).

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, the

BASES

APPLICABLE SAFETY ANALYSES (continued)

ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. One ~~motor driven~~ AFW pump would deliver to the broken MFW header ~~at the pump runout flow~~ until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump. 1

The ESFAS automatically actuates the AFW ~~turbine driven~~ pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. DC power operated valves are provided for each AFW line to control the AFW flow to each steam generator. 1

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. ~~[~~Three~~]~~ independent AFW pumps in ~~[three]~~ two diverse trains are required to be OPERABLE to ensure the availability of RHR capability for all events accompanied by a loss of offsite power and a single failure. ~~This is accomplished by powering two of the pumps from independent emergency buses.~~ The ~~third~~ AFW ~~pump is~~ pumps are powered by ~~a different means,~~ a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs. 2 1

two

The AFW System is configured into ~~[three]~~ two trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. INSERT 2 ~~This requires that the two motor driven AFW pumps be OPERABLE in [two] diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of [two] main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators.~~ The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE. 2 1

~~The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.~~ 1

1 **INSERT 2**

This requires that two independent auxiliary feedwater trains including 3 steam supply flow paths, 3 pumps and associated discharge water flow paths shall be OPERABLE.

One steam supply flow path shall be OPERABLE in each AFW train and the third steam supply flow path (via MOV-3-1404 for Unit 3 and MOV-4-1404 for Unit 4) shall be OPERABLE and aligned to either AFW train but not both simultaneously. This ensures that three AFW steam supply flow paths are OPERABLE and is typically carried out by aligning the SG/C steam supply flow path (via MOV-3/4-1405) to Train 1, the SG/A steam supply flowpath (via MOV-3/4-1403) to Train 2, and the SG/B steam supply flowpath (via MOV-3/4-1404) to either Train 1 or 2, but NOT both simultaneously. The SG/B steam supply flowpath is normally aligned to Train 2.

During single- and dual-unit operation, one AFW pump shall be OPERABLE in each train and the third AFW pump shall be OPERABLE and capable of being powered from and supplying water to either train. The third auxiliary feedwater pump (normally the "C" pump) can be aligned to either train to restore OPERABILITY in the event one of the required pumps is inoperable. This ensures that three AFW pumps are OPERABLE and is typically carried out by aligning the 'A' AFW pump to train 1, the 'B' AFW pump to train 2, and the 'C' AFW pump aligned to either train 1 or 2. The alignment applies during both single and dual Unit operation. The steam turbine, trip and throttle valve, and governor valve are support components for each AFW pump.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

Reactor Coolant System loops and the RHR system provide decay heat removal

In MODE 4 the ~~AFW System may be used for heat removal via the steam generators~~. 1

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

~~-----REVIEWER'S NOTE-----~~ 3
~~The LCO 3.0.4.b Note prohibits application of the LCO 3.0.4.b exception when entering MODE 1 if the plant does not depend on AFW for startup. If the plant does depend on AFW for startup, the Note should state, "LCO 3.0.4.b is not applicable."~~
~~----------~~

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train ~~[~~when entering MODE 1~~]~~. There is an increased risk associated with ~~[entering a MODE or other specified condition in the Applicability]~~ ~~[~~entering MODE 1~~]~~ with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance. 2

~~[~~A.1 and A.2 4

INSERT 3

~~If the turbine driven AFW train is inoperable due to one inoperable steam supply, or if a turbine driven pump is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days [or in accordance with the Risk Informed Completion Time Program]. The 7 day Completion Time is reasonable, based on the following reasons:~~ 1

~~a. For the inoperability of the turbine driven AFW pump due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump and the turbine driven train is still capable of performing its specified function for most postulated events.~~

~~b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.~~

1 **INSERT 3**

CONDITION A describes the actions to be taken when a single steam supply flowpath is inoperable. Three AFW steam supply flowpaths must be OPERABLE to satisfy the design basis requirement that the AFW System meet the single failure criterion in response to a MSLB. Per Required Action A.1, within 4 hours verification of the number of OPERABLE independent steam supply flowpaths confirms CONDITION A is the only applicable CONDITION otherwise CONDITION C or E apply. Upon verification of the OPERABILITY of two steam supply flowpaths, seven days is allotted to restore one inoperable steam supply flowpath to OPERABLE status from the time of discovery. The 7-day Completion Time is acceptable because the consequences of an inoperable steam supply flowpath are more severe than for an inoperable AFW pump. With a MSLB with one AFW steam supply flowpath out of service NOT associated with the faulted loop, failure of the remaining OPERABLE steam supply flowpath would cause a loss of AFW System function.

BASES

ACTIONS (continued)

~~c. For both the inoperability of the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps, and due to the low probability of an event requiring the use of the turbine driven AFW pump.~~

~~Condition A is modified by a Note which limits the applicability of the Condition for an inoperable turbine driven AFW pump in MODE 3 to when the unit has not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.]~~

INSERT 4

C ~~B~~.1

With one of the required AFW trains ~~(pump or flow path)~~ inoperable ~~in MODE 1, 2, or 3 [for reasons other than Condition A]~~, action must be taken to restore OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. This Condition includes the loss of two steam supply lines to the ~~turbine driven~~ AFW pump. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA occurring during this time period.

~~C.1 and C.2~~

~~With one of the required motor driven AFW trains (pump or flow path) inoperable and the turbine driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within [24] [48] hours. Assuming no single active failures when in this condition, the accident (a feedline break (FLB) or main steam line break (MSLB) could result in the loss of the remaining steam supply to the turbine driven AFW pump due to the faulted steam generator (SG). In this condition, the AFW System may no longer be able to meet the required flow to the SGs assumed in the safety analysis, [either due to the analysis requiring flow from two AFW pumps or due to the remaining AFW pump having to feed a faulted SG].~~

1 **INSERT 4**

B.1.1, B.1.2, and B.2

CONDITION B describes the actions to be taken when a single AFW Pump is inoperable. The requirement to verify that two independent Auxiliary Feedwater Trains are OPERABLE is accomplished by verifying that one OPERABLE AFW pump is aligned to each AFW train and that all three steam supply flowpaths are OPERABLE. The 30 day Completion Time is acceptable because two AFW trains remain OPERABLE, while ensuring the plant operates in a reduced configuration for a limited period.

BASES

ACTIONS (continued)

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Licensees should adopt the appropriate Completion Time based on their plant design. The 24 hour Completion Time is applicable to plants that can no longer meet the safety analysis requirement of 100% AFW flow to the SG(s) assuming no single active failure and a FLB or MSLB resulting in the loss of the remaining steam supply to the turbine driven AFW pump. The 48 hour Completion Time is applicable to plants that can still meet the safety analysis requirement of 100% AFW flow to the SG(s) assuming no single active failure and a FLB or MSLB resulting in the loss of the remaining steam supply to the turbine driven AFW pump.~~
~~---~~

~~[The 24 hour Completion Time is reasonable based on the remaining OPERABLE steam supply to the turbine driven AFW pump, the availability of the remaining OPERABLE motor driven AFW pump, and the low probability of an event occurring that would require the inoperable steam supply to be available for the turbine driven AFW pump.]~~

~~[The 48 hour Completion Time is reasonable based on the fact that the remaining motor driven AFW train is capable of providing 100% of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.]~~ 1

INSERT 5

E

~~D~~.1 and ~~D~~.2

A.2, B.2, C.1, or D.1

When Required Action ~~A.1 [B.1, C.1, or C.2]~~ cannot be completed within the required Completion Time~~, or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C,~~ the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~[18]~~ hours. 2 5 1

12

2

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.

~~In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.~~ 1

INSERT 6

1 **INSERT 5**

D.1

When two AFW trains are inoperable and both standby feedwater pumps are capable of providing makeup flow to the steam generators 2 hours are provided to restore one AFW train to OPERABLE status or commence a unit shutdown. The requirement to verify the availability of both standby feedwater pumps is to be accomplished by verifying that both pumps have successfully passed their surveillance tests within the last surveillance interval. The requirement to complete this action before beginning a unit shutdown is to ensure that an alternate feedwater train is available before putting the affected unit through a transient. If NO alternate feedwater trains are available, the affected unit is to stay at the same condition until an auxiliary feedwater train is returned to service, and then invoke CONDITION C for the other train. If both standby feedwater pumps are made available before one Auxiliary Feedwater Train is returned to an OPERABLE status, then the affected units shall be shutdown.

INSERT 6

F.1 and F.2

When Required Action A.2, B.2, C.1, or D.1, cannot be completed within the required Completion Time the units must be placed in a MODE in which the LCO does not apply. If these Required Actions affect both units and a dual unit shutdown is required, one unit is allowed to be placed in at least MODE 3 within 12 hours, and in MODE 4 within 18 hours while the other unit is shutdown in accordance with CONDITION E.

CONDITION F is modified by two Notes. The Note 1 states that CONDITION F only applies when a dual Unit shutdown is required. The Note 2 states that only one Unit can enter Condition F. Therefore, if a dual unit shutdown is required one unit will be shut down in accordance with CONDITION E while the other unit will be shut down in accordance with CONDITION F.

This will allow the orderly shutdown of one unit at a time and not jeopardize the stability of the electrical grid by imposing a simultaneous dual unit shutdown. 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.

BASES

ACTIONS (continued)

~~E~~ G.1 (4)

If ~~all [three]~~ two AFW trains are inoperable ~~in MODE 1, 2, or 3,~~ the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. (5) (2)

Required Action ~~E~~ G.1 is modified by a Note indicating that all required MODE changes are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition. (4) (4)

~~F.1~~ (1)

~~In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.~~ (1)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

~~[The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be~~ (2)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained.~~]

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

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---~~]

2

3

SR 3.7.5.2

INSERT 7

~~Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing as discussed in the ASME Code (Ref. 2) and the INSERVICE TESTING PROGRAM satisfies this requirement.~~

[~~This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.~~]

1

1

1 **INSERT 7**

This SR verifies that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or equal to 373 gpm to the entrance of the steam generators. The specified flow rate acceptance criteria conservatively bounds the limiting AFW flow rate modeled in the single unit Loss of Normal Feedwater analysis. Dual unit events such as a two unit Loss of Offsite Power require a higher pump flow rate, but it is not practical to test both units simultaneously. The flow surveillance test specified is considered to be a general performance test for the AFW System and does not represent the limiting flow requirement for AFW. Verification of correct operation will be made both from instrumentation within the Control Room and direct visual observation of the pumps.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is acceptable based on operating experience and the design reliability of the equipment.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 3

~~[The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System, OPERABILITY (i.e., the intended safety function) continues to be maintained.]~~ 2

 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. ~~In MODE 4, the required pump is already operating and the autostart function is not required. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ 5 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 3

~~This SR is modified by [a] [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The Note [2] states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW System. OPERABILITY (i.e., the intended safety function) continues to be maintained.]~~ 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[~~ SR 3.7.5.5 2

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE ~~2~~ 1 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. ~~]~~ 2

INSERT 8

~~-----REVIEWER'S NOTE-----~~
~~This SR is not required by those units that use AFW for normal startup and shutdown.~~ 3

REFERENCES

1. U FSAR, Section ~~[10.4.9].~~ 9.11 1

~~2. ASME Code for Operation and Maintenance of Nuclear Power Plants.~~ 1

1 **INSERT 8**

SR 3.7.5.6

This SR verifies that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required. Check valves in the AFW system that require full stroke testing under limiting flow conditions are tested in accordance with the INSERVICE TESTING PROGRAM. Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification of correct operation will be made both from instrumentation within the Control Room and direct visual observation of the pumps.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.5.7

This SR verifies that power is available to those components which require power for flow path OPERABILITY. This SR ensures the flow paths are available by verifying power to the components that ensure proper AFW flow path alignment.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.5 BASES, AUXILIARY FEEDWATER (AFW) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

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

5. PTN does not have a requirement to be OPERABLE in MODE 4, therefore identifying inoperabilities only in MODE 1, 2, or 3 is unnecessary and is deleted.

6. Duplicative information that was covered earlier.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.5, AUXILIARY FEEDWATER (AFW) SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

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

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

LCO 3.7.6 3.7.1.3 The condensate storage tanks (CST) system shall be OPERABLE with:

Opposite Unit in MODES 4, 5, 6, or defueled

SR 3.7.6.2 A minimum indicated water volume of 210,000 gallons in either or both condensate storage tanks.

Opposite Unit in MODES 1, 2 or 3

SR 3.7.6.1 A minimum indicated water volume of 420,000 gallons.

Applicability APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Opposite Unit in MODES 4, 5, 6, or defueled

Action C
Action D
With the CST system inoperable, within 4 hours restore the CST system to OPERABLE status or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Opposite Unit in MODES 1, 2 or 3

Action A
Action D
1) With the CST system inoperable due to indicating less than 420,000 gallons, but greater than or equal to 210,000 gallons indicated, within 4 hours restore the inoperable CST system to OPERABLE status or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Action B
Action E
Action B Note
2) With the CST system inoperable with less than 210,000 gallons indicated, within 1 hour restore the CST system to OPERABLE status or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.6.1
SR 3.7.6.2

4.7.1.3 The condensate storage tank (CST) system shall be demonstrated OPERABLE by verifying the indicated water volume is within its limit when the tank is the supply source for the auxiliary feedwater pumps in accordance with the Surveillance Frequency Control Program.

DISCUSSION OF CHANGES
ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

3.7.1.3 LCO 3.7.6 The CST System shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1, 2, and 3~~,~~
~~MODE 4 when steam generator is relied upon for heat removal~~. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
INSERT 1 ~~A. CST inoperable.~~	~~A.1 Verify by administrative means OPERABILITY of backup water supply.~~ ~~AND~~ ~~A.2 Restore CST to OPERABLE status.~~	~~4 hours~~ ~~AND~~ ~~Once per 12 hours thereafter~~ ~~7 days~~
~~B~~ D. Required Action and associated Completion Time of Condition A or C not met.	~~B~~ D.1 Be in MODE 3. AND ~~B~~ D.2 Be in MODE 4~~, without reliance on steam generator for heat removal~~.	6 hours ~~[24]~~ 12 hours

1 1 3 1

INSERT 2 1

1 **INSERT 1**

Action	A. CST System inoperable due to indicated volume < 420,000 gallons but ≥ 210,000 gallons with the opposite Unit in MODE 1, 2, or 3.	A.1 Restore CST System to OPERABLE status.	4 hours
Action	B. ------------NOTE------------ Applies to both units simultaneously. -------------------------------- CST System inoperable due to indicated volume < 210,000 gallons with the opposite Unit in MODE 1, 2, or 3.	B.1 Restore CST System to OPERABLE status.	1 hour
Action	C. CST System inoperable with the opposite Unit in MODE 4, 5, 6, or defueled.	C.1 Restore CST System to OPERABLE Status.	4 hours

1 **INSERT 2**

E. Required Action and associated Completion Time of CONDITION B not met.	E.1 Be in MODE 3 AND E.2 Be in MODE 4.	12 hours 18 hours

Action

SURVEILLANCE REQUIREMENTS

INSERT 3

	SURVEILLANCE	FREQUENCY	
4.7.1.3	SR 3.7.6.1 Verify the CST ~~level~~ indicated water volume is ≥ ~~[110,000~~ 420,000 gal~~]~~.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 3 3

INSERT 4 1

1 **INSERT 3**

3.7.1.3 --------------------------------NOTE---------------------------
Only required to be met when the opposite unit is in MODES 1, 2 or 3.
----

2 **INSERT 4**

3.7.1.3	SR 3.7.6.2	------------------------------NOTE------------------------------ Only required to be met when the opposite unit is in MODES 4, 5, 6, or defueled. --- Verify the CST indicated water volume is ≥ 210,000 gal.	In accordance with the Surveillance Frequency Control Program

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

1. Changes were made to reflect the Turkey Point Nuclear Generating Station (PTN) current licensing basis. The Condensate Storage Tank (CST) System is a shared system between the two units. There are two CSTs shared between the two units and it is called the CST system.

2. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

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

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST System provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST. 1

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass valves. The condensed steam is returned to the CST by the condensate transfer pump. ~~This has the advantage of conserving condensate while minimizing releases to the environment.~~ 1

INSERT 1

~~Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The CST is designed to Seismic Category I to ensure availability of the feedwater supply. Feedwater is also available from alternate sources.~~ 1

A description of the CST is found in the FSAR, Section ~~[9.2.6]~~ 9.11.3 (Ref. 1). 2

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the U FSAR, Chapters ~~[~~6~~]~~ and ~~[15]~~ 14 (Refs. 2 and 3, respectively). ~~For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs, followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate.~~ INSERT 2 2 1

The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the ~~following:~~ 1

~~a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine) and~~ 1

~~b.~~ Failure of ~~the~~ a steam driven AFW pump ~~(requiring a longer time for cooldown using only one motor driven AFW pump)~~. 1

1 **INSERT 1**

The CST System consists of two seismically designed CSTs each with a capacity of 250,000 gallons. A minimum indicated volume of 210,000 gallons is maintained for each unit in MODE 1, 2, or 3.

The minimum indicated volume includes an allowance for instrument indication uncertainties and for water deemed unusable because of vortex formation and the configuration of the discharge line.

1 **INSERT 2**

Normal water supply to the AFW pumps is from the two 250,000-gallon (nominal) CSTs, through locked open gate valves and check valves. Each tank contains a 210,000 gallons minimum indicated volume which assures a minimum usable volume of 195,331 gallons of demineralized water for the auxiliary feedwater pumps. The CST design sizing is based on allowing each unit to be taken from full power to MODE 3 following a loss of offsite power, and:

a. Kept at MODE 3 for 4 hours and then cooled to 350°F in 9 hours, at which point the Residual Heat Removal System will be put in service, or

b. Kept at MODE 3 for 18 hours.

BASES

APPLICABLE SAFETY ANALYSES (continued)

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for ~~[30 minutes]~~ 18 hours following a reactor trip from 102% RTP, ~~and then to cool down the RCS to RHR entry conditions,~~ assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line. 2

~~The CST level required is equivalent to a usable volume of ≥ [110,000 gallons], which is based on holding the unit in MODE 3 for [2] hours, followed by a cooldown to RHR entry conditions at [75]°F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis.~~ INSERT 3 1

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, and 3~~, and in MODE 4, when steam generator is being relied upon for heat removal,~~ the CST system is required to be OPERABLE. 1

In MODE 4, 5 or 6, the CST system is not required because the AFW System is not required. 1

ACTIONS

~~A.1 and A.2~~ INSERT 4

~~If the CST is not OPERABLE, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The CST must be restored to OPERABLE status within 7 days, because the backup supply may be~~ 1

1 **INSERT 3**

The OPERABILITY of the CST with the minimum indicated volume ensures that sufficient water is available to maintain the Reactor Coolant System at MODE 3 conditions for approximately 18 hours or maintain the Reactor Coolant System at MODE 3 conditions for 4 hours and 9 hours to cool down the Reactor Coolant System to below 350°F at which point the Residual Heat Removal System may be placed in operation.

1 **INSERT 4**

A.1

CONDITION A provides remedial actions if the CST system is not OPERABLE due to indicated volume < 420,000 gallons but ≥ 210,000 gallons with the opposite Unit in MODE 1, 2, or 3. If the opposite Unit is in MODE 1, 2, or 3 the CONDITION is applicable to both units. With both Units in MODE 1, 2, or 3, SR 3.7.6.1 is applicable and 420,000 gallons of water is required to be indicated in the CSTs. With 210, 000 gallons of indicated water volume enough water is available for one unit and 4 hours are allowed to restore the CST system to an OPERABLE status.

The 4 hour Completion Time is reasonable, considering the time required to restore the indicted volume to within limitations, a backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

B.1

CONDITION B provides remedial actions if the CST system is not OPERABLE due to indicated volume < 210,000 gallons with the opposite Unit in MODE 1, 2, or 3. If the opposite Unit is in MODE 1, 2, or 3 the CONDITION is applicable to both units. With both Units in MODE 1, 2, or 3, SR 3.7.6.1 is applicable and 420,000 gallons of water is required to be indicated in the CSTs. With < 210, 000 gallons of indicated water volume there is not enough water available for one unit and 1 hour is allowed to restore the CST system to an OPERABLE status.

Condition B is modified by a Note stating that this Condition applies to both units simultaneously. With less that indicated volume of 210,000 gallons of water there is not enough water to maintain the Reactor Coolant System in MODE 3 for the required length of time or cooldown to RHR for either.

The 1 hour Completion Time, similar to LCO 3.0.3, allows to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid.

C.1

CONDITION C provides remedial actions if the CST system is not OPERABLE with the opposite Unit in MODE 4, 5, 6, or defueled. If the opposite Unit is in MODE 4, 5, 6, or defueled the Condition is applicable to only the Unit in Mode 1, 2, or 3. With only one Unit in MODE 1, 2, or 3, SR 3.7.6.2 is applicable and 210,000 gallons of water is required to be indicated in the CSTs. If less than 210, 000 gallons is indicated 4 hour is allowed to restore the CST system to an OPERABLE status.

BASES

ACTIONS (continued)

~~performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.~~ 1

D

~~B~~.1 and ~~B~~.2 3

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, ~~without reliance on the steam generator for heat removal,~~ within ~~[24]~~ 12 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. INSERT 5 2

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. INSERT 6 ~~(The required CST volume may be single value or a function of RCS conditions.)~~ 1 ~~[The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~--]~~

INSERT 7 1

1 **INSERT 5**

E.1 and E.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the units must be placed in a MODE in which the LCO does not apply. To achieve this status, the units must be placed in at least MODE 3 within 12 hours, and in MODE 4, within 18 hours. The allowed Completion Times allow for the orderly shutdown of one unit at a time and not jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

1 **INSERT 6**

when the opposite Unit is also in MODE 1, 2, or 3. A minimum indicated volume of 210,000 gallons is maintained for each unit in MODE 1, 2 or 3. This volume provides margin over the analysis minimum required volume and includes an allowance for instrument indication uncertainties and for water deemed unusable because of vortex formation and the configuration of the discharge line.

1 **INSERT 7**

SR 3.7.6.2 verifies that the CST system contains the required volume of cooling water when the opposite Unit is also in MODE 4, 5, 6, or defueled. A minimum indicated volume of 210,000 gallons is maintained for the unit in MODE 1, 2 or 3. This volume provides margin over the analysis minimum required volume and includes an allowance for instrument indication uncertainties and for water deemed unusable because of vortex formation and the configuration of the discharge line.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

REFERENCES	1. U FSAR, Section ~~[9.2.6]~~ 9.11.3. 1 2
	2. U FSAR, Chapter ~~[~~6~~]~~. 1 2
	3. U FSAR, Chapter ~~[15]~~ 14. 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.6 BASES, CONDENSATE STORAGE TANK (CST)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes have been made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description. Where an addition/deletion has occurred, subsequent alphanumeric designators have been changed for any applicable affected Required Actions, Surveillance Requirements, Functions, or Footnotes.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

ITS 3.7.7, COMPONENT COOLING WATER (CCW) SYSTEM

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

PLANT SYSTEMS

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.7

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

Applicability

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

ACTION:

Action Note

NOTE: Enter applicable ACTIONS of LCO 3.4.1.3, "Reactor Coolant System - Hot Shutdown," for residual heat removal loops made inoperable by CCW.

Action B

Action D

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in ~~COLD SHUTDOWN within the following 30 hours~~.

Add proposed Required Action D.2 and associated Note

L01

Action C

Action D

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and ~~in COLD SHUTDOWN within the following 30 hours~~.

Add proposed Required Action D.2 and associated Note

L01

Action A

Action D

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in ~~COLD SHUTDOWN within the following 30 hours~~.

Add proposed Required Action D.2 and associated Note

L01

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

SR 3.7.7.4

a. In accordance with the Surveillance Frequency Control Program, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

SURVEILLANCE REQUIREMENTS (Continued)

Add proposed SR 3.7.7.1 Note A02

SR 3.7.7.1 b. 1) In accordance with the Surveillance Frequency Control Program verify that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position.

SR 3.7.7.5 2) In accordance with the Surveillance Frequency Control Program verify by a performance test the heat exchanger surveillance curves.*

c. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

that is not locked, sealed, or otherwise secured in position L02

SR 3.7.7.2 1) Each automatic valve servicing safety-related equipment actuates to its correct position on ~~a SI test~~ signal, and

an actual or simulated actuation L03

SR 3.7.7.3 2) Each Component Cooling Water System pump starts automatically on ~~a SI test~~ signal.

SR 3.7.7.6 3) Interlocks required for CCW operability are OPERABLE.

Not required to be performed until 72 hours after reaching a Reactor Coolant System Tavg of 547°F and prior to entry into MODE 2. A03

*~~Technical specification 4.7.2.b.2 is not applicable for entry into MODE 4 or MODE 3, provided that:~~

SR 3.7.7.5 Note ~~1) Surveillance 4.7.2.b.2 is performed no later than 72 hours after reaching a Reactor Coolant System Tavg of 547°F, and~~

~~2) MODE 2 shall not be entered prior to satisfactory performance of this surveillance.~~

DISCUSSION OF CHANGES
ITS 3.7.7, COMPONENT COOLING WATER SYSTEM (CCS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 4.7.2.b.1) does not contain explicit guidance concerning Component Cooling Water (CCW) system OPERABILITY when isolating CCW flow to individual components. ITS Surveillance Requirement (SR) 3.7.7.1 contains a Note, which states, "Isolation of CCW flow to individual components does not render the CCW system inoperable." This changes the CTS by adding an allowance that is not explicitly stated in the CTS.

The purpose of CTS 4.7.2.b.1) is to provide assurance that CCW is available to the appropriate plant components. This change is acceptable because by current use and application of the CTS, isolation of a component supplied with CCW does not necessarily result in a CCW system being considered inoperable, but the respective component may be declared inoperable for its system. This change clarifies this application.

This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 4.7.2.b.2) requires verification by a performance test the heat exchanger surveillance curves*. Footnote *, associated with CTS 4.7.2.b.2), states that "Technical specification 4.7.2.b.2 is not applicable for entry into MODE 4 or MODE 3, provided that: 1) Surveillance 4.7.2.b.2 is performed no later than 72 hours after reaching a Reactor Coolant System Tavg of 547°F, and 2) MODE 2 shall not be entered prior to satisfactory performance of this surveillance." ITS SR 3.7.7.5 requires verification by a performance test the heat exchanger surveillance curves and provides a similar Note that states, "Not required to be performed until 72 hours after reaching a Reactor Coolant System Tavg of 547°F and prior to entry into MODE 2." This changes the CTS by simplifying the wording of the CTS footnote while maintaining the footnotes intent.

This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.7.2, ACTIONS a, b, and c, identifie degraded conditions of the CCW system and provides specific Completion Times to restore the degraded condition or commence a unit shutdown. If a unit shutdown is required, each CTS 3.7.2 Action requires the unit be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.7.7 ACTION D, similarly, states that if the Required Action and associated Completion Time of CCW degraded conditions not met to be in MODE 3 in 6 hours and MODE 4 in 12 hours and is modified by a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. This changes the CTS by permitting a Required Action end state of HOT SHUTDOWN (MODE 4) rather that an end state of COLD SHUTDOWN (MODE 5).

One purpose of CTS 3.7.2, ACTIONS a, b, and c is to provide an end state, a condition that the reactor must be placed in, if the Required Actions allowing remedial measures to be taken in response to the degraded conditions with continued operation are not met. End states are usually defined based on placing the unit into a MODE or condition in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. MODE 5 is the current end state for LCOs that are applicable in MODES 1 through 4. This change is acceptable because the risk of the transition from MODE 1 to MODES 4 or 5 depends on the availability of alternating current (AC) sources and the ability to remove decay heat such that remaining in MODE 4 may be safer. During the realignment from MODE 4 to MODE 5, there is an increased potential for loss of shutdown cooling and loss of inventory events. Decay heat removal following a loss-of-offsite power event in MODE 5 is dependent on AC power for shutdown cooling whereas, in MODE 4, the turbine driven auxiliary feedwater (AFW) pump will be available. Therefore, transitioning to MODE 5 is not always the appropriate end state from a risk perspective. Thus, for specific TS conditions, Westinghouse Topical Report WCAP-16294-A R1 (ADAMS Accession No. ML103430249) justifies MODE 4 as an acceptable alternate end state to Mode 5. The proposed change to the Technical Specifications will allow time to perform short-duration repairs, which currently necessitate exiting the original mode of applicability. The MODE 4 TS end state is applied, and risk is assessed and managed in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." This proposed change is consistent with NRC approved TSTF-432-A Revision 1 (ADAMS Accession No. ML103360003), noticed for availability by the NRC in the Federal Register (77 FR 27814) on May

11, 2012. The NRC's approval of WCAP-16294-A included four limitations and conditions on its use as identified in Section 4.0 of the NRC Safety Evaluation associated with WCAP-16294-A. Implementation of these stipulations were addressed in the Bases of TSTF-432-A. Florida Power & Light implemented these limitations and conditions at PTN in the adoption of the associated TSTF-432-A Bases. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.7.2.c.1 requires verification that each automatic valve servicing safety-related equipment actuates to its correct position on a safety injection (SI) test signal. ITS SR 3.7.7.2 requires verification that each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. This changes the CTS by not requiring automatic valves that are locked, sealed or otherwise secured in position to be tested to verify that they automatically actuate to their correct position.

The purpose of CTS 4.7.2.c.1 is to provide assurance that the valves in the flowpath required to actuate in case of a design basis accident (DBA) actuate to the correct position. This change is acceptable because it has been determined that the relaxed SR acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Valves already in the correct position and are locked, sealed, or otherwise secured in position are not required to have the position verified or to be tested to automatically actuate because, in case of a DBA, the valves are already in the required position and secured to prevent changing from the required position. This change is designated as less restrictive because less stringent SRs are being applied in the ITS than were applied in the CTS.

L03 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.7.2.c.1 and CTS 4.7.2.c.2 require verifying that each automatic valve servicing safety-related equipment and each CCW system pump starts automatically on a SI test signal. ITS SR 3.7.7.2 and ITS SR 3.7.7.3 specify that the signal may be from either an "actual" or simulated (i.e., test) signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the test.

The purpose of CTS 4.7.2.c.1 and CTS 4.7.2.c.2 is to ensure that the automatic valves servicing safety related equipment and CCW system pumps operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed SR acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its specified safety functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent SRs are being applied in the ITS than were applied in the CTS.

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

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

3.7.2 LCO 3.7.7 ~~Two~~ The CCW ~~trains~~ System shall be OPERABLE with: 1

a. Three CCW pumps, and

b. Two CCW heat exchangers

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

ACTIONS INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
INSERT 2 ~~A. One CCW train inoperable.~~	~~A.1 --------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. --------------------------------------~~ ~~Restore CCW train to OPERABLE status.~~	~~72 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ 1
3.7.2 Action a, 3.7.2 Action b, 3.7.2 Action c. ~~B~~ D. Required Action and associated Completion Time ~~of Condition A~~ not met.	~~B~~ D.1 Be in MODE 3.	6 hours 2
	AND	2
DOC L01	~~B~~ D.2 --------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------------	2
DOC L01	Be in MODE 4.	12 hours

2 **INSERT 1**

Action Note

---NOTE---
Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW.
----

2 **INSERT 2**

Action c.	A. One required CCW heat exchanger inoperable.	A.1 Restore required inoperable CCW heat exchanger to OPERABLE status.	1 hour
Action a.	B. One CCW pump inoperable with independent power supplies to OPERABLE CCW pumps.	B.1 Restore inoperable CCW pump to OPERABLE status.	30 days
Action b.	C. One CCW pump inoperable without independent power supplies to OPERABLE CCW pumps. <u>OR</u> Two CCW pumps inoperable	C.1 Restore one inoperable CCW pump to OPERABLE status with independent power supply.	72 hours OR In accordance with the Risk Informed Completion Time Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.7.2.b.1)	SR 3.7.7.1	------------------------------NOTE------------------------------ Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3 3
4.7.2.c.1)	SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3 3
4.7.2.c.2)	SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3 3

INSERT 3

2

INSERT 3

4.7.2.a	SR 3.7.7.4	Verify that two heat exchangers and one pump are capable of removing design basis heat loads.	In accordance with the Surveillance Frequency Control Program
4.7.2.b.2)	SR 3.7.7.5	---------------------------NOTE--------------------------- Not required to be performed until 72 hours after reaching a Reactor Coolant System Tavg of 547°F and prior to entry into MODE 2. --- Verify by a performance test the heat exchanger surveillance curves.	In accordance with the Surveillance Frequency Control Program.
4.7.2.c.3)	SR 3.7.7.6	Verify interlocks required for CCW OPERABILITY are OPERABLE.	In accordance with the Surveillance Frequency Control Program.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.7, COMPONENT COOLING WATER SYSTEM (CCW)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes have been made to add/revise Actions and surveillance requirements consistent with the Turkey Point Nuclear Generating Station (PTN) current licensing basis. The PTN Improved Technical Specifications (ITS) requires the Component Cooling Water (CCW) System to have three CCW pumps and two CCW heat exchangers OPERABLE. The subsequent Actions, Required Actions, or surveillance requirements have been relabeled to reflect the additions and deletions.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

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

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. ~~The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment~~.

INSERT 1

~~A typical CCW System is arranged as two independent, full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components are isolated~~. 1

Additional information on the design and operation of the system, along with a list of the components served, is presented in the U FSAR, Section ~~[9.2.2]~~ 9.3.2 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the ~~Residual Heat Removal (~~RHR~~)~~ System. This may be during a normal or post accident cooldown and shutdown. 1 2

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for ~~one CCW train to remove the post loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase, with a maximum CCW temperature of [120]°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCW System, respectively. The normal temperature of the CCW is [80]°F, and, during unit cooldown to MODE 5 (Tcold < [200]°F), a maximum temperature of 95°F is assumed. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the ECCS pumps.~~ 1

INSERT 2

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

1

INSERT 1

The CCW System is considered as one of three loops in the Auxiliary Coolant System. The other two loops in the Auxiliary Coolant System are the residual heat removal (RHR) loop and the spent fuel pit (SFP) cooling loop. During normal full power operation, one component cooling water pump and two or three component cooling water heat exchangers accommodate the heat removal loads. Each of the two standby pumps provides 100% backup, during normal operation. Two pumps and three heat exchangers are used to remove the residual and sensible heat during unit shutdown to ensure the CCW shell side flow does not exceed established flow limits.

The CCW heat loads are transferred by the CCW System to the Intake Cooling Water (ICW) System. The CCW System serves as an intermediate system to provide a barrier between the ICW and the CCW-cooled components. This barrier prevents any potential leakage of radioactive fluid into the environment or any saltwater intrusion.

The CCW head tank accommodates normal expansion and limited in-leakage of water. The head tank and surge tank combine to accommodate contraction and ensure a continuous component cooling water supply until a leaking cooling line can be isolated. A single inlet/outlet line connects the head tank to the top of the surge tank. Two surge lines are provided on the surge tank, so that each surge line is connected to one of the two partitioned sections of the surge tank. The lines are connected to each of the two component cooling headers in the suction side of the component cooling water pumps.

Each pump is automatically started on receipt of a start signal from the emergency bus load sequencer, and all nonessential components are isolated. The emergency bus load sequencer is actuated by a loss of offsite power (LOOP), a safety injection (SI) signal on its associated unit, a SI from the opposite unit, or a combination LOOP/LOCA. Component cooling pumps A and B are sequenced on by their associated Train sequencer for these actuation signals except for the opposite units SI signal without a LOOP. The third safety related D switchgear, used as a swing bus, and manually aligned to either the A or B 4.16 kV bus of its respective unit is considered an extension of that power supply bus. The third (C) CCW pump of each unit, when powered from its associated unit's D bus, may be used to provide the independent power supply OPERABILITY requirement when a pump is out of service. Component cooling pump C is sequenced on based on which train Bus D is aligned to, e.g., if Bus 3D is aligned to Bus 3A the A train sequencer will control starting of the component cooling pump 3C. In addition, the C pump is interlocked with the A and B CCW pump such that for a pump start signal to initiate on a LOOP or SI signal, the supply breaker for either the A or B CCW pump (associated with the A or B 4.16 kV Bus to which it is aligned) must be OPEN and RACKED OUT.

With offsite power available the CCW pumps receive a start signal 25 seconds (nominally) after the sequencer load timing begins, 41 seconds (nominally) with offsite power unavailable.

1 **INSERT 2**

one pump and two heat exchangers to provide the heat removal capability for accidents that have been analyzed. Peak CCW system operating temperatures occur during post-accident operations due to elevated containment temperatures and unrestricted heat rejection into the CCW system. A calculated maximum CCW System supply temperature of 158.5°F is acceptable for post-accident operation.

For the Power Operation configuration, the CCW system supply temperature (CCW heat exchanger outlet temperature) should not exceed 105°F to provide adequate cooling to the reactor coolant pump (RCP) thermal barrier and motor bearings per the manufacturer's recommended guidelines. If the RCP is operating during a period when CCW temperature is above 105°F, the RCP motor bearing and seal injection water temperatures must be continuously monitored as per the applicable plant operating procedures. For the RHR Cooldown configurations, adequate Reactor Coolant System (RCS) cooldown performance is maintained with a CCW system supply temperature of 125°F.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CCW System also functions to cool the unit from RHR entry conditions (T_{cold} < ~~[~~350~~]~~°F), to MODE 5 (T_{cold} < ~~[~~200~~]~~°F), during normal and post accident operations. The time required to cool from ~~[~~350~~]~~°F to ~~[~~200~~]~~°F is a function of the number of CCW and RHR trains operating. INSERT 3 ~~One CCW train is sufficient to remove decay heat during subsequent operations with Tcold < [200]°F. This assumes a maximum service water temperature of [95]°F occurring simultaneously with the maximum heat loads on the system~~. 2 1

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

LCO

INSERT 4 ~~The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of CCW must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.~~

~~A CCW train is considered OPERABLE when:~~

~~a. The pump and associated surge tank are OPERABLE and~~

~~b~~ c. The associated piping, valves, ~~heat exchanger,~~ and instrumentation and controls required to perform the safety related function are OPERABLE. 1

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post-accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

Although the LCO for the CCW System is not applicable in MODES 5 and 6, the capability of the CCW System to perform its necessary related support functions may be required for OPERABILITY of supported systems.

1

INSERT 3

One CCW pump, two CCW heat exchangers will remove design basis accident heat loads and maintain ICW heat exchangers ≤ 100 °F. Additionally, cases were run to demonstrate cooldown times for normal operations and refueling operations.

1 **INSERT 4**

During normal full power operation, one CCW pump and two or three CCW heat exchangers accommodate the heat removal loads. Each of the two standby pumps provides 100% backup, during normal operation. Two pumps and three heat exchangers are utilized to remove the residual and sensible heat during unit shutdown. The most limiting single active failure considered was the loss of one diesel, which results in only one required CCW pump starting automatically to mitigate the consequences of the maximum hypothetical accident (MHA). One pump and two heat exchangers provide the heat removal capability for accidents that have been analyzed.

The CCW system is considered OPERABLE when:

a. Three CCW pumps are OPERABLE,

b. Two CCW heat exchangers are OPERABLE, and

BASES

INSERT 5

ACTIONS A.1 1

~~Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.~~ 1

If one required CCW ~~train~~ heat exchanger is inoperable, action must be taken to restore OPERABLE status within 1 ~~72~~ hour~~s~~ ~~[or in accordance with the Risk Informed Completion Time Program]~~. In this Condition, the remaining OPERABLE CCW ~~train~~ heat exchanger is adequate to perform the heat removal function. The 1 ~~72~~ hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE ~~train~~ heat exchanger, and the low probability of a DBA occurring during this period. 1

INSERT 6 1

~~B~~D.1 and ~~B~~D.2 3

If the CCW ~~train~~ system cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. 1

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 3, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action ~~B~~D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. 3

1 **INSERT 5**

The ACTIONS are modified by a Note indicating to enter applicable Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for RHR loops made inoperable by CCW. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

1 **INSERT 6**

B.1

If one CCW pump is inoperable with independent power supplies to OPERABLE CCW pumps 30 days is allowed to restore the inoperable CCW pump. The 30-day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy period.

C.1

If one CCW pump is inoperable without independent power supplies to OPERABLE CCW pumps or two CCW pumps are inoperable, action must be taken to restore one CCW pump to OPERABLE status with independent power supply within 72 hours or in accordance with the Risk Informed Completion Time Program. In this Condition, the remaining OPERABLE CCW pump or pumps without independent power supplies are adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the capabilities afforded by the OPERABLE pump(s), and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~---]~~

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under

BASES

SURVEILLANCE REQUIREMENTS (continued)

administrative controls. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~ 4

INSERT 7 1

1

INSERT 7

SR 3.7.7.4

This SR verifies that two heat exchangers and one pump are capable of removing design basis heat loads. OPERABILITY is confirmed prior to entry into MODE 4 by verifying that the ICW temperature is below the limits identified by curves generated based on historical data.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.5

This SR verifies that by a performance test the heat exchanger surveillance curves. Intake Cooling Water (ICW) temperatures and CCW heat exchanger performance is monitored to assure that adequate heat removal capability is maintained.

This surveillance is modified by a Note stating that the surveillance is only required to be performed in MODE 1 and MODE 2. By performing heat exchanger performance tests prior to entry into MODE 2, where more reliable data is obtained, CCW OPERABILITY and satisfactory heat exchanger performance is confirmed prior to reactor criticality.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.6

This SR verifies that the interlocks required for CCW OPERABILITY are OPERABLE. Verification the interlocks perform as required, along with performance of the other SRs, ensures the CCW System will function in accordance with the safety analysis.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

REFERENCES

1. U FSAR, Section ~~[9.2.2]~~ 9.3.2. 1 2

2. U FSAR, Section ~~[~~6.2~~]~~. 1 2

3. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.7 BASES, COMPONENT COOLING WATER (CCW) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description. Where a deletion has occurred, subsequent alphanumeric designators have been changed for any applicable affected Required Actions, Surveillance Requirements, Functions, and Footnotes.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.7, COMPONENT COOLING WATER (CCW) SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

ITS 3.7.8, INTAKE COOLING WATER (ICW)

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

PLANT SYSTEMS

3/4.7.3 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.8 3.7.3 The Intake Cooling Water System (ICW) shall be OPERABLE with:

a. Three ICW pumps, and
b. Two ICW headers.

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

ACTIONS Note NOTE: Enter applicable ACTIONS of LCO 3.4.1.3, "Reactor Coolant System - Hot Shutdown," for residual heat removal loops made inoperable by ICW.

ACTION:

Action A a. With only two ICW pumps with independent power supplies OPERABLE, restore the inoperable ICW pump to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and ~~in COLD SHUTDOWN within the following 30 hours~~.

Action D

Add proposed Required Action D.2 and associated Note L01

Action B b. With only one ICW pump OPERABLE or with two ICW pumps OPERABLE, but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and ~~in COLD SHUTDOWN within the following 30 hours~~.

Action D

Add proposed Required Action D.2 and associated Note L01

Action C c. With only one ICW header OPERABLE, restore two headers to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and ~~in COLD SHUTDOWN within the following 30 hours~~.

Action D

Add proposed Required Action D.2 and associated Note L01

SURVEILLANCE REQUIREMENTS

4.7.3 The Intake Cooling Water System (ICW) shall be demonstrated OPERABLE:

SR 3.7.8.1 a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

Add proposed SR 3.7.8.1 Note A02

b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

that is not locked, sealed, or otherwise secured in position L02

SR 3.7.8.2 1) Each automatic valve servicing safety-related equipment actuates to its correct position on ~~a SI test~~ signal, and

an actual or simulated actuation L03

SR 3.7.8.3 2) Each Intake Cooling Water System pump starts automatically on ~~a SI test~~ signal.

SR 3.7.8.4 3) Interlocks required for system operability are OPERABLE.

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 CTS 4.7.3.b.1 does not contain explicit guidance concerning Intake Cooling Water (ICW) system OPERABILITY when isolating ICW flow to individual components. ITS Surveillance Requirement (SR) 3.7.8.1 contains a Note, which states, "Isolation of ICW flow to individual components does not render the ICW system inoperable." This changes the CTS by adding an allowance that is not explicitly stated in the CTS.

The purpose of CTS 4.7.3.a is to provide assurance that ICW is available to the appropriate plant components. This change is acceptable because by current use and application of the CTS, isolation of a component supplied with ICW does not necessarily result in ICW system being considered inoperable, but the respective component may be declared inoperable for its system. This change clarifies this application. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.7.3, ACTIONS a, b, and c, identify degraded conditions of the ICW system and provides specific Completion Times to restore the degraded condition or commence a unit shutdown. If a unit shutdown is required each CTS 3.7.3 Action requires the unit be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.7.8 ACTION D states that if the Required Action and associated Completion Time of ICW degraded conditions are not met to be in MODE 3 in 6 hours and MODE 4 in 12 hours and is modified by a Note stating that Limiting Condition for Operation (LCO) 3.0.4.a is not applicable when entering MODE 4. This changes the CTS by allowing a Required Action end state of HOT SHUTDOWN (MODE 4) rather than an end state of COLD SHUTDOWN (MODE 5).

One purpose of CTS 3.7.3, ACTIONS a, b, and c, is to provide an end state, a condition that the reactor must be placed in, if the Required Actions allowing remedial measures to be taken in response to the degraded conditions with continued operation are not met. End states are usually defined based on placing the unit into a MODE or condition in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. MODE 5 is the current end state for LCOs that are applicable in MODES 1 through 4. This change is acceptable because the risk of the transition from MODE 1 to MODES 4 or 5 depends on the availability of alternating current (AC) sources and the ability to remove decay heat such that remaining in MODE 4 may be safer. During the realignment from MODE 4 to MODE 5, there is an increased potential for loss of shutdown cooling and loss of inventory events. Decay heat removal following a loss-of-offsite power event in MODE 5 is dependent on AC power for shutdown cooling whereas, in MODE 4, the turbine driven auxiliary feedwater (AFW) pump will be available. Therefore, transitioning to MODE 5 is not always the appropriate end state from a risk perspective. Thus, for specific TS conditions, Westinghouse Topical Report WCAP-16294-A R1 (ADAMS Accession No. ML103430249) justifies MODE 4 as an acceptable alternate end state to Mode 5. The proposed change to the Technical Specifications will allow time to perform short-duration repairs, which currently necessitate exiting the original mode of applicability. The MODE 4 TS end state is applied, and risk is assessed and managed in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." This proposed change is consistent with NRC approved TSTF-432-A Revision 1 (ADAMS Accession No. ML103360003), noticed for availability by the NRC in the Federal Register (77 FR 27814) on May 11, 2012. The NRC's approval of WCAP-16294-A included four limitations and conditions on its use as identified in Section 4.0 of the NRC Safety Evaluation associated with WCAP-16294-A. Implementation of these stipulations were addressed in the Bases of TSTF-432-A. Florida Power & Light implemented these limitations and conditions at PTN in the adoption of the associated TSTF-432-A Bases. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)*
CTS 4.7.3.b.1 requires verification that each automatic valve servicing safety-related equipment actuates to its correct position on a safety injection (SI) test signal. ITS SR 3.7.8.2 requires verification that each ICW System automatic valve servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. This changes the CTS by exempting automatic valves that are locked, sealed, or otherwise secured in position from being tested to verify that they automatically actuate to their correct position.

The purpose of CTS 4.7.3.b.1 is to provide assurance that the valves in the flow path that are required to actuate in case of a design basis accident (DBA) actuate to the correct position. This change is acceptable because it has been determined that the relaxed SR acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its specified safety functions. Valves already in the correct position and are locked, sealed, or otherwise secured in position are not required have the position verified or to be tested to automatically actuate because, in case of a DBA, the valves are already in the required position and secured to prevent changing from the required position. This change is designated as less restrictive because less stringent SRs are being applied in the ITS than were applied in the CTS.

L03 (*Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria*)
CTS 4.7.3.b.1 and CTS 4.7.3.b.2 require verifying that each automatic valve servicing safety-related equipment and each ICW system pump starts automatically on a SI test signal. ITS SR 3.7.8.2 and ITS SR 3.7.8.3 specify that the signal may be from either an "actual" or simulated (i.e., test) signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the test.

The purpose of CTS 4.7.3.b.1 and CTS 4.7.3.b.2 is to ensure that the automatic valves servicing safety related equipment and ICW system pumps operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed SR acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its specified safety functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change is designated as less restrictive because less stringent SRs are being applied in the ITS than were applied in the CTS.

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

 1

3.7 PLANT SYSTEMS

3.7.8 ~~Service Water System (SWS)~~ Intake Cooling Water (ICW) System 1

3.7.3 LCO 3.7.8 ~~Two SWS trains~~ The ICW System shall be OPERABLE with: 1

a. Three ICW pumps, and

b. Two ICW headers

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

ACTIONS INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action c C ~~A~~. INSERT 2 One ~~SWS train~~ ICW header inoperable.	C ~~A~~.1 ~~NOTES~~ ~~1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by SWS.~~ ~~2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SWS.~~ Restore ~~SWS train~~ ICW header to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~

2 1 3 4 1 5 5

2 INSERT 1

Action Note

---NOTES---
Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ICW.
----

2 INSERT 2

Action a.	A. One ICW pump inoperable with independent power supplies to OPERABLE ICW pumps.	A.1	Restore inoperable ICW pump to OPERABLE status.	14 days
Action b.	B. One ICW pump inoperable without independent power supplies to OPERABLE ICW pumps. <u>OR</u> Two ICW pumps inoperable	B.1	Restore one inoperable ICW pump to OPERABLE status with independent power supply.	72 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Action a, Action b, Action c — D ~~B~~. Required Action and associated Completion Time ~~of Condition A~~ not met.	D ~~B~~.1	Be in MODE 3.	6 hours 2
	AND 2		
	D ~~B~~.2	--------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- 2	
		Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.3.a SR 3.7.8.1	------------------------------NOTE------------------------------ Isolation of ICW System ~~SWS~~ flow to individual components does not render the ICW System ~~SWS~~ inoperable. --- 1	
	Verify each ICW ~~SWS~~ manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ 1 5 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 5

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.7.3.b.1)	SR 3.7.8.2	Verify each ICW ~~SWS~~ automatic valve in the flow path servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 5 5
4.7.3.b.2)	SR 3.7.8.3	Verify each ICW ~~SWS~~ pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 5 5

INSERT 3

2 **INSERT 3**

4.7.3.b.3)	SR 3.7.8.4	Verify interlocks required for system OPERABILITY are OPERABLE.	In accordance with the Surveillance Frequency Control Program.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.8, INTAKE COOLING WATER SYSTEM (ICWS)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes have been made to add/revise Actions and surveillance requirements consistent with the Turkey Point Nuclear Generating Station (PTN) current licensing basis. The PTN Improved Technical Specifications (ITS) requires the Intermediate Cooling Water (ICW) System to have three ICW pumps and two ICW heat headers OPERABLE. The subsequent Actions, Required Actions, and Surveillance Requirements have been relabeled to reflect the additions and deletions.

3. PTN Unit 3 and Unit 4 emergency diesel generators are designed with a self-contained cooling system, which consists of a forced circulation cooling water loop, to cool the engine directly, rejecting heat through an air-cooled radiator and do not depend on any support plant cooling water system. Therefore, Note 1 to ISTS 3.7.8 Required Action A.1 is unnecessary and deleted.

4. PTN ITS 3.7.8 includes four Actions while ISTS includes two Actions. To ensure the applicable Conditions and Required Actions of Limiting Condition for Operation (LCO) 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by the ICW System are entered when any Action is entered. ISTS Required Action A.1 Note is moved to show it is applicable to all the ITS Actions.

5. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

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

B 3.7 PLANT SYSTEMS

B 3.7.8 ~~Service Water System (SWS)~~ Intake Cooling Water (ICW) System

BASES

BACKGROUND

The ~~SWS~~ ICW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the ~~SWS~~ ICW System also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

INSERT 1

~~The SWS consists of two separate, 100% capacity, safety related, cooling water trains. Each train consists of two 100% capacity pumps, one component cooling water (CCW) heat exchanger, piping, valving, instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System].~~

Additional information about the design and operation of the ~~SWS~~ ICWS, along with a list of the components served, is presented in the U FSAR, Section ~~[9.2.1]~~ 9.6 (Ref. 1). The principal safety related function of the ~~SWS~~ ICWS is the removal of decay heat from the reactor via the ~~CCW~~ component cooling water (CCW) System.

APPLICABLE SAFETY ANALYSES

The design basis of the ~~SWS~~ ICW System is for one ~~SWS train~~ ICW pump and one header, in conjunction with the CCW System and ~~a 100% capacity~~ containment cooling system, to remove core decay heat following a design basis LOCA as discussed in the U FSAR, ~~Section [6.2]~~ Chapter 6.0 (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the ECCS pumps. The ~~SWS~~ ICW System is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ~~SWS~~ ICW System, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in the U FSAR, Section ~~[5.4.7]~~ 9.3, (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and RHR System trains that are operating. One ~~SWS train~~ ICW header is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ~~SWS~~ ICW temperature of ~~[95]~~ 104°F occurring simultaneously with ~~maxium~~ maximum heat loads on the system.

The ~~SWS~~ ICW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

1 **INSERT 1**

The ICW system supplies salt water to the tube side of the component cooling water (CCW) heat exchangers and to cold side of the turbine area cooling water heat exchangers. The redundant header system is provided with isolation valves that can be shut so that failure of one loop does not require immediate shutdown of the unit. The supply headers are redundant, but the return merges to a non-redundant discharge header that returns water to the discharge canal. The redundant ICW supply headers address the design for passive failure.

Three ICW pumps are provided for each unit. One, two, or three pumps are operated as required to support normal plant operating conditions. However, only one pump is required following a maximum hypothetical accident (MHA). The A and B pumps are powered by 4.16 kV buses which can be powered by each train's associated emergency diesel generator (EDG). The C pump is powered by a swing 4.16 kV safety related bus which can be powered, through aligning the bus manually, by either the train A or train B EDG associated with the same unit. This pump is interlocked, such that, it is started on a loss of offsite power (LOOP) or safety injection (SI) signal, if the supply breaker for the A or B ICW pump (associated with the A or B 4.16 kV Bus to which it is aligned) is open and racked out.

The ICW system provides sufficient redundancy so that at least one ICW pump will continue to operate to handle heat loads from DBAs following a postulated single active failure. A single ICW pump, however, is limited in its ability to supply the required cooling water to the CCW water heat exchangers during an accident when flow is also allowed to continue through the turbine plant cooling water (TPCW) heat exchangers. OPERABILITY of the ICW header during an accident is maintained by isolation of the TPCW.

The third safety related D switchgear, used as a swing bus, and manually aligned to either the A or B 4.16 kV bus of its respective unit is considered an extension of that power supply bus. The third (C) ICW pump of each unit, when powered from its associated unit's D bus, may be used to provide the independent power supply OPERABILITY requirement when a pump is out of service. The most limiting single active failure considered was the loss of one diesel, which results in only one required ICW pump starting automatically to mitigate the consequences of the MHA. However, the C pump is interlocked, and for a start signal to initiate on a LOOP or SI signal, the supply breaker for the A or B ICW Pump (associated with the A or B 4.16 kV Bus to which it is aligned) must be open and racked out. Technical Specification ACTION statements may be invoked for not ensuring that the second OPERABLE pump is powered from an independent safety related bus.

BASES

LCO

ICW headers

Two ~~SWS trains~~ are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power. 1

The ICW System

~~An SWS train~~ is considered OPERABLE during MODES 1, 2, 3, and 4 when: 1

Three ICW pumps are ,

a. ~~The pump is~~ OPERABLE ~~and~~

b. Two ICW headers are OPERABLE, and 1

c

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

APPLICABILITY

ICW System

In MODES 1, 2, 3, and 4, the ~~SWS~~ is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ~~SWS~~ and required to be OPERABLE in these MODES. 1 1

ICW System

ICW System

Although the LCO for the ~~SWS~~ is not applicable in MODES 5 and 6, the capability of the ~~SWS~~ to perform its necessary related support functions may be required for OPERABILITY of supported systems. 1

ACTIONS

C ~~A~~.1 INSERT 2 3

ICW header

If one ~~SWS train~~ is inoperable, action must be taken to restore OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this Condition, the remaining OPERABLE ~~SWS train~~ is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ~~SWS train~~ could result in loss of ~~SWS~~ function. ~~Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," should be entered if an inoperable SWS train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable SWS train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.~~ The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period. 1 2 1 1 3

ICW header

ICW header

ICW System

INSERT 3

1 **INSERT 2**

The ACTIONS Table is modified by a Note indicating to enter applicable Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for RHR loops made inoperable by ICW System. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

1 **INSERT 3**

A.1

If one ICW pump is inoperable with independent power supplies to OPERABLE ICW pumps 14 days is allowed to restore the inoperable ICW pump. The 14-day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy period.

B.1

If one ICW pump is inoperable without independent power supplies to OPERABLE ICW pumps or two ICW pumps are inoperable, action must be taken to restore one ICW pump to OPERABLE status with independent power supply within 72 hours or in accordance with the Risk Informed Completion Time Program. In this Condition, the remaining OPERABLE ICW pump or pumps without independent power supplies are adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the capabilities afforded by the OPERABLE pump(s), and the low probability of a DBA occurring during this period.

BASES

ACTIONS (continued)

~~B~~D.1 and ~~B~~D.2 3

If the ~~SWS~~ ICW System ~~train~~ cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. 1

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 4, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action ~~B~~D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. 3

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

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of ~~the SWS~~ ICW flow to components or systems may render those components inoperable, but does not affect the OPERABILITY of the ~~SWS~~ ICW System. 1 1

Verifying the correct alignment for manual, power operated, and automatic valves in the ~~SWS~~ ICW System flow path provides assurance that the proper flow paths exist for ~~SWS~~ ICW System operation. This SR does not apply to 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. 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.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~---]~~

SR 3.7.8.2

This SR verifies proper automatic operation of the ~~SWS~~ ICW System valves on an actual or simulated actuation signal. The ~~SWS~~ ICW System is a normally operating system that cannot be fully actuated as part of ~~normal~~ power operation testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. 1
~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 4

SR 3.7.8.3

This SR verifies proper automatic operation of the ICW System ~~SWS~~ pumps on an actual or simulated actuation signal. The ICW System ~~SWS~~ is a normally operating system that cannot be fully actuated as part of normal testing during normal power operation. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 1 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]~~ 4

INSERT 4

REFERENCES

1. UFSAR, Section 9.6 ~~[9.2.1]~~. 1 2
2. UFSAR, Chapter 6.0 ~~Section [6.2]~~. 1 2
3. UFSAR, Section 9.3 ~~[5.4.7]~~. 1 2
4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

1 **INSERT 4**

SR 3.7.8.4

This SR verifies that the interlocks required for ICW System OPERABILITY are OPERABLE. Verification that the interlocks perform as required, along with performance of the other SRs, ensures the ICW System will function in accordance with the safety analysis.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.8 BASES, INTAKE COOLING WATER SYSTEM (ICWS)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made to be consistent with changes made to the Specification.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.8, INTAKE COMPONENT COOLING WATER SYSTEM (ICWS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 9

ITS 3.7.9, ULTIMATE HEAT SINK (UHS)

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

PLANT SYSTEMS

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

LCO 3.7.9

Action A, SR 3.7.9.1

3.7.4 The ultimate heat sink shall be OPERABLE with an average supply water temperature less than or equal to 104°F.

Applicability

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

A02

Add proposed Required Action B.1, C.1 and associated Notes

ACTION:

Add proposed Required Action B.2, C.2 and associated Note

L01

Action B, Action C

Actions Note

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 12 hours and ~~in COLD SHUTDOWN within the following 30 hours~~. This ACTION shall be applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.4 The ultimate heat sink shall be determined OPERABLE:

SR 3.7.9.1

a. In accordance with the Surveillance Frequency Control Program by verifying the average supply water temperature* is less than or equal to 104°F.

Action A

b. At least once per hour by verifying the average supply water temperature* is less than or equal to 104°F, when water temperature exceeds 100°F.

SR 3.7.9.1 Note

*Portable monitors may be used to measure the temperature.

DISCUSSION OF CHANGES
ITS 3.7.9, ULTIMATE HEAT SINK (UHS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

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

A02 The CTS 3.7.4 ACTION, states, in part, with the requirements of the above specification (UHS temperature limitation) not satisfied to be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours. The CTS 3.7.4 further states that This ACTION shall be applicable to both units simultaneously. In addition, CTS Bases states that that when an ACTION statement requires a dual unit shutdown, the time to be in HOT STANDBY is 12 hours. This is to allow the orderly shutdown of one unit at a time and NOT jeopardize the stability of the electrical grid by imposing a dual unit shutdown. ITS 3.7.9 Condition B, (be in MODE 3 in 6 hours) is modified by a Note that states "Condition B only applies to one Unit during a dual Unit shutdown." ITS 3.7.9 Condition C (be in MODE 3 in 12 hours) is modified by two Notes: 1) "Condition C only applies when a dual Unit shutdown is required", and 2) "Only one Unit can enter Condition C." This modifies the CTS to clarify what Completion Times should be followed for an inoperable UHS.

The purpose of the CTS 3.7.4 ACTION is to ensure prompt Action is taken to restore the inoperable UHS or shutdown the unit when the UHS cannot be restored to an OPERABLE status within the specified Completion Times. The UHS temperature requirements ensure that sufficient cooling capacity is available either: (1) To provide normal cooldown of the facility, or (2) To mitigate the effects of accident conditions within acceptable limits. The Notes described above conservatively allow only one unit in a dual unit shutdown event to apply the 12-hour Completion Time to reach MODE 3. This change is intended to provide clarity with respect to the different CTS Completion Times which depend on the UHS. Because the new Notes clarify that only one unit can apply the 12-hours to be in MODE 3 consistent with the CTS ACTION as discussed in the CTS Bases, this change is considered Administrative as no technical changes are being made to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.7.4, Action, identifies a degraded condition of the Ultimate Heat Sink (UHS) and provides specific Completion Times to restore the degraded condition or commence a unit shutdown. If a unit shutdown is required, the CTS 3.7.4 Action requires the unit be in HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.7.9 ACTION B states that if the Required Action and associated Completion Time of UHS degraded conditions are not met to be in MODE 3 in 6 hours and MODE 4 in 12 hours and is modified by a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. ITS 3.7.9 ACTION C states that if the Required Action and associated Completion Time of UHS degraded conditions are not met to be in MODE 3 in 12 hours and MODE 4 in 18 hours and is modified by a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. This changes the CTS by allowing a Required Action end state of HOT SHUTDOWN (MODE 4) rather than an end state of COLD SHUTDOWN (MODE 5).

One purpose of CTS 3.7.4, Action is to provide an end state, a condition that the reactor must be placed in, if the Required Actions allowing remedial measures to be taken in response to the degraded conditions with continued operation are not met. End states are usually defined based on placing the unit into a MODE or condition in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. MODE 5 is the current end state for LCOs that are applicable in MODES 1 through 4. This change is acceptable because the risk of the transition from MODE 1 to MODES 4 or 5 depends on the availability of alternating current (AC) sources and the ability to remove decay heat such that remaining in MODE 4 may be safer. During the realignment from MODE 4 to MODE 5, there is an increased potential for loss of shutdown cooling and loss of inventory events. Decay heat removal following a loss-of-offsite power event in MODE 5 is dependent on AC power for shutdown cooling whereas, in MODE 4, the turbine driven auxiliary feedwater (AFW) pump will be available. Therefore, transitioning to MODE 5 is not always the appropriate end state from a risk perspective. Thus, for specific TS conditions, Westinghouse Topical Report WCAP-16294-A R1 (ADAMS Accession No. ML103430249) justifies MODE 4 as an acceptable alternate end state to Mode 5. The proposed change to the Technical Specifications will allow time to perform short-duration repairs, which currently necessitate exiting the original mode of applicability. The MODE 4 TS end state is applied, and risk is assessed and managed in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." This proposed change is consistent with NRC approved TSTF-432-A Revision 1 (ADAMS Accession No. ML103360003), noticed for availability by the NRC in the Federal Register (77 FR 27814) on May 11, 2012. The NRC's approval of WCAP-16294-A included four limitations and conditions on its use as identified in Section 4.0 of the NRC Safety Evaluation associated with WCAP-16294-A. Implementation of these stipulations were addressed in the Bases of TSTF-432-A. Florida Power & Light implemented these limitations and conditions at PTN in the adoption of the associated

TSTF-432-A Bases. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

3.7.4

LCO 3.7.9 The UHS shall be OPERABLE.

Applicability

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

ACTIONS

INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. [One or more cooling towers with one cooling tower fan inoperable.~~	~~A.1 Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
~~-----REVIEWER'S NOTE----- The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.~~ ~~------------------------------------~~ A ~~B~~. ~~[~~ Average supply Water temperature of the UHS > ~~[90]~~ 100 °F ~~and~~ ≤ ~~[ ]~~ 104 °F.	A ~~B~~.1 Verify average supply water temperature of the UHS is ≤ ~~[90]~~ 104 °F ~~averaged over the previous 24 hour period~~.	Once per hour~~]~~

1 **INSERT 1**

Action

------------------------------------NOTE-------------------------------------
These ACTIONS shall be applicable to both units simultaneously.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B INSERT 2 ~~C~~. ~~[~~Required Action and associated Completion Time ~~of Condition A or B~~ not met. OR~~]~~ UHS inoperable ~~[~~for reasons other than Condition A ~~or B]~~.	B ~~C~~.1 Be in MODE 3. AND B ~~C~~.2 --------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours

3 1 5 3 1 1 5

INSERT 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.7.9.1~~	~~[Verify water level of UHS is ≥ [562] ft [mean sea level].~~	~~[[24] hours~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
SR 3.7.9.~~2~~ 1	INSERT 4 ~~[~~Verify average water temperature of UHS is ≤ ~~[90]~~ 104 °F.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~

1 6 1 1

1 **INSERT 2**

Action

----------------NOTE-------------
Condition B only applies to one Unit during a dual Unit shutdown.
----

1 **INSERT 3**

Action

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTES----------- 1. Condition C only applies when both Units are being shutdown simultaneously. 2. Only one Unit is allowed to enter Condition C during simultaneous shutdown of both Units. -------------------------------- Required Action and associated Completion Time not met. OR UHS inoperable for reasons other than Condition A.	C.1 Be in MODE 3. <u>AND</u> C.2 --------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE 4.	12 hours 18 hours

1 **INSERT 4**

4.7.4 Footnote *

-----------------------------------NOTE-------------------------------------
Portable monitors may be used to measure the temperature.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.9.3~~	~~[Operate each cooling tower fan for ≥ [15] minutes.~~	~~[31 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
~~SR 3.7.9.4~~	~~[Verify each cooling tower fan starts automatically on an actual or simulated actuation signal.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

1

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.9, ULTIMATE HEAT SINK (UHS)

1. The Improved Standard Technical Specification (ISTS) contain bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

3. Changes have been made to delete Actions; the subsequent Actions and Required Actions have been renumbered to reflect the deletions.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. The Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) 3.7.9 (Ultimate Heat Sink) allows for 12 hours to be in MODE 3 if the inoperability applies to both units simultaneously. The Completion Time allows for an orderly sequential shutdown of both units when the inoperability of a component affects both units with equal severity and is reasonable based on operating experience to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Another 6 hours is allowed to reach MODE 4 and is reasonable based on operating experience to reach the required unit conditions from MODE 3 in an orderly manner.

6. Changes have been made that deleted Surveillance Requirements (SRs)' the subsequent SRs have been renumbered to reflect the deletions.

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

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the ~~Service Water System (SWS)~~ Intake Cooling Water (ICW) System and the Component Cooling Water (CCW) System. 1

INSERT 1

~~The UHS has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.~~ 1

~~A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.~~

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. INSERT 2 ~~Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on other source(s) and makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations), and multiple makeup water sources may be required~~. 1

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. For units that use UHS as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs ~~20~~ 99 minutes after a 1

1 **INSERT 1**

The heat generated by operation of Turkey Point is rejected to a closed cooling canal water system (i.e., the UHS). The cooling canal water system occupies an area approximately 2 miles wide by 5 miles long and includes 168 miles of earthen canals covering approximately 4370 acres of water surface. The average canal depth is 2.8 feet. The entire circulation route from the plant discharge back to plant intake is 13.2 miles and takes approximately 44 hours to complete.

The cooling canal system provides the coolant for the circulating water (CW) system and serves as the UHS for the safety-related intake cooling water (ICW) system. The CW system provides cooling water to the main plant condensers. The ICW system removes heat loads from the component cooling water (CCW) system during normal and accident conditions to support both reactor and containment heat removal requirements and spent fuel cooling requirements. The ICW system has three 100 percent capacity pumps. During normal operation, the ICW system provides cooling water to three 50 percent capacity CCW heat exchangers and two non-safety related turbine plant cooling water (TPCW) heat exchangers. The limit on Ultimate Heat Sink (UHS) temperature in conjunction with the surveillance requirements of LCO 3.7.7, Component Cooling Water System, will ensure that sufficient cooling capacity is available either: (1) To provide normal cooldown of the facility, or (2) To mitigate the effects of accident conditions within acceptable limits.

1 **INSERT 2**

Section 9.2.5 of U.S. Nuclear Regulatory Commission NUREG 0800, Standard Review Plan (SRP), "Ultimate Heat Sink," Revision 3, states that the UHS should be able to dissipate the maximum possible total heat load, including that of a loss of coolant accident (LOCA), under the worst combination of adverse environmental conditions, even freezing, and can cool the unit (or units, including a LOCA for one unit of a multi-unit station with one heat sink) for a minimum of 30 days without makeup. The CCW supply temperature profile to safety-related cooling loads during a design basis accident (DBA) is maintained by implementing stricter CCW cleanliness requirements, which meets the regulatory guidance in Section 9.2.5 of the SRP.

BASES

APPLICABLE SAFETY ANALYSES

design basis ~~loss of coolant accident (~~LOCA~~)~~. Near this time, the unit switches from injection to recirculation and the containment cooling systems and RHR are required to remove the core decay heat. 5

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include ~~worst expected meteorological conditions,~~ conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a manmade structure). ~~The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS.~~ 1 1

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

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it is ~~contains a sufficient volume of water~~ at or below the maximum temperature that would allow the ICW System ~~SWS~~ to operate ~~for at least 30 days~~ following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the ICW System ~~SWS~~. To meet this condition, the UHS temperature should not exceed 104 ~~[90~~°F~~]~~ ~~and the level should not fall below [562 ft mean sea level]~~ during normal unit operation. 1 1 2

APPLICABILITY

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

Although the LCO for the UHS is not applicable in MODES 5 and 6, the capability of the UHS to perform its necessary related support functions may be required for OPERABILITY of supported systems.

ACTIONS

~~[A.1~~

~~If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days [or in accordance with the Risk Informed Completion Time Program].~~

~~The 7 day Completion Time is reasonable based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable (in one or more cooling towers), the number of available systems, and the time required to reasonably complete the Required Action.]~~ 2

BASES

ACTIONS (continued)

~~[B~~A.1 3

~~REVIEWER'S NOTE~~
~~The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.~~ 4

With water temperature of the UHS > ~~[90]~~100°F, the design basis assumption 2
associated with initial UHS temperature are bounded provided the
temperature of the UHS ~~averaged over the previous 24 hour period~~ is 1
≤ ~~[90]~~104°F. With the water temperature of the UHS > ~~[90]~~104°F, long term 2
cooling capability of the ECCS loads and ~~DGs~~EDGs may be affected. 1
Therefore, to ensure long term cooling capability is provided to the ECCS
loads when water temperature of the UHS is > ~~[90]~~100°F, Required Action 2
~~B~~A.1 is provided to more frequently monitor the water temperature of the 3
UHS and verify the temperature is ≤ ~~[90]~~104°F ~~when averaged over the~~ 2 1
~~previous 24 hour period~~. The once per hour Completion Time takes into
consideration UHS temperature variations and the increased monitoring
frequency needed to ensure design basis assumptions and equipment
limitations are not exceeded in this condition. If the water temperature of
the UHS exceeds ~~[90]~~104°F ~~when averaged over the previous 24 hour period~~ 2
~~or the water temperature of the UHS exceeds [ ]°F~~, Condition ~~C~~B must be 1
entered immediately.~~]~~ 2

~~[C~~B.1 and ~~C~~B.2 3

If the Required Actions and Completion Times of Condition ~~[~~A ~~or B]~~ are
not met, or the UHS is inoperable for reasons other than Condition A ~~[or~~ 2
~~B]~~, ~~the unit~~ both units must be placed in a MODE in which overall plant risk is
reduced. To achieve this status, ~~the~~ one unit must be placed in at least 1
MODE 3 within 6 hours and in MODE 4 within 12 hours. INSERT 3

Remaining within the Applicability of the LCO is acceptable to accomplish
short duration repairs to restore inoperable equipment because the plant
risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~3~~2). In MODE 4 1
the steam generators and Residual Heat Removal System are available
to remove decay heat, which provides diversity and defense in depth. As
stated in Reference ~~3~~2, the steam turbine driven auxiliary feedwater pump 1
must be available to remain in MODE 4. Should steam generator cooling

1 **INSERT 3**

CONDITION B allows for one unit to be in Mode 3 within 6 hours because the inoperability applies to both units simultaneously. The Completion Time allows for an orderly sequential shutdown of both units when the inoperability of a component affects both units with equal severity and is reasonable based on operating experience to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Another 6 hours is allowed to reach MODE 4 and is also reasonable based on operating experience to reach the required unit conditions in an orderly manner. The other unit is shut down in accordance with CONDITION C.

BASES

ACTIONS (continued)

be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

B Required Action ~~C~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. 3

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. ~~]~~

INSERT 4

SURVEILLANCE REQUIREMENTS

~~[SR 3.7.9.1~~

~~This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. [The [24] hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ [562] ft [mean sea level].~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~ 1

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]]~~

1 **INSERT 4**

C.1 and C.2

If the Required Actions and Completion Times of CONDITION A are not met, or the UHS is inoperable for reasons other than CONDITION A, the units must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the second unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 18 hours. CONDITION C is modified by two Notes allowing one unit to be shut down in accordance with CONDITION B while the second unit is shut down in accordance with CONDITION C.

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 the steam generators and RHR System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 2, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. 3

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This also allows the orderly shutdown of one unit at a time and NOT jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[~~ SR 3.7.9.~~2~~ 1

2 3

ICW System This SR verifies that the ~~SWS~~ is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads ~~for 30 days~~ following a ~~Design Basis Accident~~. DBA INSERT 5 ~~[The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ [90°F].~~ 1 1 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]]~~ 4

~~[SR 3.7.9.3~~ 2

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

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~-----]]~~

1 **INSERT 5**

With the implementation of the CCW Heat Exchanger Performance Monitoring Program, the limiting UHS temperature can be treated as a variable with an absolute upper limit of 104°F without compromising any margin of safety. Demonstration of actual heat exchanger performance capability supports system operation with postulated canal temperatures greater than 104°F. Therefore, an upper Technical Specification limit of 104°F is conservative.

The SR is modified by a Note providing an option of monitoring UHS temperature by monitoring the temperature in the ICW System piping going to the inlet of the CCW heat exchangers. Monitoring UHS temperature after the ICW pumps, but prior to CCW heat exchangers is considered to be equivalent to temperature monitoring before the ICW pumps. The supply water leaving the ICW pumps will be mixed, and therefore, it will be representative of the bulk UHS temperature to the CCW heat exchanger inlet. The effects of pump heating on the supply water are negligible due to low ICW head and high water volume. Accordingly, monitoring UHS temperature after the ICW pumps, but prior to the CCW heat exchangers provides an equivalent location for monitoring UHS temperature.

The frequency of verifying UHS water temperature to ensure the limit of 104ºF is NOT exceeded when the water temperature is less than 100ºF is controlled under the Surveillance Frequency Control Program as there is ample (greater than or equal to 4ºF) margin to the limit.

For the verification of UHS average supply water temperature, an appropriate instrument uncertainty will be subtracted from the Acceptance Criteria to ensure the Technical Specification limit is not exceeded.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[SR 3.7.9.4~~

~~This SR verifies that each cooling tower fan starts and operates on an actual or simulated actuation signal. [The [18] month Frequency is consistent with the typical refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]]~~

2

REFERENCES

1. U FSAR, Section ~~[9.2.5]~~ 14.3.4. 1 2

~~2. Regulatory Guide 1.27.~~ 1

2 ~~3~~. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010. 4

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.9 BASES, ULTIMATE HEAT SINK (UHS)

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description. Where a deletion has occurred, subsequent alphanumeric designators have been changed for any applicable affected Required Actions, Surveillance Requirements, Functions, and Footnotes.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Editorial/grammatical changes made.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.9, ULTIMATE HEAT SINK (UHS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

ITS SECTION 3.7.10 – CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS)

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

PLANT SYSTEMS

3.7.10 3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) A01

LIMITING CONDITION FOR OPERATION

LCO 3.7.10 3.7.5 The Control Room Emergency Ventilation System shall be OPERABLE* ~~with:~~

See ITS 3.7.11
a. Three air handling units,

b. Two condensing units,

~~c. Two control room recirculation fans,~~

~~d. Two recirculation dampers,~~ LA01

~~e. One filter train,~~

~~f. Two isolation dampers in the normal outside air intake duct,~~

~~g. Two isolation dampers in the emergency outside air intake duct,~~

~~h. Control Room Envelope~~

APPLICABILITY APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:`

ACTION A
ACTION F
ACTION D & Condition Note
ACTION E Condition Note

a.1 With one air handling unit inoperable, within 7 days, restore the inoperable air handling unit to OPERABLE status or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in ~~COLD SHUTDOWN within the following 30 hours~~.

Add proposed Required Actions D.2, E.2 and Note L01

See ITS 3.7.11
a.2 With only one OPERABLE condensing unit, within 30 days, restore at least one of the inoperable condensing units to OPERABLE status or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

ACTION A
ACTION F
ACTION D & Condition Note
ACTION E & Condition Note

a.3 With one recirculation fan inoperable, within 7 days, restore the inoperable fan to OPERABLE status or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in ~~COLD SHUTDOWN within the following 30 hours~~.

Add proposed Required Actions D.2, E.2 and Note L01

LCO 3.7.10 Note *The Control Room Envelope (CRE) boundary may be opened intermittently under administrative control.

PLANT SYSTEMS

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

ACTION A
ACTION F
ACTION D & Condition Note
ACTION E & Condition

a.4 With one recirculation damper inoperable, within 7 days, restore the inoperable damper to OPERABLE status or, place and maintain at least one of the recirculation dampers in the open position and place the system in recirculation mode or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units and in COLD SHUTDOWN within the following 30 hours. LA02

ACTION B
ACTION F
REQUIRED ACTION B.2

a.5 With the filter train inoperable, e.g., an inoperable filter, and/or two inoperable recirculation fans, and/or two inoperable recirculation dampers, immediately suspend all movement of irradiated fuel and immediately initiate action to place the compensatory filtration unit in service and verify proper operation within 24 hours, following which movement of irradiated fuel may resume and within 7 days, restore the filter train to OPERABLE status.

ACTION D & Condition Note
ACTION E & Condition Note

With the above requirements not met, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

ACTION A
ACTION F
ACTION D & Condition Note
ACTION E & Condition Note

a.6 With an inoperable damper in the normal outside air intake, within 7 days, restore the inoperable damper to OPERABLE status or, place and maintain at least one of the normal outside air intake isolation dampers in the closed position and place the system in recirculation mode or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours. LA02

ACTION A
ACTION F
ACTION D & Condition
ACTION E & Condition Note

a.7 With an inoperable damper in the emergency outside air intake, within 7 days, restore the inoperable damper to OPERABLE status or, place and maintain at least one of the emergency outside air intake isolation dampers in the open position or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units and in COLD SHUTDOWN within the following 30 hours. LA02

PLANT SYSTEMS

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

ACTION C

b. With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary during MODES 1, 2, 3 or 4, immediately initiate action to implement mitigating actions. Within 24 hours, verify mitigating actions ensure CRE occupant radiological and chemical hazards will not exceed limits, and CRE occupants are protected from smoke hazards, and restore CRE boundary to OPERABLE status within 90 days.

ACTION D& Condition Note
ACTION E & Condition Note

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

ACTION G

With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary during MODES 5, 6 or during the movement of irradiated fuel assemblies, immediately suspend all movement of irradiated fuel.

ACTION H

Add Conditions D and E proposed second Condition - Two or more CREVS required redundant components inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B or C.

L02

SURVEILLANCE REQUIREMENTS

4.7.5 The Control Room Emergency Ventilation System shall be demonstrated OPERABLE:

See ITS 3.7.11

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 120°F;

SR

b. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes~~**~~;

See ITS 5.5.8

SR

c. In accordance with the Surveillance Frequency Control Program or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system, or (4) after complete or partial replacement of a filter bank by:

~~**As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and f.~~

LA03

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99.95% DOP and 99% halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%**.

SR

See ITS 5.5.8

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 - 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 2.5% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement**, and

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration**.

d.1 In accordance with the Surveillance Frequency Control Program by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%**;

See ITS 5.5.15

d.2 In accordance with the Surveillance Frequency Control Program, test the supply fans (trains A and B) and measure CRE pressure relative to external areas adjacent to the CRE boundary.**

SR

e. In accordance with the Surveillance Frequency Control Program by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation,

SR

f. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.~~**~~

~~**As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and f.~~

LA03

DISCUSSION OF CHANGES
ITS 3.7.10, CREVS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS).

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – – Removal of LCO, SR, or other TS requirement to the TS Bases, TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 3.7.5 lists the components, that comprise the Control Room Emergency Ventilation System (CREVS), required to be OPERABLE in the Limiting Condition for Operation (LCO). ITS 3.7.10 does not list the components in the LCO. This changes the CTS by removing the specific components that comprise the CREVS from the Technical Specifications to the Technical Specification Bases.

The removal of CREVS components from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS requires the CREVS to be OPERABLE and lists the components in the Bases. This change is acceptable because this type of detail will be adequately controlled in the Technical Specification Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because CREVS components are being removed from the Technical Specifications.

LA02 *(Type 4 – – Removal of LCO, SR, or other TS requirement to the TS Bases, TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 3.7.5 ACTIONS a.4, a.6, and a.7

DISCUSSION OF CHANGES
ITS 3.7.10, CREVS

contain Actions for the recirculation damper, normal outside air intake damper, and the emergency outside air intake damper, respectively, when one is inoperable. The Actions require the specific damper to be placed in its open or closed position and CREVS to be placed in the recirculation mode of operation. ITS 3.7.10 ACTION B will require the same Actions but will not specify the position of the dampers.

The purpose of these Actions is to ensure the component is restored to OPERABLE status or placed in a mode it can support CREVS operating in a manner that perform its specified safety function. Listing the specific position of the damper in the Action is not necessary to be included in order to provide adequate protection of public health and safety. The specific position of dampers will be moved to the Technical Specification Bases. This change is acceptable because this type of detail will be adequately controlled in the Technical Specifications Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because CREVS components are being removed from the Technical Specifications.

LA03 *(Type 4 – – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program*) CTS Surveillance Requirements (SRs) 4.7.5.b, 4.7.5.c.1, 4.7.5.c.2, 4.7.5.c.3, 4.7.5.d.1, 4.7.5.d.2, and 4.7.5.f contain a footnote that require these SRs to be performed on the compensatory filtration unit. ITS 3.7.10 does not contain any requirements to perform SRs on the compensatory filtration unit. This changes the CTS by removing the requirements to perform SRs on the compensatory filtration unit out of Technical Specifications to the Technical Requirements Manual (TRM).

The compensatory filtration unit is not required by the CTS LCO for CREVS but is mentioned in the ACTIONS when the primary filtration unit is inoperable and mentioned in the SRs as a footnote. The ITS will only mention the compensatory filtration unit in the ACTIONS for the inoperable filtration unit. The compensatory filtration unit is a manual, safety-related, Seismic Class I backup to the installed system with the same functional and operational capabilities as the installed filter train. The unit is surveillance tested in accordance with the same requirements as those imposed on the installed filter train and will continue to be tested as such with the removal of the SR requirements to the TRM. The removal of these requirements to the TRM is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of the public health and safety. These requirements can be adequately controlled in the TRM and changes to the TRM are controlled via 10 CFR 50.59. The 10 CFR 50.59 program provides for the evaluation of changes to ensure SRs are properly controlled. This change is designated as a less restrictive removal of detail change because the compensatory filtration unit SRs are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS 3.7.10, CREVS

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.7.5, Action, identifies a degraded condition of the CREVS and provides specific completion times to restore the degraded condition or commence a unit shutdown. If a unit shutdown is required CTS 3.7.5 Action requires the unit be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.7.11 ACTION D states that if the Required Action and associated Completion Time associated with the CREVS degraded conditions are not met to be in MODE 3 in 6 hours and MODE 4 in 12 hours and is modified by a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. ITS 3.7.11 ACTION E states that if the Required Action and associated Completion Time associated with the CREVS degraded conditions are not met to be in MODE 3 in 12 hours and MODE 4 in 18 hours and is modified by a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. This changes the CTS by allowing a Required Action end state of HOT SHUTDOWN (MODE 4) rather than an end state of COLD SHUTDOWN (MODE 5).

One purpose of CTS 3.7.5, Action is to provide an end state, a condition that the reactor must be placed in, if the Required Actions allowing remedial measures to be taken in response to the degraded conditions with continued operation are not met. End states are usually defined based on placing the unit into a MODE or condition in which the Technical Specification LCO is not applicable. MODE 5 is the current end state for LCOs that are applicable in MODES 1 through 4. This change is acceptable because the risk of the transition from MODE 1 to MODES 4 or 5 depends on the availability of alternating current (AC) sources and the ability to remove decay heat such that remaining in MODE 4 may be safer. During the realignment from MODE 4 to MODE 5, there is an increased potential for loss of shutdown cooling and loss of inventory events. Decay heat removal following a loss-of-offsite power event in MODE 5 is dependent on AC power for shutdown cooling whereas, in MODE 4, the turbine driven auxiliary feedwater (AFW) pump will be available. Therefore, transitioning to MODE 5 is not always the appropriate end state from a risk perspective. Thus, for specific TS conditions, Westinghouse Topical Report WCAP-16294-A R1 (ADAMS Accession No. ML103430249) justifies MODE 4 as an acceptable alternate end state to Mode 5. The proposed change to the Technical Specifications will allow time to perform short-duration repairs, which currently necessitate exiting the original mode of applicability. The MODE 4 TS end state is applied, and risk is assessed and managed in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." This proposed change is consistent with NRC approved Technical Specification Task Force (TSTF) traveler TSTF-432-A Revision 1 (ADAMS Accession No. ML103360003), noticed for availability by the NRC in the Federal Register (77 FR 27814) on May 11, 2012. The NRC's approval of WCAP-16294-A included four limitations and conditions on its use as identified in Section 4.0 of the NRC Safety Evaluation associated with WCAP-16294-A. Implementation of these stipulations were addressed in the Bases of TSTF-432-A. Florida Power & Light implemented these limitations and conditions at PTN in the adoption of the associated TSTF-

DISCUSSION OF CHANGES
ITS 3.7.10, CREVS

432-A Bases. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.7.5 does not contain specific Actions for the Condition where two required features are inoperable requiring entry in to CTS LCO 3.0.3 requiring a simultaneous dual unit shut down. ISTS 3.7.10 includes Condition F for two CREVS trains inoperable in MODE 1, 2, 3, or 4 and requires immediate entry into LCO 3.0.3, a similar simultaneous dual unit shutdown. However, PTN is proposing to add a separate entry condition to ITS 3.7.11 Conditions D and E similarly requiring the immediate shut down of both units when two or more CREVS required redundant components are inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B or C (inoperable filter train or Control Room boundary) but allowing for a sequential unit shut down instead of a simultaneous dual unit shut down. This changes the CTS by allowing addition time to perform a sequential dual unit shut down.

The purpose of CTS LCO 3.0.3 is to establish the actions that must be implemented when an LCO is not met and: 1) An associated Required Action and Completion Time is not met and no other Condition applies; or 2) The condition of the unit is not specifically addressed by the associated ACTIONS. This change is acceptable because the Required Actions establish similar remedial measures as in CTS that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while allowing the orderly shutdown of the units without jeopardizing the stability of the electrical grid by imposing a dual unit shutdown. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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

1

CTS

3.7 PLANT SYSTEMS

3.7.5 3.7.10 Control Room Emergency ~~Filtration~~ Ventilation System (CRE~~F~~VS)

LCO 3.7.5 LCO 3.7.10 ~~Two~~ The CRE~~F~~VS ~~trains~~ shall be OPERABLE. 1

LCO 3.7.5 Footnote *

---NOTE---
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY APPLICABILITY: MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~,
During movement of ~~[recently]~~ irradiated fuel assemblies. 2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ~~One~~ CRE~~F~~VS ~~train~~ inoperable for reasons other than Condition B or C.	A.1	Restore CRE~~F~~VS ~~train~~ to OPERABLE status.	7 days
INSERT 1			
INSERT 2			
~~B~~C. ~~One or more~~ CRE~~F~~VS ~~trains~~ inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	~~B~~C.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	~~B~~C.2	Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	AND		
	~~B~~C.3	Restore CRE boundary to OPERABLE status.	90 days

ACTIONs a.1, a.3, a.4, a.6, & a.7 — 1, 3
ACTION b — 3, 1, 3

CTS

INSERT 1 3

ACTION a.4
ACTION a.6
ACTION a.7

OR		
A.2.1	--------------NOTE-------------- Only applicable if CREVS inoperable due to an inoperable recirculation damper, normal outside air intake damper, or emergency outside air intake damper. ------------------------------------	
	Place the inoperable damper or the redundant damper in its emergency position.	7 days
AND		
A.2.2	--------------NOTE-------------- Only applicable if CREVS inoperable due to an inoperable recirculation damper or normal outside air intake damper. ------------------------------------	
	Place CREVS in the recirculation mode.	7 days

INSERT 2 3

ACTION a.5

B. CREVS inoperable due to inoperable Filter train.	B.1	--------------NOTE-------------- Irradiated fuel movement may resume once compensatory filtration unit is placed in Service. ------------------------------------ Suspend movement of irradiated fuel.	Immediately
	AND		
	B.2	Place compensatory filtration unit in service.	24 hours
	AND		
	B.3	Restore filter train to OPERABLE status.	7 days

 1

---NOTE---
Condition D only applies to one Unit during a dual Unit shutdown.

ACTIONS (continued)

CONDITION		REQUIRED ACTION	COMPLETION TIME
~~C~~ D. Required Action and associated Completion Time of Condition A , B, or C ~~or B~~ not met in MODE 1, 2, 3, or 4. OR Two or more CREVS required redundant components inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B or C. INSERT 3	~~C~~ D.1	Be in MODE 3.	6 hours
	AND		
	~~C~~ D.2	---NOTE--- LCO 3.0.4.a is not applicable when entering MODE 4. --- Be in MODE 4.	12 hours
~~D~~ F. Required Action and associated Completion Time of Condition A or B not met ~~[~~in MODE 5 or 6, or~~]~~ during movement of irradiated fuel assemblies.	~~D~~ F.1	~~---NOTE---~~ ~~[Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.]~~ ~~---~~ ~~Place OPERABLE CREFS train in emergency mode.~~	~~Immediately~~
	~~OR~~		
	~~D.2~~	Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately

ACTIONs a.1, a.3, a.4, a.5, a.6, & a.7, b

DOC A02

ACTIONs a.3, a.4, a.6, & a.7

3 3 3 2 1 3 2

 1

CTS

INSERT 3

3

ACTIONs a.1, a.3, a.4, a.5, a.6, & a.7, b

E. ------------NOTES----------- 1. Condition E only applies when both Units are being shutdown simultaneously. 2. Only one Unit is allowed to enter Condition E during simultaneous shutdown of both Units. -------------------------------- Required Action and associated Completion Time of Condition A, B, or C not met. <u>OR</u> Two or more CREVS required redundant components inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B or C	E.1 Be in MODE 3. <u>AND</u> E.2 -------------NOTE---------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE 4.	12 hours 18 hours

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G ~~E~~. ~~Two CREFS trains inoperable [in MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies.~~ ~~OR~~ V ~~One or more~~ CRE~~F~~S ~~trains~~ inoperable due to an inoperable CRE boundary ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	G ~~E~~.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
~~F~~. ~~Two CREFS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B~~.	~~F.1~~ ~~Enter LCO 3.0.3.~~	~~Immediately~~

ACTION b

3 1 3 2 1 x

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	V Operate each CRE~~F~~S train for ≥ 15 continuous minutes ~~[with heaters operating]~~.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SR 4.7.5.b

2

1

SURVEILLANCE REQUIREMENTS (continued)

		SURVEILLANCE	FREQUENCY	
SR 4.7.5.c SR 4.7.5.d.1	SR 3.7.10.2	Perform required CRE~~F~~VS filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~	2
SR 4.7.5.e	SR 3.7.10.3	Verify each CRE~~F~~VS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
SR 4.7.5.f	SR 3.7.10.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program	

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.10, CREVS

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes made to reflect the current licensing basis. The Control Room Emergency Ventilation System (CREVS) Current Technical Specifications (CTS) for the Turkey Point Nuclear Generating Station (PTN) is being retained because of the uniqueness of the system. In addition, renumbering is required due to added Actions.

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

B 3.7 PLANT SYSTEMS

Ventilation V

B 3.7.10 Control Room Emergency ~~Filtration~~ System (CRE~~F~~S) 1

BASES

V

BACKGROUND The CRE~~F~~S provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. 1

INSERT 1

~~The CREFS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREFS train consists of a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provides backup in case of failure of the main HEPA filter bank.~~ 1

~~The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.~~

~~The CREFS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Both the demister and heater are important to the effectiveness of the charcoal adsorbers.~~

INSERT 1 (B 3.7.10 Background) 1

The CREVS consists of the following components:

a. Air Handling Units (AHUs)
b. Condensing (air conditioning (A/C)) Units
c. Recirculation Fans
d. Recirculation Dampers,
e. Recirculation Filter unit,
f. Normal Outside Air Intake Dampers,
g. Emergency Outside Air Intake Dampers.

The air handling units and condensing units are part of the overall system but not included in the OPERABILITY requirements for CREVS. The OPERABILITY of the air handling units and condensing units are covered in LCO 3.7.11, "Control Room Emergency Air Temperature Control System (CREATCS)."

Units 3 and 4 share a common control room envelope (CRE). The CRE is the area within the confines of the CRE boundary that contains the spaces that Control Room occupants inhabit to control the units during normal and accident conditions. This area encompasses the Control Room, including the Control Room offices, rack area, kitchen, and lavatory, and mechanical equipment room located below the Control Room which contains the CREVS equipment. The CRE is protected during normal operation, natural events, and accident conditions. The CRE Boundary is the combination of walls, floor, roof, ducting, doors, penetrations, and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the radiological dose consequence analyses and that CRE occupants are protected from hazardous chemicals and smoke. The CRE and its boundary are defined in Control Room Envelope Habitability Program.

During normal operation, fresh makeup air is admitted to this system through an intake louver and two dampers in series located in the west wall of the Control Building. This system maintains a positive pressure in the control room envelope greater than that in the cable spreading room in order to prevent smoke from a hypothesized fire in the cable spreading room from entering the control room. All control room penetrations are designed for leak tightness standards, including doors per NFPA 80. Since the control room is maintained at slightly more than atmospheric pressure, the infiltration of contaminated air into the control room is negligible.

Two radiation monitors located in the normal air intake ducting continuously monitor for radiation in the incoming air. In the unlikely event of a maximum hypothetical accident (MHA), the Control Room ventilation will automatically be placed in a recirculation mode as described below.

1 INSERT 1 (B 3.7.10 Background)
(continued)

The control room emergency mode is initiated by a containment Phase A signal, a high radiation signal from the containment air Radiation Monitors (R-11 and R-12), the manual initiation from a test switch, or a high radiation signal from the redundant monitors in the control room normal air intake. Following initiation, all exhaust fans are shut off, and the redundant exhaust isolation dampers in series are closed. Redundant normal air intake isolation dampers in series are closed. Redundant emergency air intake dampers in parallel are opened. Likewise, the recirculation air path is opened. A single air supply fan is energized to move the appropriate mixture of recirculation control room air and new outdoor air through the HEPA and charcoal filter system.

BASES

BACKGROUND (continued)

~~Actuation of the CREFS places the system in either of two separate states (emergency radiation state or toxic gas isolation state) of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the emergency radiation state of the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA and the charcoal filters. The emergency radiation state also initiates pressurization and filtered ventilation of the air supply to the CRE.~~ 1

~~Outside air is filtered, diluted with building air from the electrical equipment and cable spreading rooms, and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. The actions taken in the toxic gas isolation state are the same, except that the signal switches the CREFS to an isolation alignment to minimize any outside air from entering the CRE through the CRE boundary.~~

~~The air entering the CRE is continuously monitored by radiation and toxic gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required. The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.~~

~~A single CREFS train operating at a flow rate of < [3000] cfm will pressurize the CRE to about [0.125] inches water gauge relative to external areas adjacent to the CRE boundary. The CREFS operation in maintaining the CRE habitable is discussed in the FSAR, Section [9.4] (Ref. 1).~~

~~Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS is designed in accordance with Seismic Category I requirements.~~

~~The CREFS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a DBA without exceeding a [5 rem whole body dose or its equivalent to any part of the body] [5 rem total effective does equivalent (TEDE)].~~

BASES

APPLICABLE SAFETY ANALYSES

~~The CREFS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREFS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the FSAR, Chapter [15] (Ref. 2).~~

INSERT 2 1

~~The CREFS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 3). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 4).~~

The worst case single active failure of a component of the CRE~~F~~S, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. 1

V

The CRE~~F~~S satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

~~Two independent and redundant CREFS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of [5 rem whole body or its equivalent to any part of the body] [5 rem TEDE] to the CRE occupants in the event of a large radioactive release.~~ 1

INSERT 3

~~Each CREFS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREFS train is OPERABLE when the associated:~~

~~a. Fan is OPERABLE,~~

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

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

~~In order for the CREFS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.~~

INSERT 2 (B 3.7.10 ASA) 1

The location of CREVS components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. CREVS provides airborne radiological protection for the CRE occupants, as demonstrated by radiological dose consequence analyses for the most limiting DBA presented in UFSAR Chapter 14. CREVS also provides protection from chemical hazards and smoke hazards. CREVS pressurizes the CRE relative to external areas adjacent to the CRE boundary. The analysis of hazardous chemical releases for NUREG-0737 Item III.D.3.4, "Control Room Habitability Requirement," and the subsequent reanalysis included in PC/M 06-004, "Addition of Unit 5 to the Turkey Point Site," for new chemical release hazards demonstrate that the toxicity limits of Regulatory Guide 1.78 are not exceeded in the CRE following a hazardous chemical release. Thus, neither automatic nor manual actuation of CREVS is required for an analyzed hazardous chemical release. Analysis of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactors either from the Control Room or alternate shutdown panels.

The Alternative Source Term radiological analyses assume both emergency outside air intake flow paths are available with parallel dampers ensuring outside makeup air can be drawn through both intake locations during a design basis accident and a single active failure. These analyses rely on a provision in Regulatory Guide 1.194 Section 3.3.2 that allows a reduction in the atmospheric dispersion factors (X/Qs) for dual intake arrangements with balanced flow rates to one half of the more limiting X/Q value provided the two intakes are not within the same wind direction window for each release / receptor location.

INSERT 3 (B 3.7.10 LCO) 1

The OPERABILITY of the CREVS ensures that the Control Room envelope (CRE) will remain habitable for occupants during and following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The OPERABILITY of this system in conjunction with Control Room design provisions is based on limiting the radiation exposure to personnel occupying the CRE to 5 rem Total Effective Dose Equivalent (TEDE) for the duration of the accident. The radiological limits are consistent with the requirements of 10 CFR Part 50.67. CRE occupants are protected from chemical hazards in accordance with the limits of Regulatory Guide 1.78.

The following CREVS components are required to be OPERABLE.

a. Two Recirculation Fans,
b. Two Recirculation Dampers,

INSERT 3 (B 3.7.10 LCO) 1
(Continued)

c. One Recirculation Filter train,
d. Two Normal Outside Air Intake Dampers, and
e. Two Emergency Outside Air Intake Dampers.

The recirculation filter train includes the recirculation filter unit, two recirculation fans, and two recirculation dampers. An OPERABLE recirculation filter train requires the recirculation filter unit, one recirculation fan, and one recirculation damper to be OPERABLE.

BASES

LCO (continued)

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, 4, ~~[~~5, and 6,~~]~~ and during movement of ~~[recently]~~ irradiated fuel assemblies, the CRE~~F~~VS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA. 2

In ~~[~~MODES 5 and 6~~]~~, the CRE~~F~~VS is required to cope with the release from the rupture of a~~n~~ ~~outside~~ waste gas decay tank. 2

During movement of ~~[recently]~~ irradiated fuel assemblies, the CRE~~F~~VS must be OPERABLE or an inoperable filter train replaced by the compensatory filter unit to cope with the release from a fuel handling accident ~~[~~involving handling ~~recently~~ irradiated fuel~~]. [The CREFS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days), due to radioactive decay.]~~ 2 2

ACTIONS

A.1, A.2.1, and A.2.2 4

INSERT 4

~~When one CREFS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREFS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREFS train could result in loss of CREFS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.~~ 4

~~B.1, B.2, and B.3~~

~~If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to [5 rem whole body or its equivalent to~~ 4

INSERT 4 (B 3.7.10 ACTION A) 4

When one CREVS component inoperable, for reasons other than an inoperable CRE boundary or filtration train, action must be taken to restore the OPERABLE status. The LCO actions allow inoperability of the redundant active CREVS components (one recirculation fan, one recirculation damper, one normal outside air intake damper, and/or one emergency outside air intake damper) for a period of up to 7 days based on the low probability of occurrence of a Design Basis Accident (DBA) challenging the Control Room Habitability during this time period and the continued capability of the remaining OPERABLE system components to perform the required CREVS safety function. The kitchen and toilet area exhaust ventilation ducts have been permanently blocked off with seismic Class 1 solid plates coated in accordance with SPEC-C-004. The kitchen and toilet area motor-operated and gravity backdraft dampers are no longer credited for CREVS OPERABILITY.

B.1, B.2. and B.3

When the filter train is inoperable for reasons other than an inoperable CRE boundary, e.g., the filter is inoperable, and/or two recirculation fans are inoperable, and/or two recirculation dampers are inoperable, all movement of irradiated fuel is required to be immediately suspended and use of the Compensatory Filtration Unit is required to be immediately initiated and proper operation verified within 24 hours, ensuring Control Room occupant radiological exposures will not exceed limits. Within 7 days, the inoperable filter train is required to be restored to OPERABLE status. Consistent with 0-ADM-211 and NUREG-1431, “immediately” indicates that the required action should be pursued without delay and in a controlled manner, i.e., placing the Compensatory Filtration Unit into service should be initiated within approximately 1 hour. The 24 hour allowance to verify proper operation is reasonable based on the low probability of a DBA occurring during this time period. The 7 day Completion Time is reasonable based on the verification that the Compensatory Filtration Unit will continue to provide the CREVS safety function. As with the active components, this has the effect of temporarily suspending the single failure criterion for the affected components while assuring the continued functionality of the system. The 7 day Completion Time is also a reasonable time to diagnose, plan, repair, and test most problems with the inoperable filter train.

The compensatory filtration unit is designed as a manual, safety-related, Seismic Class I backup to the installed system with the same functional and operational capabilities as the installed filter train. The requirements for the compensatory filtration unit are located in the Technical Requirements Manual.

INSERT 4 (B 3.7.10 ACTIONS A, B, and C) 1
(Continued)

C.1, C.2, and C.3

In order for the CREVS to be considered OPERABLE, the CRE Boundary must be maintained such that the CRE occupant radiological dose does not exceed that calculated in the DBA Radiological Dose Consequence Analyses and CRE occupants are protected from hazardous chemicals and smoke. Since the CREVS and CRE are common to both units, the ACTION requirements are applicable to both units simultaneously, and must be applied according to each unit's operational MODE. If the CRE is inoperable Actions must be taken to restore an OPERABLE CRE boundary within 90 days

1

BASES

ACTIONS (continued)

~~any part of the body] [5 rem TEDE]), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.~~ 4

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

D.1 and D.2

~~C.1 and C.2~~ 4

Condition D is modified by a Note that Condition D only applies to one Unit during a dual Unit shutdown. Since both Units share the same Control Room, both units are affected by any inoperabilities. If both units are in an Operating Mode and a TS required shutdown were required due to CREVS, both units would be required to shut down. However, only one Unit could enter Condition D. The other Unit would have to enter Condition E.

CREVS

In MODE 1, 2, 3, or 4, if the inoperable ~~CREFS train~~ or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. 4 1 1

or multiple non-filter train redundant components are inoperable for reasons other than an inoperable CRE boundary,

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~5~~). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As

1 1

1

BASES

ACTIONS (continued)

1

stated in Reference ~~5~~, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective. 1

D.2

Required Action ~~C.2~~ is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. 4

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.

INSERT 5

4

~~D.1 and D.2~~ 4

F.1

~~[~~In MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, if the inoperable CRE~~F~~S ~~train~~ cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately ~~place the OPERABLE CREFS train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.~~ 2 1 1

V

~~An alternative to Required Action D.1 is to immediately~~ suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position. 4

~~[Required Action D.1 is modified by a Note indicating to place the system in the toxic gas protection mode if automatic transfer to the toxic gas protection mode is inoperable.]~~ 2

INSERT 5 (B 3.7.10 ACTION E) 4

E.1 and E.2

Condition E is modified by two notes. The first Note states that Condition E only applies when both Units are require to be shutdown simultaneously. The second Note states that only one Unit is allowed to enter Condition E during simultaneous shutdown of both Units. Since both Units share the same Control Room, both units are affected by any inoperabilities. If both units are in an Operating Mode and a TS required shutdown were required due to CREVS, both units would be required to shut down. However, only one Unit could enter Condition E. The other Unit would have to enter Condition D.

In MODE 1, 2, 3, or 4, if the inoperable CREVS or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, or multiple non-filter train redundant components are inoperable for reasons other than an inoperable CRE boundary, the unit must be placed in a MODE in which the overall plant risk is reduced. To achieve this status, one unit must be placed in at least MODE 3 within 12 hours, and in MODE 4 within 18 hours. The extra 6 hours is to allow for a dual unit shutdown such that the units can be shut down sequentially.

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 1). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 1, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action E.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

 1

BASES

ACTIONS (continued)

~~E.1~~ 4
G.1

~~[~~In MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, with ~~two~~ CRE~~F~~S ~~trains inoperable or with one or more CREVS trains~~ inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position. 2 1
V

~~F.1~~ 4

~~If both CREFS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition B), the CREFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.~~ 1 4

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Operation ~~[with the heaters on]~~ for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that ~~[heater failure,]~~ blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~[The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.~~ 2 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----REVIEWER'S NOTE-----~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3
~~--]~~

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.2

V

This SR verifies that the required CRE~~F~~S testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. The ~~[~~VFTP~~]~~ includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~. 2

SR 3.7.10.3

V

This SR verifies that each CRE~~F~~S train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency~~. [The Frequency of [18] months is based on industry operating experience and is consistent with the typical refueling cycle.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3
~~---]~~

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than ~~[~~5 rem whole body or its equivalent to any part of the body~~][~~5 rem TEDE] and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action ~~B.3~~ C.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. ~~6~~ 2) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. ~~7~~ 3). These compensatory measures may also be used as mitigating actions as required by Required Action ~~B.2~~ C.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. ~~8~~ 4). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

REFERENCES

~~1. FSAR, Section [9.4].~~

~~2. FSAR, Chapter [15].~~

~~3. FSAR, Section [6.4].~~

~~4. FSAR, Section [9.5]~~

~~5~~ 1. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

1

1

BASES

REFERENCES (continued)

2 ~~6~~. Regulatory Guide 1.196.

3 ~~7~~ NEI 99-03, "Control Room Habitability Assessment," June 2001.

4 ~~8~~. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability," (ADAMS Accession No. ML040300694).

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.10 BASES, CREVS

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.
2. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.
3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.
4. Changes are made to the Improved Technical Specifications (ITS) Bases to reflect changes made to the ITS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.10, CREVS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 11

ITS SECTION 3.7.11 – CONTROL ROOM EMERGENCY AIR TEMPERATURE CONTROL SYSTEM (CREATCS)

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

A01

PLANT SYSTEMS

Air Temperature Control System (CREATCS) A01

3/4.7.5 CONTROL ROOM EMERGENCY ~~VENTILATION SYSTEM~~

LIMITING CONDITION FOR OPERATION

See ITS 3.7.10

LCO 3.7.11 3.7.5 ~~The Control Room Emergency Ventilation System~~ shall be OPERABLE* ~~with:~~

CREATCS A02

~~a. Three air handling units,~~

~~b. Two condensing units,~~ LA01

See ITS 3.7.10

c. Two control room recirculation fans,

d. Two recirculation dampers,

e. One filter train,

f. Two isolation dampers in the normal outside air intake duct,

g. Two isolation dampers in the emergency outside air intake duct,

h. Control Room Envelope

Applicability APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

Action A, Action F, Action D, Action E, Action D & E

a.1 With one air handling unit inoperable, within 7 days, restore the inoperable air handling unit to OPERABLE status or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in ~~COLD SHUTDOWN within the following 30 hours~~.

Add proposed Required Actions D.2, E.2 and Notes L01

Action B, Action F, Action D, Action E, Action D & E

a.2 With only one OPERABLE condensing unit, within 30 days, restore at least one of the inoperable condensing units to OPERABLE status or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in ~~COLD SHUTDOWN within the following 30 hours~~.

Add proposed Required Actions D.2, E.2 and Notes L01

See ITS 3.7.10

a.3 With one recirculation fan inoperable, within 7 days, restore the inoperable fan to OPERABLE status or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

*The Control Room Envelope (CRE) boundary may be opened intermittently under administrative control.

Add proposed Condition C and Required Action C.1 L02

Add proposed Condition G and Required Action G.1 A03

PLANT SYSTEMS

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

a.4 With one recirculation damper inoperable, within 7 days, restore the inoperable damper to OPERABLE status or, place and maintain at least one of the recirculation dampers in the open position and place the system in recirculation mode or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.7.10

a.5 With the filter train inoperable, e.g., an inoperable filter, and/or two inoperable recirculation fans, and/or two inoperable recirculation dampers, immediately suspend all movement of irradiated fuel and immediately initiate action to place the compensatory filtration unit in service and verify proper operation within 24 hours, following which movement of irradiated fuel may resume, and within 7 days, restore the filter train to OPERABLE status.

With the above requirements not met, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

a.6 With an inoperable damper in the normal outside air intake, within 7 days, restore the inoperable damper to OPERABLE status or, place and maintain at least one of the normal outside air intake isolation dampers in the closed position and place the system in recirculation mode or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

a.7 With an inoperable damper in the emergency outside air intake, within 7 days, restore the inoperable damper to OPERABLE status or, place and maintain at least one of the emergency outside air intake isolation dampers in the open position or, immediately suspend all movement of irradiated fuel and be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

PLANT SYSTEMS

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

See ITS 3.7.10

b. With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary during MODES 1, 2, 3 or 4, immediately initiate action to implement mitigating actions. Within 24 hours, verify mitigating actions ensure CRE occupant radiological and chemical hazards will not exceed limits, and CRE occupants are protected from smoke hazards, and restore CRE boundary to OPERABLE status within 90 days.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours for one Unit, or 12 hours for both Units, and in COLD SHUTDOWN within the following 30 hours.

With the Control Room Emergency Ventilation System inoperable due to an inoperable CRE boundary during MODES 5, 6 or during the movement of irradiated fuel assemblies, immediately suspend all movement of irradiated fuel.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1 4.7.5 The Control Room Emergency Ventilation System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 120°F;

See ITS 3.7.10

b. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes**;

See ITS 5.5.8

c. In accordance with the Surveillance Frequency Control Program or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system, or (4) after complete or partial replacement of a filter bank by:

See ITS 3.7.10

**As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and f.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

See ITS 5.5.8

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99.95% DOP and 99% halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%**.

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 - 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 2.5% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement**, and

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration**.

d.1 In accordance with the Surveillance Frequency Control Program by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%**;

See ITS 5.5.15

d.2 In accordance with the Surveillance Frequency Control Program, test the supply fans (trains A and B) and measure CRE pressure relative to external areas adjacent to the CRE boundary.**

See ITS 3.7.10

e. In accordance with the Surveillance Frequency Control Program by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation,

f. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.**

**As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and f.

DISCUSSION OF CHANGES
ITS 3.7.11, CREACTS

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications - Westinghouse Plants" (ISTS).

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

A02 CTS 3.7.5 requires, in part, that the Control Room Emergency Ventilation System shall be OPERABLE* with: a. Three air handling units, and b. Two condensing units. ITS Limiting Condition for Operation (LCO) 3.7.11 states that the Control Room Emergency Air Temperature Control System (CREATCS) shall be OPERABLE. This changes the CTS by providing a separate LCO for the CREATCS.

CTS 3.7.5 includes requirements for both the Control Room Emergency Ventilation System (CREVS) and for the CREATCS. ITS 3.7.10 contains the requirements for the CREVS while ITS 3.7.11 contains the requirements for the CREATCS. CTS 3.7.5, CREVS, requires that three air handling units are OPERABLE; this requirement for three air handling units to be OPERABLE is retained in ITS LCO 3.7.11. In CTS 3.7.5 two condensing units are required to be OPERABLE for control room temperature and humidity control; this requirement is retained in ITS 3.7.11. Because an air handling unit and a condensing unit work together to form a control room air conditioning unit these components are being placed in the same LCO.

A03 CTS 3.7.5 does not provide any actions to take if three air handling units or two required condensing units are inoperable. ITS 3.7.11 Condition G provides Required Action G.1 if three air handling units or two required condensing units are inoperable. This changes the CTS by explicitly providing action to be taken if three air handling units or two required condensing units are inoperable.

CTS 3.7.5 does not provide any actions to be taken if three air handling units or two required condensing units are inoperable. CTS 3.0.3 states, in part, that when a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit, as applicable, in the listed conditions within the specified times. ITS 3.7.11 Condition G states that if three air handling units or two required condensing units are inoperable to immediately enter LCO 3.0.3, providing similar direction and in CTS. This change is designated as Administrative because no technical changes are being made to the CTS.

DISCUSSION OF CHANGES
ITS 3.7.11, CREACTS

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – – Removal of LCO, SR, or other TS requirement to the TS Bases, TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 3.7.5 lists the components that comprise the CREATCS which are required to be OPERABLE in the LCO. ITS 3.7.11 does not list the components in the LCO. This changes the CTS by removing the specific components that comprise the CREATCS from the Technical Specifications to the Technical Specification Bases.

The removal of CREATCS components from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS requires the CREATCS to be OPERABLE and lists the components in the Bases. This change is acceptable because this type of detail will be adequately controlled in the Technical Specification Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because CREATCS components are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (*Category 4 – Relaxation of Required Action*) CTS 3.7.5, Action, identifies a degraded condition of the CREVS and provides specific completion times to restore the degraded condition or commence a unit shutdown. If a unit shutdown is required, CTS 3.7.5 Action requires the unit be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.7.11 ACTION D states that if the Required Action and associated Completion Time of CREATCS degraded conditions are not met to be in MODE 3 in 6 hours and MODE 4 in 12 hours and is modified by a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. ITS 3.7.11 ACTION E states that if the Required Action and associated Completion Time of CREATC degraded conditions are not met to be in MODE 3 in 12 hours and MODE 4 in 18 hours and is modified by a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. This changes the CTS by allowing a Required Action end state of HOT SHUTDOWN (MODE 4) rather than an end state of COLD SHUTDOWN (MODE 5).

One purpose of CTS 3.7.5, Action is to provide an end state, a condition that the reactor must be placed in, if the Required Actions allowing remedial measures to be taken in response to the degraded conditions with continued operation are not met. End states are usually defined based on placing the unit into a MODE or condition in which the Technical Specification LCO is not applicable. MODE 5 is the current end state for LCOs that are applicable in MODES 1 through 4. This change is acceptable because the risk of the transition from MODE 1 to MODES 4 or 5 depends on the availability of alternating current (AC) sources and the ability to remove decay heat such that remaining in MODE 4 may be safer. During the realignment from MODE 4 to MODE 5, there is an increased potential for loss of shutdown cooling and loss of inventory events. Decay heat removal following a loss-of-offsite power event in MODE 5 is dependent on AC power for shutdown cooling whereas, in MODE 4, the turbine driven auxiliary feedwater (AFW) pump will be available. Therefore, transitioning to MODE 5 is not always the appropriate end state from a risk perspective. Thus, for specific TS conditions, Westinghouse Topical Report WCAP-16294-A R1 (ADAMS Accession No. ML103430249) justifies MODE 4 as an acceptable alternate end state to Mode 5. The proposed change to the Technical Specifications will allow time to perform short-duration repairs, which currently necessitate exiting the original mode of applicability. The MODE 4 TS end state is applied, and risk is assessed and managed in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." This proposed change is consistent with NRC approved Technical Specification Task Force (TSTF) traveler TSTF-432-A Revision 1 (ADAMS Accession No. ML103360003), noticed for availability by the NRC in the Federal Register (77 FR 27814) on May 11, 2012. The NRC's approval of WCAP-16294-A included four limitations and conditions on its use as identified in Section 4.0 of the NRC Safety Evaluation associated with WCAP-16294-A. Implementation of these stipulations were addressed in the Bases of TSTF-432-A. Florida Power & Light implemented these limitations and conditions at PTN in the adoption of the associated TSTF-432-A Bases. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.7.5 Action a.1 provides actions to take when one of three air handling units is inoperable. CTS 3.7.5 does not contain any actions to be taken if two of three air handling units are inoperable. ITS 3.7.11 Condition C provides Required Action C.1 if two air handling units are inoperable. This changes the CTS by explicitly providing action to be taken if two air handling units are inoperable.

CTS 3.7.5 does not provide any actions to be taken if two air handling units are inoperable. CTS 3.0.3 states, in part, that when a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit, as

DISCUSSION OF CHANGES
ITS 3.7.11, CREACTS

applicable, in: a. At least HOT STANDBY within the next 6 hours. CTS 3.0.3 does not include the extended Completion Time of 12 hours be in HOT STANDBY (MODE 3) if a dual units shut down is required, which this Condition would require. By providing a separate Condition in ITS 3.7.11 for when two air handling units are inoperable allow the units to be shut down sequentially using Condition D for the first unit's shutdown requirements and Condition E for the second unit's shut down requirements. Condition E allows the second unit 12 hours to be in MODE 3. This change is acceptable because the Required Actions establish similar remedial measures as in CTS that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while allowing the orderly shutdown of the units without jeopardizing the stability of the electrical grid by imposing a dual unit shutdown. This change is designated as less restrictive because less stringent Required Actions are being applied in the S than were applied in the CTS.

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

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.11 ~~Two~~ CREATCS ~~trains~~ shall be OPERABLE. 3

APPLICABILITY: MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~,
During movement of ~~[recently]~~ irradiated fuel assemblies. 1

INSERT 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.1 A. ~~One~~ CREATCS ~~train~~ inoperable due to one inoperable air handling unit. INSERT 2	A.1 Restore CREATCS ~~train~~ to OPERABLE status.	7 ~~30~~ days 3
Action a.1, Action a.2 D ~~B~~. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4. --------NOTE-------- Condition D only applies to one Unit during a dual unit shutdown. ----------------	D ~~B~~.1 Be in MODE 3. AND D ~~B~~.2 --------NOTE-------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------------- Be in MODE 4.	6 hours 12 hours 2
INSERT 3		2
Action a.1, Action a.2 F ~~C~~. Required Action and associated Completion Time of Condition A, B, or C not met ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	F ~~C~~.1 ~~Place OPERABLE CREATCS train in operation.~~ ~~OR~~ ~~C.2~~ Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately ~~Immediately~~ 2 1 1 1

CTS

3 **INSERT 1**

ACTION a.1,
ACTION a.2

---NOTE---

Actions apply to both units simultaneously

----

3 **INSERT 2**

ACTION a.2	B. CREATCS inoperable due to one required condensing unit inoperable.	B.1	Restore CREATCS to OPERABLE status.	30 days
DOC L02	C. CREATCS inoperable due to two inoperable air handling units.	C.1	Restore one inoperable air handling unit to OPERABLE status.	1 hour

3

INSERT 3

E. -----------NOTES------------ 1. Condition C only applies when both Units are being shutdown simultaneously. 2. Only one Unit is allowed to enter Condition C during simultaneous shutdown of both Units. --------------------------- Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4.	E.1 Be in MODE 3. <u>AND</u> E.2 -------------NOTE---------------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------------- Be in MODE 4.	12 hours 18 hours

ACTION a.1,
ACTION a.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D. Two CREATCS trains inoperable [in MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies.~~	~~D.1 Suspend movement of [recently] irradiated fuel assemblies.~~	~~Immediately~~
DOC A03 ~~E~~ G. ~~Two CREATCS trains inoperable in MODE 1, 2, 3, or 4.~~ CREATCS inoperable due to three air handling units or two required condensing units inoperable in MODE 1, 2, 3, or 4.	~~E~~ G.1 Enter LCO 3.0.3.	Immediately

3
2
3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.7.5.a SR 3.7.11.1 Verify ~~each CREATCS train has the capability to remove the assumed heat load.~~ control room air temperature is ≤ 120°F.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2
1

3

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.11, CREATCS

1. The Improved Standard Technical Specification (ISTS) contain bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes made to reflect the current licensing basis. In addition, renumbering is required due to added Actions.

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

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

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of ~~two~~ three independent and redundant ~~trains~~ air conditioning units that provide cooling ~~and heating~~ of recirculated control room air. ~~Each train consists~~ INSERT 1 ~~of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control.~~ The CREATCS is a subsystem of CREVS providing air temperature control for the control room. 1

The CREATCS is an emergency system, parts of which ~~may~~ also operates during normal unit operations. A single ~~train~~ air conditioning unit will provide the required temperature control to maintain the control room ~~between [70]° and [85]°~~ ≤ 120ºF. The CREATCS operation in maintaining the control room temperature is discussed in the UFSAR, Section ~~[6.4]~~ 9.9 (Ref. 1). 1 2

APPLICABLE SAFETY ANALYSES

The design basis of the CREATCS is to maintain the control room temperature for 30 days of continuous occupancy. INSERT 2

The CREATCS components are arranged in redundant, safety related ~~trains~~ units. During emergency operation, the CREATCS maintains the temperature ~~between [70]° and [85]°~~ ≤ 120ºF. A single active failure of a component of the CREATCS, with 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 CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY. 1 2

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

LCO

~~Two independent and redundant trains~~ Three air handling units and two condensing units of the CREATCS are required to be OPERABLE to ensure that at least ~~one is~~ two air handling units and one condensing unit are available, assuming a single failure disabling ~~the other train~~ one. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident. 1

1 **INSERT 1**

The control room is maintained at the personnel comfort level of (70 + 10)°F. Protective equipment inside the room is designed to operate within design tolerance over this temperature range and will perform its protective function in an ambient of 120°F and 95% relative humidity (i.e., there will be no loss-of-function in an ambient temperature of 120°F).

Each air conditioning unit consists of a condensing unit, an air handling unit, instrumentation, and controls to provide for control room temperature control. All three control room air condition (CRAC) units (air handling unit and condensing unit) are powered by swing power sources, each of which can be powered by the emergency diesel generators. One CRAC unit is powered by MCC 3D, one unit by MCC 4D, and the third unit is powered via a transfer switch which automatically transfers between MCCs 3B and 4B. This configuration precludes the loss of more than one CRAC unit for any postulated single failure. Control room equipment is designed to operate in an environment of 120°F and 95% relative humidity. If two of three units were inoperative, the third would maintain the environment within these limits.

1 **INSERT 2**

In addition, as a subsystem of the Control Room Ventilation System, the CREATCS supports the CREVS function to provide a control room from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents by use of the air handling units to circulate air and pressurize the control room.

BASES

LCO (continued)

The CREATCS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE ~~in both trains~~. These components include the ~~heating and~~ cooling coils and associated temperature control instrumentation. In addition, the CREATCS must be operable to the extent that air circulation can be maintained. 1

APPLICABILITY

In MODES 1, 2, 3, 4, ~~[~~5, and 6, ~~]~~ and during movement of ~~[recently]~~ irradiated fuel assemblies, the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room ~~. [The CREATCS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days), due to radioactive decay.]~~ 2

and recirculation and pressurization of the control room air.

2

~~[In MODE 5 or 6,] CREATCS may not be required for those facilities that do not require automatic control room isolation.~~ 2 1

ACTIONS

A.1

due to inoperable air handling unit

With ~~one~~ CREATCS ~~train~~ inoperable, action must be taken to restore OPERABLE status within ~~30~~ 7 days. In this Condition, the remaining OPERABLE CREATCS ~~train is~~ units are adequate to maintain the control room temperature and occupant's dose within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CREATCS ~~train~~ air handling unit could result in loss of CREATCS function. The ~~30~~ 7 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available. 1

INSERT 3

~~B~~ D.1 and ~~B~~ D.2 4

In MODE 1, 2, 3, or 4, if the inoperable CREATCS ~~train~~ cannot be restored to OPERABLE status within the required Completion Time, the unit s must be placed in a MODE in which the overall plant risk is reduced. To achieve this status, the first unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. 1 1

Condition D is modified by a Note that Condition D only applies to one Unit during a dual Unit shutdown. Since both Units share the same Control Room, both units are affected by any inoperabilities. If both units are in an Operating Mode and a shutdown were required due to CREATCS, both units would be required to shut down. However, only one Unit could enter Condition D. The second Unit would enter Condition E.

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As 1

1 **INSERT 3**

B.1

With CREATCS inoperable due to an inoperable required condensing unit, action must be taken to restore OPERABLE status immediately. In this Condition, the remaining OPERABLE CREATCS components are adequate to maintain the control room temperature and occupant doses within limits. However, the overall reliability is reduced because a condensing unit is in a degraded condition. Taking immediate action is based on the low probability of an event requiring control room isolation, the consideration that the remaining components can provide the required protection considering a single failure, and that alternate safety or nonsafety related cooling means are available.

C.1

With CREATCS inoperable due to two inoperable air handling units, action must be taken to restore one inoperable air handling unit to OPERABLE status within 1 hour. In this Condition, the remaining OPERABLE CREATCS air handling unit is not adequate to maintain the control room temperature within limits, air circulation, and control room pressurization. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions.

BASES

ACTIONS (continued)

stated in Reference 2, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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.

Insert 4

F.1 ~~C.1 and C.2~~ 4

~~[~~In MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel, if the inoperable CREATCS ~~train~~ cannot be restored to OPERABLE status within the required Completion Time, ~~the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.~~ 2 1

~~An alternative to Required Action C.1 is to immediately~~ suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position. 4

1 **INSERT 4**

E.1 and E.2

Condition E is modified by two notes. The first Note states that Condition E only applies when both Units are being shutdown simultaneously. The second Note states that only one Unit is allowed to enter Condition E during simultaneous shutdown of both Units. Since both Units share the same Control Room, both units are affected by any inoperabilities. If both units are in an Operating Mode and a TS required shutdown were required due to CREATCS, both units would be required to shut down. However, only one Unit could enter Condition E. The other Unit would have to enter Condition D.

In MODE 1, 2, 3, or 4, if the inoperable CREATCS is not restored to OPERABLE status within the required Completion Time the units must be placed in a MODE in which the overall plant risk is reduced. To achieve this status, the second unit must be placed in at least MODE 3 within 12 hours, and in MODE 4 within 18 hours. The extra 6 hours is to facilitate a sequential dual unit shutdown.

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 2, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action E.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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.

BASES

ACTIONS (continued)

~~D.1~~

~~[In MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies, with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.~~ 4

~~E.~~1 4

If CREATCS is inoperable due to three air handling units or two required condensing units.

~~If both CREATCS trains are~~ inoperable in MODE 1, 2, 3, or 4, the control room CREATCS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately. 1

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

control room air temperature is ≤ 120°F. This ensures that the temperature in the control room is below the design temperatures for equipment in the control room.

This SR verifies that the ~~heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses] in the control room. This SR consists of a combination of testing and calculations.~~ 1 ~~[The [18] month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3
~~---]~~

REFERENCES

1. U FSAR, Section ~~[6.4].~~ 9.9 1 2

2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.11 BASES, CREATCS

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specification (ISTS) Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.
2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.
3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.
4. Changes are made to the Improved Technical Specifications (ITS) Bases to reflect changes made to the ITS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.11, CREATCS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 12

ITS 3.7.12, STORAGE POOL WATER LEVEL

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

A01

REFUELING OPERATIONS

3/4.9.11 WATER LEVEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

LCO 3.7.12 3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10" the spent fuel storage pool.*

During Movement of irradiated fuel assemblies in the fuel storage pool.

Applicability APPLICABILITY: ~~Whenever irradiated fuel assemblies are in the storage pool.~~ L01

ACTION:

irradiated L01

ACTION A a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies ~~and crane operations with loads in the fuel storage areas.~~ A02 L02

immediately

ACTION A.1 Note b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1 4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program ~~when irradiated fuel assemblies are in the fuel storage pool.~~ L01

~~3/4.9.12 DELETED~~

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

A02 CTS 3.9.11 ACTION states, in part, that with the requirements of the Specification not satisfied, to suspend all movement of fuel assemblies. ITS 3.7.12 Required Action A.1 requires the immediate suspension of movement of irradiated fuel assemblies in the spent fuel pool. This changes the CTS by explicitly specifying that the compensatory action to suspend all movement of fuel assemblies requires an immediate response, not to preclude movement of a fuel assembly to a safe position.

The purpose of the CTS 3.9.11 ACTION a is to preclude a fuel handling accident from occurring. The current action does not specify a time; however, it implies that the action is immediate. This change is acceptable because it only provides clarification that the compensatory action requires an immediate response. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.9.11 Applicability states "Whenever irradiated fuel assemblies are in the spent fuel pool." CTS Surveillance Requirement (SR) 4.9.11 also contains a statement that it is required to be performed when, "irradiated fuel assemblies are in the fuel storage pool." CTS ACTION a requires suspension of "movement of fuel assemblies" when the Limiting Condition for Operation (LCO) is not satisfied. ITS 3.7.12 LCO is applicable "During movement of irradiated fuel assemblies in the spent fuel pool." In addition, ITS ACTION A requires the suspension of "movement of <u>irradiated</u> fuel assemblies" when the minimum water level requirements cannot be met. This changes the CTS by relaxing the Applicability and performance of the SR to only during movement of irradiated fuel assemblies, as well as relaxing the Action to suspending movement of <u>irradiated</u> fuel assemblies. These changes are consistent with the assumptions of the fuel handling accident analysis.

The purpose of CTS 3.9.11 is to ensure that sufficient shielding will be available during fuel movement and for removal of iodine in the event of a fuel handling accident. The spent fuel minimum water depth is consistent with the assumptions of the safety analysis. This change is acceptable because the proposed changes are consistent with the fuel handling accident assumptions. The proposed changes will continue to ensure that the conditions assumed in the safety analyses and licensing basis are maintained. This change is designated as less restrictive because less stringent Technical Specification requirements are allowed in the ITS than are allowed in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.9.11 ACTION a states, in part, that when the spent fuel pool water level is not met, suspend all crane operations with loads in the fuel storage areas. ITS 3.7.12 Required Action A.1 states that when spent fuel pool water level is not within limits, immediately suspend movement of irradiated fuel assemblies in the spent fuel pool. This changes the CTS by deleting the requirements to suspend crane operations over the spent fuel storage areas.

The purpose of the CTS 3.9.11 ACTION is to preclude a fuel handling accident from occurring; however, the fuel handling accident assumes a spent fuel assembly is dropped and not loads carried by the crane. In 2011, the spent fuel cask handling crane was upgraded to the single-failure-proof criteria of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The NRC approved the use of the upgraded cask crane as single-failure-proof in License Amendments 243 and 239 which authorized the deletion of the cask drop accident from the safety analysis. Therefore, a fuel handling accident is only assumed to occur when an irradiated fuel assembly is being moved over the fuel storage pool. ITS 3.7.12 ACTION A requires suspending movement of irradiated fuel. Therefore, requiring the suspension of crane operations over the fuel storage pool is not required per Technical Specifications; however, PTN has administrative controls in place to prevent handling heavy loads over the fuel storage pool. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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

3.7 PLANT SYSTEMS

3.7.~~15~~ 12 Fuel Storage Pool Water Level 1 2

3.9.11 LCO 3.7.~~15~~ 12 The fuel storage pool water level shall be ≥ elevation 56' - 10" ~~23 ft over the top of irradiated fuel assemblies seated in the storage racks~~. 1 2

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

ACTIONS

ACTION a
ACTION b
Note

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. 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 fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SR 4.9.11

SURVEILLANCE		FREQUENCY
SR 3.7.~~15~~ 12. 1	Verify the fuel storage pool water level is ≥ elevation 56' – 10" ~~23 ft above the top of the irradiated fuel assemblies seated in the storage racks~~.	~~[7 days~~ 1 2 3 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.12, STORAGE POOL WATER LEVEL

1. Improved Standard Technical Specification (ISTS) 3.7.15 has been renumbered as Improved Technical Specifications (ITS) 3.7.12. Additionally, the title "Fuel Storage Pool Water Level" has been changed to "Storage Pool Water Level."

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

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

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

1

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Water Level

2

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

2

A general description of the fuel storage pool design is given in the U FSAR, Section ~~[9.1.2]~~ 9.5.2 (Ref. 1). A description of the Spent Fuel Pool Cooling ~~and Cleanup~~ System is given in the U FSAR, Section ~~[9.1.3]~~ 9.5.2 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section ~~[15.7.4]~~ 14.2.1 (Ref. 3).

2 2 3

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.~~25~~ 183 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR ~~100~~ 50.67 (Ref. 5) limits.

2

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. ~~In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.~~ The requirement to maintain ≥ elevation 56' 10" ensure these assumptions are met.

2

The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool. The requirement to maintain ≥ elevation 56' 10" ensure these assumptions are met.

2

2 1

BASES

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~15~~12.1 1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. ~~[The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------REVIEWER'S NOTE-----------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~--]~~

During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

BASES

REFERENCES

1. U FSAR, Section ~~[~~9.~~1~~ 5.2~~]~~.
2. U FSAR, Section ~~[~~9.~~1~~ 5.3~~]~~.
3. U FSAR, Section ~~[15.7.4]~~ 14.2.1.
4. Regulatory Guide 1.~~25~~ 183, ~~[~~Rev. 0~~]~~.
5. 10 CFR ~~100.11~~ 50.67.

(2) (3) (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.12 BASES, STORAGE POOL WATER LEVEL

1. Improved Standard Technical Specification (ISTS) B 3.7.15, "Fuel Storage Pool Water Level" has been renumbered as Improved Technical Specifications (ITS) B 3.7.12, "Storage Pool Water Level."
2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
3. The ISTS Bases contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.
4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.12, STORAGE POOL WATER LEVEL

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 13

ITS 3.7.13, FUEL STORAGE POOL BORON CONCENTRATION

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

REFUELING OPERATIONS

3/4.9.14 ~~SPENT FUEL STORAGE~~ Fuel Storage Pool Boron Concentration

A01

LIMITING CONDITION FOR OPERATION

LCO 3.7.13 3.9.14 The following conditions shall apply to spent fuel storage:

a. The minimum boron concentration in the ~~Spent~~ fuel storage pool ~~Pit~~ shall be 2300 ppm.

b. The combination of initial enrichment, burnup, and cooling time of each fuel assembly stored in the Spent Fuel Pit shall be in accordance with Specification 5.5.1. See ITS 3.7.14

Applicability APPLICABILITY: ~~At all times~~ When fuel assemblies are ~~is~~ stored in the ~~Spent~~ fuel storage pool and a fuel storge pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool ~~Pit~~.

A01

L01

ACTION:

ACTION A.1
ACTION A.2

a. With boron concentration in the ~~Spent~~ fuel storage pool ~~Pit~~ less than 2300 ppm, suspend movement of spent fuel in the ~~Spent~~ fuel storage pool ~~Pit~~ and initiate action to restore boron concentration to 2300 ppm or greater or Initiate action to perform a fuel storage pool verification, immediately.

A01

L01

b. With condition b not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions. See ITS 3.7.14

ACTION A Note c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 4.9.14.1 The boron concentration of the Spent Fuel Pit shall be verified to be 2300 ppm or greater in accordance with the Surveillance Frequency Control Program.

4.9.14.2 A representative sample of inservice Metamic inserts shall be visually inspected in accordance with the Metamic Surveillance Program described in UFSAR Section 16.2. The surveillance program ensures that the performance requirements of Metamic are met over the surveillance interval. See ITS 3.7.14

DISCUSSION OF CHANGES
ITS 3.7.13, FUEL STORAGE POOL BORON CONCENTRATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.9.14 Limiting Condition for Operation (LCO) a states "The minimum boron concentration in the fuel storage pool shall be 2300 ppm." CTS 3.9.14 ACTION a states "With boron concentration in the fuel storage pool less than 2300 ppm, suspend movement of spent fuel in the fuel storage pool and initiate action to restore boron concentration to 2300 ppm or greater." ITS 3.7.13 requires that the fuel pool boron concentration shall be ≥ 2300 ppm when fuel assemblies are stored in the fuel pool and a fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool. Additionally, a new Required Action (Required Action A.2.2) has been added to allow the initiation of an action to perform a fuel pool verification as an option to initiating an action to restore fuel pool boron concentration to within limit. This changes the CTS by changing the Applicability and by adding an option for restoring the fuel pool boron concentration to within its limit.

The purpose of CTS 3.9.14 is to ensure adequate dissolved boron is in spent fuel storage water to maintain the required subcriticality margin in the event of a fuel handling accident. This change is acceptable because the requirements continue to ensure that the boron concentration is maintained during the specified conditions assumed in the safety analyses and licensing basis (i.e., during fuel movement). Performing a spent fuel pool verification provides assurance that no fuel assemblies have been inadvertently misplaced in the spent fuel storage pool. This change is designated as less restrictive because the LCO requirements in the ITS are more relaxed than the requirements in the CTS.

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

3.7 PLANT SYSTEMS

3.7.~~16~~ 13 ~~[~~ Fuel Storage Pool Boron Concentration ~~]~~ 1 2

3.9.14 LCO 3.7.~~16~~ 13 The fuel storage pool boron concentration shall be ≥ ~~[~~2300~~]~~ ppm. 1 2

Applicability APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a ACTION c	A. Fuel storage pool boron concentration not within limit.	--------------------NOTE-------------------- LCO 3.0.3 is not applicable. ---	
		A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
		AND	
		A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
		OR	
DOC L01		A.2.2 Initiate action to perform a fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.9.14.1 SR 3.7.~~16~~.1 13	Verify the fuel storage pool boron concentration is within limit.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.13, FUEL STORAGE POOL BORON CONCENTRATION

1. Improved Standard Technical Specification (ISTS) 3.7.16 has been renumbered as Improved Technical Specifications (ITS) 3.7.13.

2. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

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

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

B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ 13 ~~[~~ Fuel Storage Pool Boron Concentration ~~]~~ 2 1

BASES

BACKGROUND

~~In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design, the spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup. [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure [3.7.17-1], in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure [3.7.17-1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.~~ 3

INSERT 1 3

The water in the spent fuel storage ~~pool~~ normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI /ANS-8.1-1983 ~~N-16.1-1975 and the April 1978 NRC letter~~ (Ref. 1 ~~3~~) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. ~~For example, the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the criticality of [Region 2]. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.17, "Spent Fuel Assembly Storage."~~ Prior to movement of an assembly, it is necessary to perform SR 3.7.~~16~~ 14.1. 3 3 3

APPLICABLE SAFETY ANALYSES

~~Most accident conditions do not result in an increase in the activity of either of the two regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the~~ 3

3

INSERT 1

The Spent Fuel Storage racks provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison to assure: a) Keff less than or equal to 0.95 with a minimum soluble boron concentration of 500 ppm present, and b) Keff less than 1.0 when flooded with unborated water for normal operations and postulated accidents. The 500 ppm value is needed to assure keff less than 0.95 for normal operating conditions. The criticality analysis needs 1700 ppm to assure keff less than 0.95 under the worst case accident condition. There is significant margin between the calculated ppm requirement and the spent fuel boron concentration requirement of 2300 ppm. The higher boron concentration value is chosen because, during refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass.

The spent fuel racks are divided into two regions, Region I and Region II. The Region I permanent racks have a 10.6 inch center-to-center spacing. The Region II racks have a 9.0 inch center-to-center spacing. The cask area storage rack has a nominal 10.1 inch center to center spacing in the east-west direction and a nominal 10.7 inch center-to-center spacing in the north-south direction.

Any fuel for use at Turkey Point, and enriched to less than or equal to 5.0 wt % U-235, may be stored in the Cask Area Storage Rack. Fresh or irradiated fuel assemblies not stored in the Cask Area Storage Rack shall be stored in accordance with LCO 3.7.14.

Fresh unirradiated fuel may be placed in the permanent Region I racks in accordance with the restrictions of Figure 3.7.14-1. Prior to placement of irradiated fuel in Region I or II spent fuel storage rack cell locations, strict controls are employed to evaluate burnup of the fuel assembly. Upon determination that the fuel assembly meets the nominal burnup and associated post-irradiation cooling time requirements of Table 3.7.14-1 or Table 3.7.14-2, it may be placed in a Region I or II cell in accordance with the restrictions of Figures 3.7.14-1 through 3.7.14-3, respectively.

For all assemblies with blanketed fuel, the initial enrichment is based on the central zone enrichment (i.e., between the axial blankets) consistent with the assumptions of the analysis. These positive controls assure that the fuel enrichment limits, burnup, and post irradiation cooling time requirements assumed in the safety analyses will NOT be violated.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~storage pool prevents criticality in both regions. The postulated accidents are basically of two types. A fuel assembly could be incorrectly transferred from [Region 1 to Region 2] (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded [Region 2] storage rack. This could have a small positive reactivity effect on [Region 2]. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is provided in the FSAR, Section [15.7.4]~~ (Ref. ~~4~~). 3

2 INSERT 2 3

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The fuel storage pool boron concentration is required to be ≥ ~~[~~2300~~]~~ ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference ~~4~~. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool. 1 2 3

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage ~~pool~~, until a complete spent fuel storage ~~pool~~ verification has been performed following the last movement of fuel assemblies in the spent fuel storage ~~pool~~. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly. 3 3

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in ~~the~~ fuel storage ~~pool~~ is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position. 3

3

INSERT 2

The following fuel handling accidents are evaluated to ensure that no hazards are created: a) A fuel assembly is dropped in containment. b) A spent fuel cask is dropped in the passage between the spent fuel pits of Units 3 & 4 while transferring a fuel element between the spent fuel pits. The consideration of a cask drop accident is historical and is retained as discussed in UFSAR Section 14.2.1.3. (Ref.2)

Since the spent fuel cask will not be handled over or in the vicinity of spent fuel except as provided for in UFSAR Section 14.2.1.3.1, the re-racking does not result in a significant increase in the probability of the cask drop accident previously evaluated in the Turkey Point Updated UFSAR. Furthermore, as shown in UFSAR Section 14.2.1.3.2, by requiring the decay time of spent fuel to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit, the potential offsite doses will be well within 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies. The proposed spent fuel pit modifications will not increase the radiological consequences of a cask drop accident previously evaluated. (Ref.2)

BASES

ACTIONS (continued)

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~16~~13.1 3

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. ~~[The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.~~ 1

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~--]~~

REFERENCES

~~[1. Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."~~

~~2. Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).]~~

1 ~~3~~. Double contingency principle of ANSI /ANS-8.1-1983 ~~N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A)~~. 3

U 2 ~~4~~. FSAR, Section 14.2.1 ~~[15.7.4]~~. 3 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.13 BASES, FUEL STORAGE POOL BORON CONCENTRATION

1. The Improved Standard Technical Specification (ISTS) contain bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS B 3.7.16 has been renumbered as Improved Technical Specifications (ITS) B 3.7.13. "Fuel Storage Pool Boron Concentration" does not change.

3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.13, Fuel Storage Pool Boron Concentration

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 14

ITS 3.7.14, SPENT FUEL STORAGE

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

REFUELING OPERATIONS

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

LCO 3.7.14 3.9.14 The following conditions shall apply to spent fuel storage:

a. The minimum boron concentration in the Spent Fuel Pit shall be 2300 ppm. See ITS 3.7.13

b. ~~The combination of initial enrichment, burnup, and cooling time of each fuel assembly stored in the Spent Fuel Pit shall be in accordance with Specification 5.5.1.~~ Add proposed ITS 3.7.14 LCO. A02

Applicability APPLICABILITY: ~~At all times when fuel is stored in the Spent Fuel Pit~~. Whenever any fuel assembly is stored in Region I or Region II of the spent fuel pit A01

ACTION:

See ITS 3.7.13

a. With boron concentration in the Spent Fuel Pit less than 2300 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 2300 ppm or greater.

ACTION A b. With condition b not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

ACTION A.1 Note c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14.1 The boron concentration of the Spent Fuel Pit shall be verified to be 2300 ppm or greater in accordance with the Surveillance Frequency Control Program. See ITS 3.7.13

~~4.9.14.2 A representative sample of inservice Metamic inserts shall be visually inspected in accordance with the Metamic Surveillance Program described in UFSAR Section 16.2. The surveillance program ensures that the performance requirements of Metamic are met over the surveillance interval.~~ LA01

Add proposed SR 3.7.14.1 M01

5.5 FUEL STORAGE

5.5.1 CRITICALITY

LCO 3.7.14 Note

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} less than 1.0 when flooded with unborated water, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

b. A k_{eff} less than or equal to 0.95 when flooded with water borated to 500 ppm, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

c. A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for the two region spent fuel pool storage racks. A nominal 10.1 inch center-to-center distance in the east-west direction and a nominal 10.7 inch center-to-center distance in the north-south direction for the cask area storage rack.

d. A maximum enrichment loading for fuel assemblies of 5.0 weight percent of U-235.

LCO 3.7.14 Note

e. No restriction on storage of fresh or irradiated fuel assemblies in the cask area storage rack.

f. Fresh or irradiated fuel assemblies not stored in the cask area storage rack shall be stored in accordance with Specification 5.5.1.3.

g. The Metamic neutron absorber inserts shall have a minimum certified ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

5.5.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than a nominal 4.5 weight percent of U-235 if the assembly contains no burnable absorber rods and no more than 5.0 weight percent of U-235 if the assembly contains at least 16 IFBA rods.

A01

ITS

5.5.1.3 Credit for burnup and cooling time is taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. Fresh or irradiated fuel assemblies in the Region I or Region II racks shall be stored in compliance with the following:

a. Any 2x2 array of Region I storage cells containing fuel shall comply with the storage patterns in Figure 5.5-1 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array.

A02

LCO 3.7.14

b. Any 2x2 array of Region II storage cells containing fuel shall:

i. Comply with the storage patterns in Figure 5.5-2 and the requirements of Tables 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less reactive than that shown for the 2x2 array,

ii. Have the same directional orientation for Metamic inserts in a contiguous group of 2x2 arrays where Metamic inserts are required, and

iii. Comply with the requirements of 5.5.1.3.c for cells adjacent to Region I racks.

c. Any 2x2 array of Region II storage cells that interface with Region I storage cells shall comply with the rules of Figure 5.5-3.

d. Any fuel assembly may be replaced with a fuel rod storage basket or non-fuel hardware.

e. Storage of Metamic inserts or RCCAs is acceptable in locations designated as empty (water-filled) cells.

See ITS 4.0

DRAINAGE

5.5.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1535 fuel assemblies.

A01

A02

Table 5.5-1

Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-6 for use of Table 5.5-1

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	5.66439153	-14.7363682	-7.74060457	-7.63345029	24.4656526	8.5452608	26.2860949
A2	-7.22610116	11.0284547	5.13978237	10.7798957	-20.3141124	-4.47257395	-18.0738662
A3	2.98646188	-1.80672781	-0.360186309	-2.81231555	6.53101471	2.09078914	5.8330891
A4	-0.287945644	0.119516492	0.0021681285	0.29284474	-0.581826027	-0.188280562	-0.517434342
A5	-0.558098618	0.0620559676	-0.0304713673	0.0795058096	-0.16567492	0.157548739	-0.0614152031
A6	0.476169245	0.0236575787	0.098844889	-0.0676341983	0.243843226	-0.0593584027	0.134626308
A7	-0.117591963	-0.0088144551	-0.0277584786	0.0335130877	-0.0712130368	0.0154678626	-0.0383060399
A8	0.0095165354	0.0008957348	0.0024057185	-0.0040803875	0.0063998706	-0.0014068318	0.0033419846
A9	-47.1782783	-20.2890089	-21.424984	14.6716317	-41.1150	-0.881964768	-12.1780
A10	33.4270029	14.7485847	16.255208	-10.0312224	43.9149156	9.69128392	23.6179517
A11	-6.11257501	-1.22889103	-1.77941882	5.62580894	-9.6599923	-0.18740168	-4.10815592
A12	0.490064351	0.0807808548	0.127321203	-0.539361868	0.836931842	0.0123398618	0.363908736

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ -\ (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal central zone U-235 enrichment. Axial blanket material is not considered when determining enrichment. Any enrichment between 2.0 and 5.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 0 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. ~~DELETED~~

5. ~~DELETED~~

6. This Table applies for any blanketed fuel assembly.

A01

A02

Table 5.5-2

Non-Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-4 for use of Table 5.5-2

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	2.04088171	-27.6637884	-11.2686777	20.7284208	29.8862876	-83.5409405	35.5058622
A2	-4.83684164	26.1997193	2.0659501	11.9673275	-37.0771132	94.7973724	-30.1986997
A3	2.59801889	-7.2982252	2.66204924	-14.4072388	16.3986049	-31.9583373	11.0102438
A4	-0.300597247	0.723731768	-0.513334362	2.83623963	-2.1571669	3.55898487	-1.27269125
A5	-0.610041808	0.401332891	-0.0987986108	-1.49118695	1.02330848	0.299948492	1.34723758
A6	0.640497159	-0.418616707	-0.0724198633	1.75361041	-1.21889631	-0.312341996	-1.19871392
A7	-0.219000712	0.144304039	0.106248806	-0.659046438	0.467440882	0.107463895	0.352920811
A8	0.0252870451	-0.0154239536	-0.0197359109	0.080884618	-0.0560129443	-0.0108814287	-0.0325155213
A9	-4.48207836	-5.54507376	-1.34620551	-245.825283	12.1549	39.4975573	-5.2576
A10	-2.12118634	-5.76555416	-10.1728821	243.59979	-22.7755385	-50.5818253	10.1733379
A11	2.91619317	6.29118025	8.71968815	-75.7805818	14.3755458	23.3093829	0.369083041
A12	-0.196645176	-0.732079719	-1.14461356	8.10936356	-1.80803352	-2.69466612	0.0443577624

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “cooling time” and “initial enrichment.” The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [- (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal U-235 enrichment. Any enrichment between 1.8 and 4.0 may be used.
3. Cooling time, Ct, is in years. Any cooling time between 15 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.
4. This Table applies only for pre-EPU non-blanketed fuel assemblies. If a non-blanketed assembly is depleted at EPU conditions, none of the burnup accrued at EPU conditions can be credited (i.e., only burnup accrued at pre-EPU conditions may be used as burnup credit).

Table 5.5-3 A02

Fuel Categories Ranked by Reactivity
See notes 1-5 for use of Table 5.5-3

Region I	**I-1**	**High Reactivity**
	I-2	
	I-3	
	I-4	**Low Reactivity**
Region II	**II-1**	**High Reactivity**
	II-2	
	II-3	
	II-4	
	II-5	**Low Reactivity**

Notes:

1. Fuel Category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category I-2 is less reactive than Category I-1, etc. The more reactive fuel categories require compensatory measures to be placed in Regions I and II of the SFP, e.g., use of water filled cells, Metamic inserts, or full length RCCAs.

2. Any higher numbered fuel category can be used in place of a lower numbered fuel category from the same Region.

3. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.

4. Category I-2 is fresh unburned fuel that obeys the IFBA requirements of Table 5.5-4.

5. All Categories except I-1 and I-2 are determined from Tables 5.5-1 and 5.5-2.

Table 5.5-4 A02

IFBA Requirements for Fuel Category I-2

Nominal Enrichment (wt% U-235)	Minimum Required Number of IFBA Pins
Enr. ≤ 4.3	0
4.3 < Enr. ≤ 4.4	32
4.4 < Enr. ≤ 4.7	64
4.7 < Enr. ≤ 5.0	80

FIGURE 5.5-1 A02

ALLOWABLE REGION I STORAGE ARRAYS

See notes 1-8 for use of Figure 5.5-1

DEFINITION	**ILLUSTRATION**

Array I-A
Checkerboard pattern of Category I-1 assemblies and empty (water-filled) cells.

I-1	X
X	I-1

Array I-B
Category I-4 assembly in every cell.

I-4	I-4
I-4	I-4

Array I-C
Combination of Category I-2 and I-4 assemblies. Each Category I-2 assembly shall contain a full length RCCA.

I-2	I-4
I-4	I-4

I-2	I-2
I-4	I-4

I-2	I-2
I-2	I-4

I-2	I-2
I-2	I-2

Array I-D
Category I-3 assembly in every cell. One of every four assemblies contains a full length RCCA.

I-3	I-3
I-3	I-3

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.
3. Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 5.5-4.
4. Categories I-3 and I-4 are determined from Tables 5.5-1 and 5.5-2.
5. Shaded cells indicate that the fuel assembly contains a full length RCCA.
6. **X** indicates an empty (water-filled) cell.
7. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-2

ALLOWABLE REGION II STORAGE ARRAYS
See notes 1-6 for use of Figure 5.5-2

DEFINITION / **ILLUSTRATION**

Array II-A
Category II-1 assembly in three of every four cells;
one of every four cells is empty (water-filled);
the cell diagonal from the empty cell contains a Metamic insert or full length RCCA.

II-1	II-1
X	II-1

X	II-1
II-1	II-1

Array II-B
Checkerboard pattern of Category II-3 and II-5 assemblies with two of every four cells containing a Metamic insert or full length RCCA.

II-3	II-5
II-5	II-3

II-3	II-5
II-5	II-3

Array II-C
Category II-4 assembly in every cell with two of every four cells containing a Metamic insert or full length RCCA.

II-4	II-4
II-4	II-4

II-4	II-4
II-4	II-4

Array II-D
Category II-2 assembly in every cell with three of every four cells containing a Metamic insert or full length RCCA.

II-2	II-2
II-2	II-2

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
6. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

FIGURE 5.5-3

INTERFACE RESTRICTIONS BETWEEN REGION I AND REGION II ARRAYS

See notes 1-8 for use of Figure 5.5-3

DEFINITION | **ILLUSTRATION**

Array II-A, as defined in Figure 5.5-2, when placed on the interface with Region I shall have the empty cell in the row adjacent to the Region I rack.

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-1	X	II-1	X
II-1	II-1	II-1	II-1

Array II-A

Arrays II-B, II-C and II-D, as defined in Figure 5.5-2, when placed on the interface with Region I shall have an insert in every cell in the row adjacent to the Region I rack.

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Array II-B

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-C

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2

Array II-D

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 5.5-1 and 5.5-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only. Region I Array I-B is depicted as the example; however, any Region I array is allowed provided that

    a. For Array I-D, the RCCA shall be in the row adjacent to the Region II rack, and

    b. Array I-A shall not interface with Array II-D.

6. If no fuel is stored adjacent to Region II in Region I, then the interface restrictions are not applicable.
7. Figure 5.5-3 is applicable only to the Region I - Region II interface. There are no restrictions for the interfaces with the cask area rack.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

DISCUSSION OF CHANGES
ITS 3.7.14, SPENT FUEL STORAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

A02 CST 3.9.14 LCO states "The combination of initial enrichment, burnup, and cooling time of each fuel assembly stored in the Spent Fuel Pit shall be in accordance with Specification 5.5.1." ITS 3.7.14 Limiting Condition for Operation (LCO) states something similar; however, instead of referencing the sections from the Design Features Section (ITS 4.0), the applicable Section is being moved to ITS 3.7.14, including a portion that is added to the LCO, and the associated Tables and Figures. This changes the CTS by moving the items referenced in the Design Features Section of CTS to ITS Section 3.7.14.

The purpose of the CTS is to ensure there is proper loading of fuel in the fuel storage racks. Whether the requirements for doing so are in the Design Features or in the Plant Systems section of Technical Specifications will not impact accomplishing this goal. Thus, moving the requirements from the one section of Technical Specifications to another will continue to ensure the fuel loading requirements are met and that appropriate action is taken if not. These changes are designated as administrative because the changes do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS Surveillance Requirement (SR) 3.7.14.1 verifies by administrative means the fuel assemblies stored in Regions I and II are stored in accordance with the requirements of Figures 3.7.14-1 through Figure 3.7.14-3 and Tables 3.7.14-1 through 3.7.14.3 with credited for burnup and cooling time taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. The CTS do not contain this SR. This changes the CTS by adding a new SR to Technical Specifications.

The purpose of the ITS SR is to verify by administrative means that the fuel in the fuel storage pool is in accordance with the requirements contained in the figures and tables in the Technical Specifications. While the CTS did not explicitly contain a requirement to perform this verification, these requirements were verified per the figure and tables that were located in the Design Features section of Technical Specifications. This change is designated as more restrictive because an SR is being added to the ITS that was not in the CTS.

DISCUSSION OF CHANGES
ITS 3.7.14, SPENT FUEL STORAGE

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS SR 4.9.14.2 states, "A representative sample of inservice metamic inserts shall be visually inspected in accordance with the Metamic Surveillance Program described in UFSAR Section 16.2. The surveillance program ensures that the performance requirements of Metamic are met over the surveillance interval." ITS 3.7.14 does contain this SR. This changes the CTS by moving a SR out of Technical Specifications and into the Technical Requirements Manual (TRM).

The purpose of this SR is to ensure the performance requirements of the Metamic are met in accordance with the Metamic Surveillance Program located in the Updated Safety Analysis Report (UFSAR). The CTS LCO does not discuss the Metamic inserts. The ITS is proposing to move this SR to the TRM. This change is acceptable because neither the LCO nor the Actions discuss the Metamic inserts. The Metamic inserts will still be required to be inspected if moved to the TRM in accordance with the UFSAR program and can be adequately controlled via the TRM. The TRM is incorporated by reference into the UFSAR and any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail because an SR is being moved from the Technical Specifications to the TRM.

LESS RESTRICTIVE CHANGES

None

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

3.7 PLANT SYSTEMS

3.7.~~17~~ 14 [Spent Fuel ~~Pool~~ Storage] 2 3

3.9.14 LCO 3.7.~~17~~ 14

INSERT 1

~~The combination of initial enrichment and burnup of each fuel assembly stored in [Region 2] shall be within the Acceptable [Burnup Domain] of Figure 3.7.17-1 or in accordance with Specification 4.3.1.1.~~ 1 2 2 4

INSERT 2

Applicability APPLICABILITY: Whenever any fuel assembly is stored in ~~[Region 2]~~ Region I or Region II of the spent fuel ~~storage pool~~ pit. 1 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b, c A. Requirements of the LCO not met.	A.1 --------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------- Initiate action to move the noncomplying fuel assembly ~~from [Region 2].~~ to an acceptable location	Immediately 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.~~17~~ 14.1 Verify by administrative means the ~~initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 or Specification 4.3.1.1~~ fuel assemblies stored in Regions I and II are stored in accordance with the requirements of Figure 3.7.14-1 through Figure 3.7.14-3 with credit for burnup and cooling time taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks.	Prior to storing the fuel assembly in ~~[Region 2]~~ Region I or II 2 3 2 3 1

4 **INSERT 1**

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5.5.1.3
DOC A02

Fuel assemblies stored in Region I and II shall be stored in accordance with the requirements of Figures 3.7.14-1 through Figure 3.7.14-3 with credit for burnup and cooling time taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. Fresh or irradiated fuel assemblies in the Region I or Region II racks shall be stored in compliance with the following:

a. Any 2x2 array of Region I storage cells containing fuel shall comply with the storage patterns in Figure 3.7.14-1 and the requirements of Tables 3.7.14-1 and 3.7.14-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 3.7.14-3) shall be equal to or less reactive than that shown for the 2x2 array.

b. Any 2x2 array of Region II storage cells containing fuel shall:

   i. Comply with the storage patterns in Figure 3.7.14-2 and the requirements of Tables 3.7.14-1 and 3.7.14-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 3.7.14-3) shall be equal to or less reactive than that shown for the 2x2 array,

   ii. Have the same directional orientation for Metamic inserts in a contiguous group of 2x2 arrays where Metamic inserts are required, and

   iii. Comply with the requirements of LCO 3.7.14.c. for cells adjacent to Region I racks.

c. Any 2x2 array of Region II storage cells that interface with Region I storage cells shall comply with the rules of Figure 3.7.14-3.

d. Any fuel assembly may be replaced with a fuel rod storage basket or non-fuel hardware.

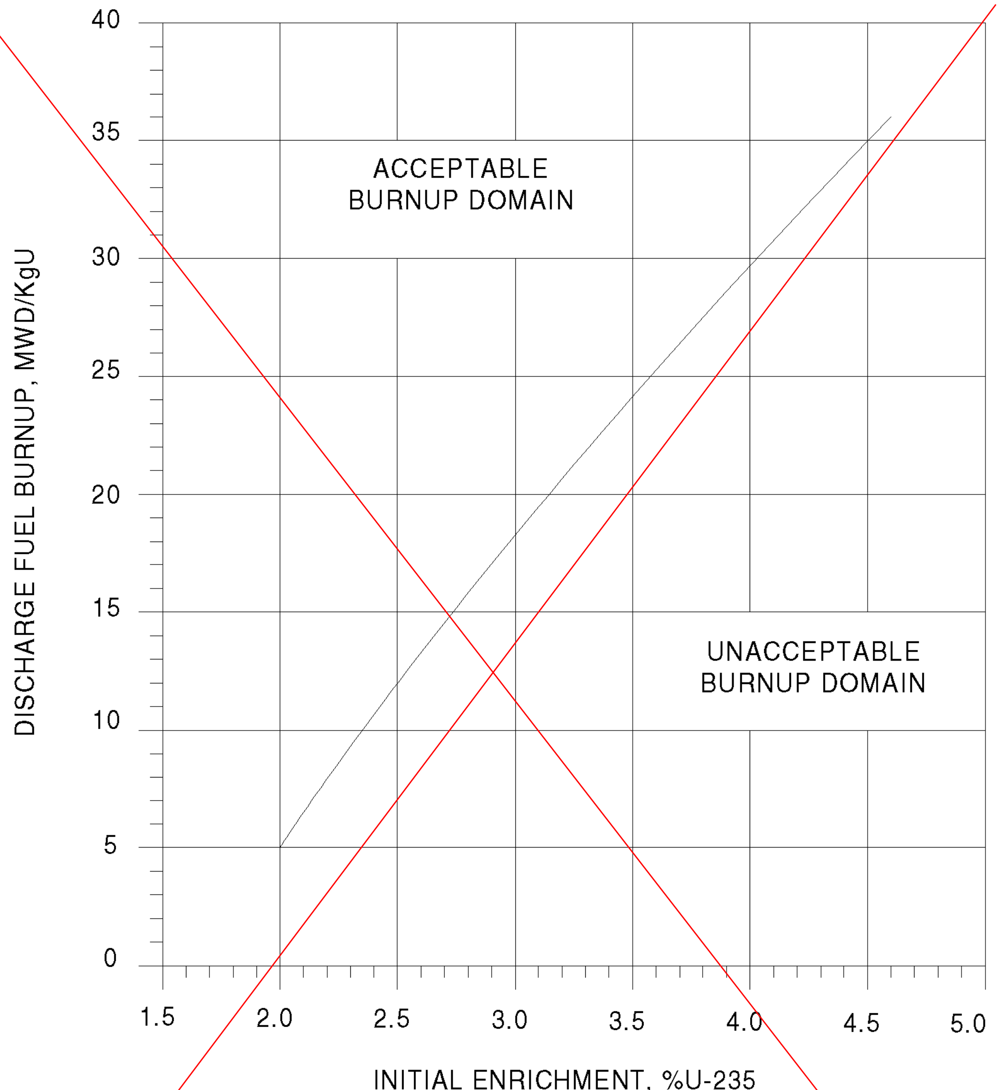
e. Storage of Metamic inserts or RCCAs is acceptable in locations designated as empty (water-filled) cells.

4 **INSERT 2**

--NOTE--

5.5.1.1.e

No restrictions on storage of fresh or irradiated fuel assemblies in the cask area storage rack are applicable.



Not to be used for Operation.
For illustration purposes only.

Figure 3.7.17-1 (page 1 of 1)
Fuel Assembly Burnup Limits in Region 2

INSERT 3
4

4

INSERT 3

CTS

Table 3.7.14-1

Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

Table 5.5-1
DOC A02

See notes 1-4 for use of Table 3.7.14-1

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	5.66439153	-14.7363682	-7.74060457	-7.63345029	24.4656526	8.5452608	26.2860949
A2	-7.22610116	11.0284547	5.13978237	10.7798957	-20.3141124	-4.47257395	-18.0738662
A3	2.98646188	-1.80672781	-0.360186309	-2.81231555	6.53101471	2.09078914	5.8330891
A4	-0.287945644	0.119516492	0.0021681285	0.29284474	-0.581826027	-0.188280562	-0.517434342
A5	-0.558098618	0.0620559676	-0.0304713673	0.0795058096	-0.16567492	0.157548739	-0.0614152031
A6	0.476169245	0.0236575787	0.098844889	-0.0676341983	0.243843226	-0.0593584027	0.134626308
A7	-0.117591963	-0.0088144551	-0.0277584786	0.0335130877	-0.0712130368	0.0154678626	-0.0383060399
A8	0.0095165354	0.0008957348	0.0024057185	-0.0040803875	0.0063998706	-0.0014068318	0.0033419846
A9	-47.1782783	-20.2890089	-21.424984	14.6716317	-41.1150	-0.881964768	-12.1780
A10	33.4270029	14.7485847	16.255208	-10.0312224	43.9149156	9.69128392	23.6179517
A11	-6.11257501	-1.22889103	-1.77941882	5.62580894	-9.6599923	-0.18740168	-4.10815592
A12	0.490064351	0.0807808548	0.127321203	-0.539361868	0.836931842	0.0123398618	0.363908736

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “cooling time” and “initial enrichment.” The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ -\ (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal central zone U-235 enrichment. Axial blanket material is not considered when determining enrichment. Any enrichment between 2.0 and 5.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 0 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. This Table applies for any blanketed fuel assembly.

CTS

Table 5.5-2
DOC A02

4 **INSERT 3 (Continued)**

Table 3.7.14-2

Non-Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See notes 1-4 for use of Table 3.7.14-2

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	2.04088171	-27.6637884	-11.2686777	20.7284208	29.8862876	-83.5409405	35.5058622
A2	-4.83684164	26.1997193	2.0659501	11.9673275	-37.0771132	94.7973724	-30.1986997
A3	2.59801889	-7.2982252	2.66204924	-14.4072388	16.3986049	-31.9583373	11.0102438
A4	-0.300597247	0.723731768	-0.513334362	2.83623963	-2.1571669	3.55898487	-1.27269125
A5	-0.610041808	0.401332891	-0.0987986108	-1.49118695	1.02330848	0.299948492	1.34723758
A6	0.640497159	-0.418616707	-0.0724198633	1.75361041	-1.21889631	-0.312341996	-1.19871392
A7	-0.219000712	0.144304039	0.106248806	-0.659046438	0.467440882	0.107463895	0.352920811
A8	0.0252870451	-0.0154239536	-0.0197359109	0.080884618	-0.0560129443	-0.0108814287	-0.0325155213
A9	-4.48207836	-5.54507376	-1.34620551	-245.825283	12.1549	39.4975573	-5.2576
A10	-2.12118634	-5.76555416	-10.1728821	243.59979	-22.7755385	-50.5818253	10.1733379
A11	2.91619317	6.29118025	8.71968815	-75.7805818	14.3755458	23.3093829	0.369083041
A12	-0.196645176	-0.732079719	-1.14461356	8.10936356	-1.80803352	-2.69466612	0.0443577624

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

   $$\mathbf{Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ -\ (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3}$$

2. Initial enrichment, En, is the nominal U-235 enrichment. Any enrichment between 1.8 and 4.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 15 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. This Table applies only for pre-EPU non-blanketed fuel assemblies. If a non-blanketed assembly is depleted at EPU conditions, none of the burnup accrued at EPU conditions can be credited (i.e., only burnup accrued at pre-EPU conditions may be used as burnup credit).

CTS

4

INSERT 3 (Continued)

Table 5.5-3
DOC A02

Table 3.7.14-3

Fuel Categories Ranked by Reactivity

See notes 1-5 for use of Table **3.7.14-3**

Region I	**I-1**	**High Reactivity**
	I-2	
	I-3	
	I-4	**Low Reactivity**
Region II	**II-1**	**High Reactivity**
	II-2	
	II-3	
	II-4	
	II-5	**Low Reactivity**

Notes:

1. Fuel Category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category I-2 is less reactive than Category I-1, etc. The more reactive fuel categories require compensatory measures to be placed in Regions I and II of the SFP, e.g., use of water filled cells, Metamic inserts, or full length RCCAs.

2. Any higher numbered fuel category can be used in place of a lower numbered fuel category from the same Region.

3. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.

4. Category I-2 is fresh unburned fuel that obeys the IFBA requirements of Table 3.7.14-4.

5. All Categories except I-1 and I-2 are determined from Tables 3.7.14-1 and 3.7.14-2.

Table 5.5-4
DOC A02

Table 3.7.14-4

IFBA Requirements for Fuel Category I-2

Nominal Enrichment (wt% U-235)	Minimum Required Number of IFBA Pins
Enr. ≤ 4.3	0
4.3 < Enr. ≤ 4.4	32
4.4 < Enr. ≤ 4.7	64
4.7 < Enr. ≤ 5.0	80

CTS

Figure 5.5-1
DOC A02

4 **INSERT 3 (Continued)**

FIGURE 3.7.14-1

ALLOWABLE REGION I STORAGE ARRAYS
See notes 1-8 for use of Figure 3.7.14-1

DEFINITION | **ILLUSTRATION**

Array I-A
Checkerboard pattern of Category I-1 assemblies and empty (water-filled) cells.

I-1	**X**
X	**I-1**

Array I-B
Category I-4 assembly in every cell.

I-4	**I-4**
I-4	**I-4**

Array I-C
Combination of Category I-2 and I-4 assemblies. Each Category I-2 assembly shall contain a full length RCCA.

I-2	**I-4**
I-4	**I-4**

I-2	**I-2**
I-4	**I-4**

I-2	**I-2**
I-2	**I-4**

I-2	**I-2**
I-2	**I-2**

Array I-D
Category I-3 assembly in every cell. One of every four assemblies contains a full length RCCA.

I-3	**I-3**
I-3	**I-3**

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.
3. Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 3.7.14-4.
4. Categories I-3 and I-4 are determined from Tables 3.7.14-1 and 3.7.14-2.
5. Shaded cells indicate that the fuel assembly contains a full length RCCA.
6. **X** indicates an empty (water-filled) cell.
7. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

CTS

4

INSERT 3 (Continued)

FIGURE 3.7.14-2

Figure 5.5-2
DOC A02

ALLOWABLE REGION II STORAGE ARRAYS

See notes 1-6 for use of Figure 3.7.14-2

DEFINITION | **ILLUSTRATION**

Array II-A
Category II-1 assembly in three of every four cells; one of every four cells is empty (water-filled); the cell diagonal from the empty cell contains a Metamic insert or full length RCCA.

II-1	II-1
X	II-1

X	II-1
II-1	II-1

Array II-B
Checkerboard pattern of Category II-3 and II-5 assemblies with two of every four cells containing a Metamic insert or full length RCCA.

II-3	II-5
II-5	II-3

II-3	II-5
II-5	II-3

Array II-C
Category II-4 assembly in every cell with two of every four cells containing a Metamic insert or full length RCCA.

II-4	II-4
II-4	II-4

II-4	II-4
II-4	II-4

Array II-D
Category II-2 assembly in every cell with three of every four cells containing a Metamic insert or full length RCCA.

II-2	II-2
II-2	II-2

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 3.7.14-1 and 3.7.14-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
6. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

4 **INSERT 3 (Continued)**

CTS

FIGURE 3.7.14-3

Figure 5.5-3
DOC A02

INTERFACE RESTRICTIONS BETWEEN REGION I AND REGION II ARRAYS

See notes 1-8 for use of Figure 3.7.14-3

DEFINITION **ILLUSTRATION**

Array II-A, as defined in Figure 3.7.14-2, when placed on the interface with Region I shall have the empty cell in the row adjacent to the Region I rack.

2

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-1	X	II-1	X
II-1	II-1	II-1	II-1

Array II-A

Arrays II-B, II-C and II-D, as defined in Figure 3.7.14-2, when placed on the interface with Region I shall have an insert in every cell in the row adjacent to the Region I rack.

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Array II-B

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-C

Region I Rack

I-4	I-4	I-4	I-4
I-4	I-4	I-4	I-4
II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2

Array II-D

Notes:

1. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Fuel categories are determined from Tables 3.7.14-1 and 3.7.14-2.
3. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
4. **X** indicates an empty (water-filled) cell.
5. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only. Region I Array I-B is depicted as the example; however, any Region I array is allowed provided that

   a. For Array I-D, the RCCA shall be in the row adjacent to the Region II rack, and

   b. Array I-A shall not interface with Array II-D.

6. If no fuel is stored adjacent to Region II in Region I, then the interface restrictions are not applicable.
7. Figure 3.7.14-3 is applicable only to the Region I - Region II interface. There are no restrictions for the interfaces with the cask area rack.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in all storage arrays.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.14, SPENT FUEL STORAGE

1. The Improved Standard Technical Specification (ISTS) contain bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS 3.7.17, "Spent Fuel Pool Storage" has been renumbered as Improved Technical Specifications (ITS) 3.7.14, "Spent Fuel Storage."

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

4. The ITS incorporates the spent fuel storage section of the Turkey Point Nuclear Generating Station Current Technical Specifications (CTS) Design Features Chapter (CTS 5.0) which consists of requirements, and tables and figures. The fuel storage section requirements are incorporated in the Limiting Condition for Operation (LCO). The LCO also references the tables and figures which are being added to the end of the Technical Specifications.

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

2 1

B 3.7 PLANT SYSTEMS

B 3.7.~~17~~ 14 ~~[~~ Spent Fuel ~~Pool~~ Storage ~~]~~

2 1

BASES

BACKGROUND

~~In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design, the spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup. [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.17-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure [3.7.17-1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.~~ 3

INSERT 1 3

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of ~~0.95~~ 1.0 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI /ANS-8.1-1983 ~~N-16.1-1975 and the April 1978 NRC letter~~ (Ref. ~~3~~ 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. ~~For example, the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the criticality of [Region 2]. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO.~~ Prior to movement of an assembly, it is necessary to perform SR 3.7.~~16~~ 14.1.

3 3 3 3

3

INSERT 1

The Spent Fuel Storage racks provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison to assure: a) Keff less than or equal to 0.95 with a minimum soluble boron concentration of 500 ppm present, and b) Keff less than 1.0 when flooded with unborated water for normal operations and postulated accidents. The 500 ppm value is needed to assure keff less than 0.95 for normal operating conditions. The criticality analysis needs 1700 ppm to assure keff less than 0.95 under the worst case accident condition. There is significant margin between the calculated ppm requirement and the spent fuel boron concentration requirement of 2300 ppm. The higher boron concentration value is chosen because, during refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass.

The spent fuel racks are divided into two regions, Region I and Region II. The Region I permanent racks have a 10.6 inch center-to-center spacing. The Region II racks have a 9.0 inch center-to-center spacing. The cask area storage rack has a nominal 10.1 inch center to center spacing in the east-west direction and a nominal 10.7 inch center-to-center spacing in the north-south direction.

Any fuel for use at Turkey Point, and enriched to less than or equal to 5.0 wt % U-235, may be stored in the Cask Area Storage Rack. Fresh or irradiated fuel assemblies NOT stored in the Cask Area Storage Rack shall be stored in accordance with ITS 3.7.14.

Fresh unirradiated fuel may be placed in the permanent Region I racks in accordance with the restrictions of Figure 3.7.14-1. Fresh unirradiated fuel may be placed in the permanent Region II racks in accordance with the restrictions of Figure 3.7.14-2. Prior to placement of irradiated fuel in Region I or II spent fuel storage rack cell locations, strict controls are employed to evaluate burnup of the fuel assembly. Upon determination that the fuel assembly meets the nominal burnup and associated post-irradiation cooling time requirements of Table 3.7.14-1 or Table 3.7.14-2, it may be placed in a Region I or II cell in accordance with the restrictions of Figures 3.7.14-1 through 3.7.14-3, respectively.

For all assemblies with blanketed fuel, the initial enrichment is based on the central zone enrichment (i.e., between the axial blankets) consistent with the assumptions of the analysis. These positive controls assure that the fuel enrichment limits, burnup, and post irradiation cooling time requirements assumed in the safety analyses will NOT be violated.

BASES

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (Ref. ~~4~~ 2). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.~~16~~ 13, "Fuel Storage Pool Boron Concentration") prevents criticality in both regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel ~~pool~~ storage, in accordance with Figure 3.7.~~17~~ 14-1 through 3.7.14-3 and Tables 3.7.14-1 through 3.7.14.4, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain < ~~0.95~~ 1.0, assuming the pool to be flooded with unborated water. ~~Fuel assemblies not meeting the criteria of Figure [3.7.17-1] shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.~~ INSERT 2

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in ~~[Region 2]~~ Region I or Region II of the spent fuel storage ~~pool~~.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in ~~[Region 2]~~ the spent fuel storage pool is not in accordance with ~~Figure 3.7.17-1, or paragraph 4.3.1.1~~ Figure 3.7.14-1, Figure 3.7.14-2, or Figure 3.7.14-3, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance ~~with Figure 3.7.17-1 or Specification 4.3.1.1~~.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of Freactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

INSERT 2

UFSAR Section 9.5.2.3

The LCO is modified by a Note indicating that there are no restrictions on storage of fresh or irradiated fuel assemblies in the cask area storage rack. This is because in the cask area rack criticality is prevented by the design of the rack which limits fuel assembly interaction by fixing the separation distance between stored assemblies and/or by placing a neutron absorber panel between storage cells.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~17~~ 14 .1

This SR verifies by administrative means that fuel assemblies stored in Regions I and II are stored in accordance with the requirements of Figure 3.7.14-1 through Figure 3.7.14 with credit for burnup and cooling time taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. ~~the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.17-1] in the accompanying LCO~~. For fuel assemblies not meeting requirements of Figures 3.7.14-1 through 3.7-14-3 ~~in the unacceptable range of Figure 3.7.17-1~~, performance of this SR will ensure compliance with Specification 4.3.1.1. 3

REFERENCES

~~[1. Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."~~

~~2. Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).]~~

1 ~~3~~. Double contingency principle of ANSI /ANS-8.1-1983 ~~N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A)~~. 3

2 ~~4~~. U FSAR, Section 14.2.1 ~~[15.7.4]~~. 3 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.14 BASES, SPENT FUEL STORAGE

1. The Improved Standard Technical Specification (ISTS) contain bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. ISTS 3.7.17, "Spent Fuel Pool Storage" has been renumbered as Improved Technical Specifications (ITS) 3.7.14, "Spent Fuel Storage."

3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.14, SPENT FUEL STORAGE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 15
Relocated/Deleted Current Technical Specifications (CTS) in the Turkey Point Unit 3 and Unit 4 ITS Conversion

- **CTS 3.7.6 – Snubbers**
- **CTS 3.7.7 – Sealed Source Contamination**

CTS 3.7.6 SNUBBERS

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

~~PLANT SYSTEMS~~

LA01

~~3/4.7.6 SNUBBERS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.6 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.~~

~~APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.~~

~~ACTION:~~

~~With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and determine the impact on the attached component by evaluation in accordance with Specification 4.7.6, or declare the attached system inoperable and follow the appropriate ACTION statement for that system.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program in Specification 6.8.4.m.~~

DISCUSSION OF CHANGES
CTS 3.7.6, SNUBBERS

ADMINISTRATIVE CHANGES

NONE

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 4 - Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or ISI Program*) In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), CTS 3.7.6 provides requirements for all safety-related snubbers. This specification, with the exception of the Action to restore an inoperable snubber within 72 hours, is not included in the ITS. This changes the CTS by moving the explicit snubber requirements from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specification is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of the snubber requirements is to ensure that the structural integrity of the Reactor Coolant System and other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

This change is acceptable because the Limiting Condition for Operation (LCO) requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Specifically, ITS LCO 3.0.8 continues to require, in part, action to assess and manage risk during the period one or more required snubbers are unable to perform the associated support function(s). ITS LCO 3.0.8 also places limits on the delay period when one or more required snubbers are unable to perform the associated support function(s) based on the affect the degraded snubber has on the support system (e.g., affects one train or multiple trains). Refer to Section 3.0 Discussion of Changes related to the addition of ITS LCO 3.0.8. The requirement to perform snubber inspections is specified in 10 CFR 50.55a and the requirement to perform snubber inspections and testing is specified in ASME Section XI, as modified by approved relief requests. Therefore, both PTN commitments and NRC regulations or generic guidance contain the necessary programmatic requirements for the inspection and testing of safety related snubbers without repeating them in the ITS. ASME code requirements associated with snubber inspections and testing will continue to be controlled pursuant 10 CFR 50.55a. This change is acceptable because the removed information will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. Snubber limitations do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Snubber limitations are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This Technical Specification specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the Design Basis Accident (DBA).

3. Snubber limitations are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Snubber limitations were found to be non-significant risk contributor to core damage frequency and offsite releases. These indications are not structures, systems, or components that operating experience or probabilistic safety assessment has shown to be significant to the public health and safety.

Because 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied, the Snubber Specification may be relocated to a licensee-controlled document outside the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

NONE

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.6, SNUBBERS

There are no specific No Significant Hazards Considerations for this Specification.

CTS 3.7.7, SEALED SOURCE CONTAMINATION

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

A01

~~PLANT SYSTEMS~~

~~3/4.7.7 SEALED SOURCE CONTAMINATION~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.7 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With a sealed source having removable contamination in excess of the above limits, 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 Commission Regulations.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

R01

~~4.7.7.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:~~

~~a. The licensee, or~~

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

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

~~4.7.7.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.~~

~~a. Sources in use - in accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive materials:~~

~~1) With a half-life greater than 30 days (excluding Hydrogen 3), and~~

~~2) In any form other than gas.~~

A01

~~PLANT SYSTEMS~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and~~

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

R01

~~4.7.7.3 Deleted~~

~~4.7.7.4 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.~~

A01

~~PLANT SYSTEMS~~

~~3/4 3.7.8 DELETED~~

A01

~~PLANT SYSTEMS~~

~~3/4 3.7.9 DELETED~~

DISCUSSION OF CHANGES
CTS 3.7.7, SEALED SOURCE CONTAMINATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 5.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 CTS 3.7.7 provides the requirements that each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

This requirement and the associated surveillance requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. Sealed Source Contamination limitations do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Sealed Source Contamination limitations are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This Technical Specification specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the DBA.

3. Sealed Source Contamination limitations are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Sealed Source Contamination limitations were found to be non-significant risk contributor to core damage frequency and offsite releases. These indications are not structures, systems, or components that operating experience or probabilistic safety assessment has shown to be significant to the public health and safety.

Because 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied, the Sealed Source Contamination Specification may be relocated to a licensee-controlled document outside the Technical Specifications. Requirements associated with the sealed sources are governed by 10 CFR Part 70. Compliance with applicable portions of 10 CFR Part 70 is required by the operating licenses of PTN Units 3 and 4. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification does not meet the criteria in 10 CFR 50.36(c)(2)(ii) and will be relocated to the TRM.

<u>REMOVED DETAIL CHANGES</u>

None

<u>LESS RESTRICTIVE CHANGES</u>

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.7, SEALED SOURCE CONTAMINATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 16
Improved Standard Technical Specifications (ISTS) Not Adopted in the Turkey Point Unit 3 and Unit 4 ITS Conversion

- **ISTS 3.7.4 – Atmospheric Dump Valves**
- **ISTS 3.7.12 – Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)**
- **ISTS 3.7.13 – Fuel Building Air Cleanup System (FBACS)**
- **ISTS 3.7.14 - Penetration Room Exhaust Air Cleanup System (PREACS)**

ISTS 3.7.4, ATMOSPHERIC DUMP VALVES (ADVS)

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

1

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valves (ADVs)

LCO 3.7.4 [Three] ADV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ADV line inoperable.	A.1	Restore required ADV line to OPERABLE status.	7 days [<u>OR</u> In accordance with the Risk Informed Completion Time Program]
B. Two or more required ADV lines inoperable.	B.1	Restore all but one ADV line to OPERABLE status.	24 hours [<u>OR</u> In accordance with the Risk Informed Completion Time Program]
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 4 without reliance upon steam generator for heat removal.	[24] hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ADV.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.4.2	[Verify one complete cycle of each ADV block valve.	[[18] months OR In accordance with the Surveillance Frequency Control Program]]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.4, ADVs

1. Improved Standard Technical Specification (ISTS) 3.7.4, Atmospheric Dump Valves (ADVs), are not being included in the Turkey Point Nuclear Generating Station (PTN) Improved Technical Specifications (ITS). The PTN Current Technical Specifications (CTS) does not contain ADVs nor does the safety analysis credit ADVs for accident mitigation. However, the accident analysis describes the use of the ADVs if available.

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

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

1

BASES

BACKGROUND

The ADVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section [10.3] (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV line for each of the [four] steam generators is provided. Each ADV line consists of one ADV and an associated block valve.

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are usually provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen supply is sized to provide the sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System cooldown to RHR entry conditions.

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions. The design rate of [75]°F per hour is applicable for two steam generators, each with one ADV. This rate is adequate to cool the unit to RHR entry conditions with only one steam generator and one ADV, utilizing the cooling water supply available in the CST.

In the accident analysis presented in Reference 1, the ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the ADVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the

APPLICABLE SAFETY ANALYSES (continued)

operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ADVs. The number of ADVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the number of unit loops and consideration of any single failure assumptions regarding the failure of one ADV to open on demand.

The ADVs are equipped with block valves in the event an ADV spuriously fails to open or fails to close during use.

The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

[Three] ADV lines are required to be OPERABLE. One ADV line is required from each of [three] steam generators to ensure that at least one ADV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second ADV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ADV line. A closed block valve does not render it or its ADV line inoperable if operator action time to open the block valve is supported in the accident analysis.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days [or in accordance with the Risk Informed Completion Time Program]. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines, a nonsafety grade backup in the Steam Bypass System, and MSSVs.

1

BASES

ACTIONS (continued)

B.1

With two or more ADV lines inoperable, action must be taken to restore all but one ADV line to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]

C.1 and C.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within [24] hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. [Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. The Frequency is acceptable from a reliability standpoint.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
--]

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.7.4.2

The function of the block valve is to isolate a failed open ADV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. [Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. The Frequency is acceptable from a reliability standpoint.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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REFERENCES

1. FSAR, Section [10.3].

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.4 BASES, Atmospheric Dump Valves (ADVs)

1. Improved Standard Technical Specification (ISTS) 3.7.4 Bases are being deleted to be consistent with the deletion of ISTS 3.7.4.

ISTS 3.7.12, EMERGENCY CORE COOLING SYSTEM (ECCS) PUMP ROOM EXHAUST AIR CLEANUP SYSTEM (PREACS)

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

1

3.7 PLANT SYSTEMS

3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.12 Two ECCS PREACS trains shall be OPERABLE.

--NOTE--
The ECCS pump room boundary may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ECCS PREACS train inoperable.	A.1 Restore ECCS PREACS train to OPERABLE status.	7 days
B. Two ECCS PREACS trains inoperable due to inoperable ECCS pump room boundary.	B.1 Restore ECCS pump room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 --------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ECCS PREACS train for ≥ 15 continuous minutes [with heaters operating].	[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.12.2	Perform required ECCS PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
SR 3.7.12.3	Verify each ECCS PREACS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.12.4	Verify one ECCS PREACS train can maintain a pressure ≤ [-0.125] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of ≤ [3000] cfm.	[[18] months on a STAGGERED TEST BASIS OR In accordance with the Surveillance Frequency Control Program]

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.12.5	[Verify each ECCS PREACS filter bypass damper can be closed, except for dampers that are locked, sealed, or otherwise secured in the closed position.	[[18] months <u>OR</u> In accordance with the Surveillance Frequency Control Program]]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.12, ECCS PREACS

1. Improved Standard Technical Specification (ISTS) 3.7.12, Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS), is not being included in the Turkey Point Nuclear Generating Station (PTN) Improved Technical Specifications (ITS). The PTN Current Technical Specifications (CTS) does not contain ECCS PREACS nor does the safety analysis credit ECCS PREACS.

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

1

B 3.7 PLANT SYSTEMS

B 3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower reaches of the Auxiliary Building.

The ECCS PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of a safety injection (SI) signal.

The ECCS PREACS is a standby system, aligned to bypass the system HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned, and fans are started to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ECCS pump room isolate, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively) since it may be used for normal, as well as post accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 4).

1

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS PREACS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an SI pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 100 (Ref. 5) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of Reference 5 limits). The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The ECCS PREACS also actuates following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing.

Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

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

LCO

Two independent and redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding 10 CFR 100 limits in the event of a Design Basis Accident (DBA).

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

An ECCS PREACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE,

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

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

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

1

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCS PREACS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ECCS PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ECCS PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

Concurrent failure of two ECCS PREACS trains would result in the loss of functional capability; therefore, LCO 3.0.3 must be entered immediately.

B.1

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

If the ECCS pump room boundary is inoperable, the ECCS PREACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ECCS pump room boundary within 24 hours. During the period that the ECCS pump room boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump room boundary.

1

BASES

ACTIONS (continued)

C.1 and C.2

If the ECCS PREACS train or ECCS pump room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 6). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 6, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Operation [with the heaters on] for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that [heater failure,] blockage, fan or motor failure, or excessive vibration can be detected for corrective action. [The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

1

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
--]

SR 3.7.12.2

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

SR 3.7.12.3

This SR verifies that each ECCS PREACS train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is consistent with that specified in Reference 4.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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SR 3.7.12.4

This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the ECCS PREACS. During the [post accident] mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ECCS PREACS is designed to maintain a ≤[-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm from the ECCS pump room. [The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

This test is conducted with the tests for filter penetration; thus, an [18] month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 4.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.7.12.5

Operating the ECCS PREACS bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the ECCS PREACS bypass damper is verified if it can be specified in Reference 4. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the closed position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the closed position since the affected dampers were verified to be in the closed position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be opened to support the accident analysis. Restoration of an automatic damper to the opened position requires verification that the SR has been met within its required Frequency. [An [18] month Frequency is consistent with that specified in Reference 4.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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REFERENCES

1. FSAR, Section [6.5.1].

2. FSAR, Section [9.4.5].

3. FSAR, Section [15.6.5].

4. Regulatory Guide 1.52 (Rev. 2).

5. 10 CFR 100.11.

6. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.12 BASES, ECCS PREACS

1. Improved Standard Technical Specification (ISTS) 3.7.12 Bases are being deleted to be consistent with the deletion of ISTS 3.7.12.

ISTS 3.7.13, FUEL BUILDING AIR CLEANUP SYSTEM (FBACS)

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

1

3.7 PLANT SYSTEMS

3.7.13 Fuel Building Air Cleanup System (FBACS)

LCO 3.7.13 Two FBACS trains shall be OPERABLE.

---NOTE---
The fuel building boundary may be opened intermittently under administrative control.

APPLICABILITY: [MODES 1, 2, 3, and 4,]
During movement of [recently] irradiated fuel assemblies in the fuel building.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One FBACS train inoperable.	A.1 Restore FBACS train to OPERABLE status.	7 days
B. Two FBACS trains inoperable due to inoperable fuel building boundary in MODE 1, 2, 3, or 4.	B.1 Restore fuel building boundary to OPERABLE status.	24 hours

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. [Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Two FBACS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 AND C.2	Be in MODE 3. --------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours]
D. Required Action and associated Completion Time [of Condition A] not met during movement of [recently] irradiated fuel assemblies in the fuel building.	D.1 OR D.2	Place OPERABLE FBACS train in operation. Suspend movement of [recently] irradiated fuel assemblies in the fuel building.	Immediately Immediately
E. Two FBACS trains inoperable during movement of [recently] irradiated fuel assemblies in the fuel building.	E.1	Suspend movement of [recently] irradiated fuel assemblies in the fuel building.	Immediately

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FBACS train for ≥ 15 continuous minutes [with heaters operating].	[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.13.2	Perform required FBACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
SR 3.7.13.3	[Verify each FBACS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]]
SR 3.7.13.4	Verify one FBACS train can maintain a pressure ≤ [-0.125] inches water gauge with respect to atmospheric pressure during the [post accident] mode of operation at a flow rate ≤ [20,000] cfm.	[[18] months on a STAGGERED TEST BASIS OR In accordance with the Surveillance Frequency Control Program]

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.5	[Verify each FBACS filter bypass damper can be closed, except for dampers that are locked, sealed, or otherwise secured in the closed position.	[[18] months <u>OR</u> In accordance with the Surveillance Frequency Control Program]]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.13, FBACS

1. Improved Standard Technical Specification (ISTS) 3.7.13, Fuel Building Air Cleanup System (FBACS), is not being included in the Turkey Point Nuclear Generating Station (PTN) Improved Technical Specifications (ITS). The PTN Current Technical Specifications (CTS) does not contain FBACS nor does the safety analysis credit FBACS.

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

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Building Air Cleanup System (FBACS)

1

BASES

BACKGROUND

The FBACS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or loss of coolant accident (LOCA). The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters, functioning to reduce the relative humidity of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal.

The FBACS is a standby system, parts of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharges from the building, the fuel handling building is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The FBACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.7.4] (Refs. 1, 2, and 3, respectively) because it may be used for normal, as well as post accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The FBACS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident [involving handling recently irradiated fuel]. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the FBACS. The DBA analysis of the fuel handling accident assumes that only one train of the FBACS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive

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APPLICABLE SAFETY ANALYSES (continued)

material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident and for a LOCA. [Due to radioactive decay, FBACS is only required to isolate during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days).] These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

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

LCO

Two independent and redundant trains of the FBACS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR 100 (Ref. 5) limits in the event of a fuel handling accident [involving handling recently irradiated fuel].

The FBACS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE in both trains. An FBACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE,

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

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

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

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BASES

APPLICABILITY

In MODE 1, 2, 3, or 4, the FBACS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

In MODE 5 or 6, the FBACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

During movement of [recently] irradiated fuel in the fuel handling area, the FBACS is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1

With one FBACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the FBACS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable FBACS train, and the remaining FBACS train providing the required protection.

B.1

-----------------------------------REVIEWER'S NOTE------------------------
Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.

If the fuel building boundary is inoperable in MODE 1, 2, 3, or 4, the FBACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE fuel building boundary within 24 hours. During the period that the fuel building boundary is inoperable, appropriate compensatory measures [consistent with the intent, as

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BASES

ACTIONS (continued)

applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the fuel building boundary.

[C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both FBACS trains are inoperable for reasons other than an inoperable fuel building boundary (i.e., Condition B), the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in MODE 3 within 6 hours, and in MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 6). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 6, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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BASES

ACTIONS (continued)

The 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.]

D.1 and D.2

When Required Action A.1 cannot be completed within the required Completion Time, during movement of [recently] irradiated fuel assemblies in the fuel building, the OPERABLE FBACS train must be started immediately or [recently] irradiated fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.

If the system is not placed in operation, this action requires suspension of [recently] irradiated fuel movement, which precludes a fuel handling accident [involving handling recently irradiated fuel]. This does not preclude the movement of fuel assemblies to a safe position.

E.1

When two trains of the FBACS are inoperable during movement of [recently] irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of [recently] irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Operation [with the heaters on] for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that [heater failure,] blockage, fan or motor failure, or excessive vibration can be detected for corrective action. [The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

OR

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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[SR 3.7.13.2

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

[SR 3.7.13.3

This SR verifies that each FBACS train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is consistent with Reference 7.]

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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SR 3.7.13.4

This SR verifies the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBACS. During the [post accident] mode of operation, the FBACS is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered LEAKAGE. The FBACS is designed to maintain a ≤[-0.125] inches water gauge with respect to atmospheric pressure at a flow rate of [20,000] cfm to the fuel building. [The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 8).

An [18] month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 7.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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[SR 3.7.13.5

Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the closed position. The SR does not apply to dampers that are locked, sealed, or

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BASES

SURVEILLANCE REQUIREMENTS (continued)

otherwise secured in the closed position since the affected dampers were verified to be in the closed position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be opened to support the accident analysis. Restoration of an automatic damper to the opened position requires verification that the SR has been met within its required Frequency. [An [18] month Frequency is consistent with Reference 7.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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REFERENCES

1. FSAR, Section [6.5.1].

2. FSAR, Section [9.4.5].

3. FSAR, Section [15.7.4].

4. Regulatory Guide 1.25.

5. 10 CFR 100.

6. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

7. Regulatory Guide 1.52, Rev. [2].

8. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.13 BASES, FBACS

1. Improved Standard Technical Specification (ISTS) 3.7.13 Bases are being deleted to be consistent with the deletion of ISTS 3.7.13.

ISTS 3.7.14, PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM (PREACS)

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

3.7 PLANT SYSTEMS

1

3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.14 Two PREACS trains shall be OPERABLE.

--NOTE--
The penetration room boundary may be opened intermittently under administrative control.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One PREACS train inoperable.	A.1	Restore PREACS train to OPERABLE status.	7 days
B. Two PREACS trains inoperable due to inoperable penetration room boundary.	B.1	Restore penetration room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	--------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE 4.	12 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Operate each PREACS train for ≥ 15 continuous minutes [with heaters operating].	[31 days <u>OR</u> In accordance with the Surveillance Frequency Control Program]
SR 3.7.14.2	Perform required PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
SR 3.7.14.3	[Verify each PREACS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months <u>OR</u> In accordance with the Surveillance Frequency Control Program]]
SR 3.7.14.4	[Verify one PREACS train can maintain a pressure ≤ [-0.125] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of ≤ [3000] cfm.	[[18] months on a STAGGERED TEST BASIS <u>OR</u> In accordance with the Surveillance Frequency Control Program]]

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.14.5	[Verify each PREACS filter bypass damper can be closed, except for dampers that are locked, sealed, or otherwise secured in the closed position.	[[18] months <u>OR</u> In accordance with the Surveillance Frequency Control Program]]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.14, PREACS

1. Improved Standard Technical Specification (ISTS) 3.7.14, Pump Room Exhaust Air Cleanup System (PREACS), is not being included in the Turkey Point Nuclear Generating Station (PTN) Improved Technical Specifications (ITS). The PTN Current Technical Specifications (CTS) does not contain PREACS nor does the safety analysis credit PREACS.

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

B 3.7 PLANT SYSTEMS

1

B 3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND	The PREACS filters air from the penetration area between containment and the Auxiliary Building. The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation, as well as demisters, functioning to reduce the relative humidity of the air stream, also form part of the system. A second bank of HEPA filters, which follows the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection signal. The PREACS is a standby system, parts of which may also operate during normal unit operations. During emergency operations, the PREACS dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating signal(s), normal air discharges from the penetration room, the penetration room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters and charcoal adsorbers. The PREACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively) since it may be used for normal, as well as post accident, atmospheric cleanup functions. Heaters may be included for moisture removal on systems operating in high humidity conditions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 4).
APPLICABLE SAFETY ANALYSES	The PREACS design basis is established by the large break loss of coolant accident (LOCA). The system evaluation assumes a passive failure outside containment, such as valve packing leakage during a Design Basis Accident (DBA). In such a case, the system restricts the failure outside containment, such as valve packing leakage during a

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APPLICABLE SAFETY ANALYSES (continued)

Design Basis Accident (DBA). In such a case, the system restricts the radioactive release to within the 10 CFR 100 (Ref. 4) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The analysis of the effects and consequences of a large break LOCA are presented in Reference 3.

Two types of system failures are considered in the accident analysis: a complete loss of function, and excessive LEAKAGE. Either type of failure may result in less efficient removal of any gaseous or particulate material released to the penetration room following a LOCA.

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

LCO

Two independent and redundant trains of the PREACS are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power.

The PREACS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PREACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE,

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

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

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

APPLICABILITY

In MODES 1, 2, 3, and 4, the PREACS is required to be OPERABLE, consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODE 5 or 6, the PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

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BASES

ACTIONS

A.1

With one PREACS train inoperable, the action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the PREACS function. The 7 day Completion Time is appropriate because the risk contribution of the PREACS is less than that of the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this period, and the remaining train providing the required capability.

B.1

----------------------------------REVIEWER'S NOTE----------------------------------
Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.

If the penetration room boundary is inoperable, the PREACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE penetration room boundary within 24 hours. During the period that the penetration room boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the penetration room boundary.

C.1 and C.2

If the inoperable train or penetration room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

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BASES

ACTIONS (continued)

Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 5). In MODE 4 the steam generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 5, the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Operation [with the heaters on] for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that [heater failure,] blockage, fan or motor failure, or excessive vibration can be detected for corrective action. [The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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SR 3.7.14.2

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

[SR 3.7.14.3

This SR verifies that each PREACS starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is consistent with that specified in Reference 6.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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SURVEILLANCE REQUIREMENTS (continued)

----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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[SR 3.7.14.4

This SR verifies the integrity of the penetration room enclosure. The ability of the penetration room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of PREACS. During the [post accident] mode of operation, the PREACS is designed to maintain a ≤[-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm in the penetration room, with respect to adjacent areas, to prevent unfiltered LEAKAGE.

The minimum system flow rate maintains a slight negative pressure in the penetration room area, and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is operating. The number of filter elements is selected to limit the flow rate through any individual element to about [3000] cfm. This may vary based on filter housing geometry. The maximum limit ensures that the flow through, and pressure drop across, each filter element are not excessive.

The number and depth of the adsorber elements ensure that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125] second residence time is necessary for an assumed [99]% efficiency.

The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance, and is based on manufacturers' recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop or a decrease in flow indicates that the filter is being loaded or that there are other problems with the system.

[This test is conducted along with the tests for filter penetration; thus, the [18] month Frequency is consistent with that specified in Reference 6. The Frequency of [18] months is also consistent with the guidance provided in NUREG-0800 (Ref. 7).

OR

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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[SR 3.7.14.5

It is necessary to operate the PREACS filter bypass damper to ensure that the system functions properly. The OPERABILITY of the PREACS filter bypass damper is verified if it can be closed. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the closed position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the closed position since the affected dampers were verified to be in the closed position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be opened to support the accident analysis. Restoration of an automatic damper to the opened position requires verification that the SR has been met within its required Frequency. [An [18] month Frequency is consistent with that specified in Reference 6.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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BASES

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.6.5].
4. 10 CFR 100.
5. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.
6. Regulatory Guide 1.52, Rev. [2].
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.14 BASES, PREACS

1. Improved Standard Technical Specification (ISTS) 3.7.14 Bases are being deleted to be consistent with the deletion of ISTS 3.7.14.