SUMMARY OF CHANGES
ITS SECTION 3.9

Change Description	Affected Pages
A self-identified change for ITS 3.9.2 has been made. CTS Amendments 283 (Unit 1) and 267 (Unit 2) have been incorporated into the ITS submittal. This CTS change deleted CTS 4.9.2 CHANNEL FUNCTIONAL TEST requirements, modified the CTS 4.9.2 CHANNEL CHECK requirement (new CTS 4.9.2.a), and added a CHANNEL CALIBRATION requirement (new CTS 4.9.2.b). This change does not affect the ITS. However, since the CTS 4.9.2.b Frequency is 18 months, and the proposed ITS SR 3.9.2.2 Frequency is 24 months, a new ITS 3.9.2 DOC L.4 has been provided, adding this item to the scope of Beyond Scope Issue 21.	Pages 26, 27, 28, 29, 30, 31, 32, 33, and 36 of 188.
A self-identified change for ITS 3.9.2 has been made. This change revises ITS 3.9.2 and Required Action A.2 to be consistent with the wording in TSTF-286 as used in other ITS sections.	Pages 35 and 37 of 188.
A self-identified change for ITS 3.9.2 Bases has been made. This change revises the ITS 3.9.2 Bases LCO Section to clarify that audible count rate in the control room is required.	Page 41 of 188.
The change described in the response to Question 200406041439 for ITS 3.9.4 has been made. This change revises ITS 3.9.4 LCO Note and Required Action A.1 to be consistent with the wording in TSTF-286.	Page 90 of 188.
The change described in the response to Question 200406041441 for ITS 3.9.5 has been made. This change revises ITS 3.9.5 Required Action B.1 to be consistent with the wording in TSTF-286.	Page 115 of 188.

VOLUME 14

CNP UNITS 1 AND 2
IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

ITS SECTION 3.9
REFUELING OPERATIONS

Revision 1

LIST OF ATTACHMENTS

1. ITS 3.9.1
2. ITS 3.9.2
3. ITS 3.9.3
4. ITS 3.9.4
5. ITS 3.9.5
6. ITS 3.9.6
7. Relocated/Deleted Current Technical Specifications (CTS)
8. Improved Standard Technical Specifications (ISTS) not adopted in the CNP ITS

ATTACHMENT 1

ITS 3.9.1, Boron Concentration

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

ITS 3.9.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

and the refueling cavity — A.2

LCO 3.9.1 — 3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

within the limit specified in the COLR — LA.1

APPLICABILITY: MODE 6

Add proposed Applicability Note — L.1

ACTION:

ACTION A — a. With the requirements of the above specification not satisfied, 1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2, and 2) initiate and continue boration at greater than or equal to 34 gpm of 6,550 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive. — M.1, L.2

b. The provisions of Specification 3.0.3 are not applicable. — A.3

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to: — L.3

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

SR 3.9.1.1
SR 3.9.1.2 — 4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours. — LA.2

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

and the refueling cavity

A.2

LCO 3.9.1 3.9.1 The boron concentration of ~~all filled portions of~~ the Reactor Coolant System ~~and~~ the refueling canal shall be maintained ~~uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:~~

~~a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or~~

~~b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.~~

within the limit specified in the COLR

LA.1

APPLICABILITY: MODE 6

Add proposed Applicability Note

L.1

ACTION:

ACTION A a. With the requirements of the above specification not satisfied, 1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes ~~except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2, and~~ 2) initiate and continue boration ~~at greater than or equal to 24 gpm of 6,550 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive.~~

M.1

L.2

~~b. The provisions of Specification 3.0.3 are not applicable.~~

A.3

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:~~

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.~~

L.3

SR 3.9.1.1
SR 3.9.1.2 4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined ~~by chemical analysis~~ at least once per 72 hours.

LA.2

DISCUSSION OF CHANGES
ITS 3.9.1, BORON CONCENTRATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 3.9.1 provides requirements on the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal. ITS 3.9.1 provides requirements on the boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity. This changes the CTS by explicitly including the refueling cavity in the volumes required to have boron concentration maintained.

This change is acceptable because the technical requirements have not changed. The refueling cavity is considered to be governed by the CTS requirements because the refueling cavity is typically connected to the RCS, the refueling canal, or both. This change is designated as administrative because the technical requirements of the specifications have not changed.

A.3 CTS 3.9.1 Action b contains the statement, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.1 does not contain an equivalent statement. This changes the CTS by deleting the Specification 3.0.3 exception.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.1 Action a requires the immediate suspension of positive reactivity changes "except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2" (i.e., 2400 ppm). ITS 3.9.1 Required Action A.2 requires positive reactivity additions to be suspended, but does not provide any allowance for positive reactivity changes due to the addition of water from the RWST to continue. This changes the CTS by removing the allowance to allow a positive reactivity change from the addition of water from the RWST, provided the boron concentration of the RWST is greater than 2400 ppm.

The purpose of CTS 3.9.1 Action a is to provide assurance that an inadvertent criticality will not result when the boron concentration is not within limits in MODE 6. The CTS 3.9.1 Action requires the suspension of all operations involving CORE ALTERATIONS or positive reactivity changes and initiation of activities to restore boron concentration to within its limit. However, allowing a

DISCUSSION OF CHANGES
ITS 3.9.1, BORON CONCENTRATION

positive reactivity addition conflicts with the requirement to restore boron concentration to its limit. Therefore, this exception is deleted. This change is acceptable because the ITS requires actions that provide assurance that an inadvertent criticality will not result while boron concentration is not within limits in MODE 6, and requires initiation of activities to restore boron concentration to within its limit. This change is designated as more restrictive because it provides more restrictive corrective actions in the ITS than in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.9.1 states that the boron concentration in MODE 6 shall be the more restrictive reactivity condition of a k_{eff} of 0.95 or less or a boron concentration of $\geq$ 2400 ppm. ITS LCO 3.9.1 states that the boron concentration shall be within the limit specified in the COLR. This changes the CTS by relocating the MODE 6 boron concentration limit, which must be confirmed on a cycle-specific basis, to the CORE OPERATING LIMITS REPORT (COLR).

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that 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 requirements and Surveillances that verify that the cycle-specific parameter limits are being met. ITS 3.9.1 continues to require that boron concentration limit is met. ITS SR 3.9.1.1 requires periodic verification that boron concentration is within the limits provided in the COLR. The method of determining or utilizing the boron concentration limit has not changed. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.9.1.2 requires that the boron concentration of the Reactor Coolant System and the refueling canal be determined "by chemical analysis" at least once per 72 hours. ITS SR 3.9.1.1 and SR 3.9.1.2 require verification that boron concentration is within the limit specified in the COLR. ITS

DISCUSSION OF CHANGES
ITS 3.9.1, BORON CONCENTRATION

SR 3.9.1.1 and SR 3.9.1.2 do not specify that the boron concentration be determined by chemical analysis. This changes the CTS by moving details of how the boron concentration is determined from the CTS to the Bases.

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

LESS RESTRICTIVE CHANGES

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.9.1 provides limits on the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal when in MODE 6. ITS 3.9.1 modifies this requirement with a Note which states "Only applicable to the refueling canal and refueling cavity when connected to the RCS." This changes the CTS by eliminating the applicability of the boron concentration limits on the refueling canal and refueling cavity when those volumes are not connected to the RCS. In addition, ITS SR 3.9.1.2 requires a verification that the boron is within the limit specified in the COLR once within 72 hours prior to connecting the refueling canal and refueling cavity to the RCS.

The purpose of CTS 3.9.1 is to ensure the boron concentration of the water surrounding the reactor fuel is sufficient to maintain the required SHUTDOWN MARGIN. This change is acceptable because the requirements continue to ensure that process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. If the refueling canal and refueling cavity are not connected to the RCS (such as when the reactor vessel head is on the reactor vessel), the boron concentration of those volumes cannot affect the SHUTDOWN MARGIN. In addition, prior to connecting the refueling canal and refueling cavity to the RCS, a boron concentration verification will be performed to ensure the newly connected portions cannot decrease the boron concentration below the limit. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.9.1 Action a states that when the boron concentration requirement is not met, initiate and continue boration at $\geq$ 34 gpm of 6,550 ppm boric acid solution or its equivalent until k_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 2400 ppm, whichever is the more restrictive. ITS 3.9.1 Required Action A.3 requires initiation of action to restore boron concentration to within limit. This changes the

DISCUSSION OF CHANGES
ITS 3.9.1, BORON CONCENTRATION

CTS by eliminating the specific requirements for the boric acid solution to be used to restore compliance with the LCO.

The purpose of CTS 3.9.1 Action a is to restore the required SHUTDOWN MARGIN in a timely manner. 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. Specifying the boric acid solution requirements in the Action is not necessary, since the ITS requires that action to restore the boron concentration be initiated immediately. This prompt action will result in the boron concentration being restored as quickly, or more quickly, than the CTS requirement. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.9.1.1 requires the LCO reactivity condition to be determined prior to removing or unbolting the reactor vessel head, and prior to withdrawal of any full length control rod in excess of 3 feet from its fully inserted position. ITS 3.9.1 does not contain this Surveillance Requirement.

The purpose of CTS 4.9.1.1 is to ensure that the LCO requirements are met prior to entering MODE 6 and that the reactor has sufficient SHUTDOWN MARGIN prior to withdrawing any control rods. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the values used to meet the LCO are consistent with the safety analyses. Thus, appropriate values continue to be tested in a manner and at a frequency necessary to give confidence that the assumptions in the safety analyses are protected. ITS 3.9.1 requires that the boron concentration be met in MODE 6 or that action be immediately initiated to restore the boron concentration and that all positive reactivity additions be suspended. Therefore, verification that the boron concentration requirement is met must be performed prior to entering MODE 6 in order to avoid immediately entering into an Action and withdrawal of control rods is prohibited when the boron concentration requirement is not met. While the CTS Surveillance is not required, the level of protection provided is appropriate. This change is designated as less restrictive because Surveillances required in the CTS will not be required in the ITS.

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

CTS

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

(RCS) ①

3.9.1 LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

\- NOTE -
Only applicable to the refueling canal ~~and refueling canal~~ and refueling cavity when connected to the RCS. ①

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

Action a

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

4.9.1.2

INSERT 1 ②

2 **INSERT 1**

SR 3.9.1.2	Verify boron concentration of refueling canal and refueling cavity is within the limit specified in the COLR.	Once within 72 hours prior to connecting the refueling canal and refueling cavity to the RCS

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.1, BORON CONCENTRATION

1. Typographical/grammatical error corrected.

2. ISTS SR 3.9.1.1 requires a verification that the boron concentration is within limit every 72 hours. The Bases for the SR states that prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. SR 3.0.4 requires the SR to be met prior to entering a MODE or other specified condition in the Applicability. However, SR 3.0.4 is only applicable in MODES 1, 2, 3, and 4; it is not applicable in MODE 6, the MODE in which ISTS 3.9.1 is applicable. Therefore, to meet the intent of the Bases requirement, a new SR has been added, SR 3.9.1.2, which requires a verification that the boron concentration of the refueling canal and refueling cavity is within the limit specified in the COLR once within 72 hours prior to connecting the refueling canal and refueling cavity to the RCS.

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

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

INSERT 1 GDC 26 of 10 CFR ~~50~~, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration. 1

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is
5 slowly removed to form the refueling cavity. The refueling ~~canal and the~~ 3
refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling
4 (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and 5 4
Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

1 **INSERT 1**

Plant Specific Design Criterion (PSDC) 27 requires that two independent reactivity control systems, preferably of different design principles, be provided. According to PSDC 28 (Ref. 1), the reactivity controls must be capable of making and holding the core subcritical from any hot standby or hot operating condition.

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BASES

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

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. As a result, the soluble boron concentration is relatively the same in each of these volumes.

~~The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."~~ 7

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal ~~and the refueling cavity~~ while in MODE 6. including the transfer canal 3 The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the ~~Reactor Coolant System~~ RCS 6. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

(i.e., are in communication with) 5

BASES

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action. (8)

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1 and SR 3.9.1.2

These SRs ensure that the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity or canal were disconnected from the RCS, this (5) (7)

and

and

INSERT A
Move to next page

Part of INSERT A

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR ensures the~~ correct boron concentration is verified prior to communication with the RCS. (7)

The ~~A minimum~~ SR 3.9.1.1 Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate. (7)

The SR 3.9.1.2 Frequency of once within 72 hours prior to connecting the refueling canal and refueling cavity to the RCS ensures that

INSERT A from previous page

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26.~~ UFSAR, Section 1.4.5 (1)
2. ~~FSAR, Chapter [15].~~ (3) (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.1 BASES, BORON CONCENTRATION

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A, criteria have been replaced with references to the appropriate section of the UFSAR.
2. The brackets have been removed and the proper plant specific information/value has been provided.
3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
4. Changes are made to be consistent with changes made to the Specification.
5. Editorial change made for clarity.
6. Changes have been made to be consistent with similar words in other places in the ITS Bases.
7. Changes made to be consistent with the Specification.
8. Typographical/grammatical error corrected.
9. The paragraph and associated reference have been deleted since it is discussing a MODE 5 analysis, and this Specification is applicable in MODE 6.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.1, BORON CONCENTRATION

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 2

ITS 3.9.2, Nuclear Instrumentation

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

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

OPERABLE M.1

LCO 3.9.2 3.9.2 As a minimum, two source range neutron flux monitors shall be ~~operating~~, ~~each with continuous visual indication in the control room~~ and one with audible count rate circuit to be OPERABLE. LA.1

APPLICABILITY: MODE 6.

ACTION: L.2

ACTION A a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2. M.2

b. The provisions of Specification 3.0.3 are not applicable. A.2

Add proposed ACTION B M.3

Add proposed ACTION C L.3

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

SR 3.9.2.1 a. A CHANNEL CHECK at least once per 12 hours.

SR 3.9.2.2 b. A CHANNEL CALIBRATION at least every ~~18~~ 24 months.* L.4

Note to SR 3.9.2.2 * Neutron detectors may be excluded from CHANNEL CALIBRATION

ITS 3.9.2

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

OPERABLE — M.1

LCO 3.9.2 3.9.2 As a minimum, two source range neutron flux monitors shall be ~~operating~~ each with continuous visual indication in the control room and one with audible count rate circuit to be OPERABLE. — LA.1

APPLICABILITY: MODE 6.

ACTION: — L.2

ACTION A a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3/4.2.7.b.2. — M.2

b. The provisions of Specification 3.0.3 are not applicable. — A.2

Add proposed ACTION B — M.3

Add proposed ACTION C — L.3

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

SR 3.9.2.1 a. A CHANNEL CHECK at least once per 12 hours.

24 — L.4

SR 3.9.2.2 b. A CHANNEL CALIBRATION at least every 18 months.*

Note to SR 3.9.2.2 * Neutron detectors may be excluded from CHANNEL CALIBRATION

DISCUSSION OF CHANGES
ITS 3.9.2, NUCLEAR INSTRUMENTATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 3.9.2 Action b contains the statement, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.2 does not contain an equivalent statement. This changes the CTS by deleting the Specification 3.0.3 exception.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.2 states, in part, that two source range neutron flux monitors shall be "operating." ITS 3.9.2 states, in part, that two source range neutron flux monitors shall be "OPERABLE." This changes the CTS by requiring the source range neutron flux monitors to be OPERABLE, instead of just operating.

The purpose of CTS 3.9.2 is to ensure that the source range neutron flux monitors are capable of performing the safety functions assumed in the accident analysis. However, as written, the CTS LCO wording could be interpreted to allow the source range neutron flux monitors to be operating in a location or condition that would prevent them from performing the assumed safety function. The ITS wording eliminates this possible misinterpretation. This change is acceptable because the source range neutron flux monitors must be OPERABLE (i.e., capable of performing their safety function) instead of just operating. This change is designated as more restrictive because the ITS contains more specific requirements for a specific component.

M.2 CTS 3.9.1 Action a requires the immediate suspension of positive reactivity changes except for the addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2 (i.e., 2400 ppm). ITS 3.9.2 Required Action A.2 requires suspension of operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1. This changes the CTS by replacing the allowance to allow a positive reactivity change from the addition of water from the RWST, provided the boron concentration of the RWST is greater than 2400 ppm with a requirement that the boron concentration must meet the boron concentration of LCO 3.9.1.

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The purpose of CTS 3.9.2 Action a is to provide assurance that activities that could result in reducing boron concentration such that the required SHUTDOWN MARGIN is not met will not occur when any source range neutron flux monitor is inoperable in MODE 6. Allowing positive reactivity additions from sources with boron concentrations meeting the requirements of ITS 3.9.1 preserves the required SHUTDOWN MARGIN. This change is acceptable because the ITS requires actions that prohibit activities that could result in reducing boron concentration such that the required SHUTDOWN MARGIN is not met. This change is designated as more restrictive because it provides more restrictive corrective actions in the ITS than in the CTS.

M.3 CTS 3.9.2 Action a states that with fewer than two source range channels operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2 (i.e., 2400 ppm). The ITS provides similar ACTIONS as the CTS (except where changed as described in DOCs M.2 and L.2). In addition, ITS 3.9.2 ACTION B requires additional actions when two source range neutron flux monitors are inoperable. The ITS requires immediate initiation of action to restore one source range neutron flux monitor to OPERABLE status and to perform a verification of boron concentration (per ITS SR 3.9.1.1) once per 12 hours. This changes the CTS requirements by requiring an additional verification of boron concentration every 12 hours when both source ranges are inoperable and by requiring an additional action to initiate immediate action to restore one source range neutron flux monitor to OPERABLE status.

The purpose of this change is to provide necessary Required Actions that are appropriate for a possible condition that could be encountered. This change is acceptable because the proposed Required Actions are reasonable and necessary to ensure the reactor is maintained in a safe condition. This change is more restrictive because it provides for additional actions that the CTS does not require.

M.4 Not used.

M.5 Not used.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.9.2 states that two source range neutron flux monitors shall be operating, "each with continuous visual indication in the control room." ITS 3.9.2 LCO states that two source range neutron flux monitors shall be OPERABLE. This changes the CTS by moving the requirement that each

DISCUSSION OF CHANGES
ITS 3.9.2, NUCLEAR INSTRUMENTATION

channel has a continuous visual indication in the control room from the CTS to the Bases.

The removal of this detail, which is 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 retains the requirement that two channels be OPERABLE and continues to require the associated Surveillance to verify OPERABILITY. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 Not used.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.9.2 Action a states that with fewer than two source range neutron flux monitors operating, immediately suspend all operations involving positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2 (i.e., 2400 ppm). ITS 3.9.2 Required Action A.2 states "Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1, Boron Concentration." This allows positive reactivity changes provided they do not reduce the boron concentration below the refueling limit. This changes the CTS requirements by allowing limited positive reactivity additions from sources in addition to the RWST.

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 DBA occurring during the repair period. The requirement to maintain refueling boron concentration within limits will continue to ensure the unit will be operated within the assumptions of the safety analyses. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 4 – Relaxation of Required Action)* CTS 3.9.2 Action a requires the immediate suspension of CORE ALTERATIONS or positive reactivity changes except for the addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2, in the event one source range neutron flux monitor with

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audible indication in the containment is not operating. ITS 3.9.2 ACTION C requires initiation of action to isolate unborated water sources in the event the required source range audible count rate circuit is inoperable. This changes the CTS by replacing the Action to immediately suspend CORE ALTERATIONS or positive reactivity changes except for the addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by CTS 3.1.2.7.b.2, in the event one source range monitor with audible indication in the containment is not operating, with the Action to initiate action to isolate unborated water sources.

The purpose of CTS 3.9.2 Action a is to provide assurance that activities that could result in an inadvertent criticality will not occur when the required source range audible count rate circuit is inoperable in MODE 6. 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 DBA occurring during the repair period. ITS 3.9.2 ACTION C requires actions to be taken to isolate sources of unborated water. This provides assurance that rapid dilution of boron concentration, which could result in rapid reduction in shutdown margin, will not occur. This change preserves the assumptions and conclusions of the boron dilution analysis. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.4 *(Category 11 – 18 to 24 Month Surveillance Frequency Change, Channel Calibration Type)* CTS 4.9.2.b requires a CHANNEL CALIBRATION of each source range neutron flux monitor every 18 months. ITS SR 3.9.2.2 requires the performance of a CHANNEL CALIBRATION every 24 months. This changes the CTS by extending the Frequency of the Surveillance from 18 months (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and ITS SR 3.0.2) to 24 months (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and ITS SR 3.0.2).

The purpose of the CHANNEL CALIBRATION required by CTS 4.9.2.b is to ensure the source range neutron flux monitor will function correctly to ensure the safety analysis can be met. Extending the SR Frequency is acceptable because the source range neutron flux monitoring channels are designed to be highly reliable. Furthermore, a CHANNEL CHECK for the source range neutron flux monitoring channels is performed on a more frequent basis (ITS SR 3.9.2.1). The CHANNEL CHECK provides a qualitative demonstration of the OPERABILITY of the instrument.

This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. The

DISCUSSION OF CHANGES
ITS 3.9.2, NUCLEAR INSTRUMENTATION

impacted source range neutron flux monitoring instrumentation was evaluated through a failure analysis and a qualitative drift analysis:

Westinghouse Source Range, Neutron Flux

This function is performed by SRM Neutron Flux Detectors (Westinghouse Model WL-23706), SRM Neutron Flux Drawers (Westinghouse Model 6051D50G01), a Weschler HX-252 Indicator, and a Tracor Westronics Recorder (Model 4200 (Unit 1) and Model 4220 (Unit 2)). These system components were not evaluated for drift but were justified for extension based on engineering judgment. SRMs satisfy their design function if calibration is sufficient to ensure neutron level is observable when the reactor is shutdown. This is verified by CHANNEL CHECKS at least every 12 hours when the reactor is shutdown. The SRMs must be operational in MODE 6. SRM response to reactivity changes is distinctive and well known to plant operators, and SRM response is closely monitored during these reactivity changes. Additionally, since there is very little neutron activity during loading, refueling, shutdown, and approach to criticality, a neutron source is placed in the reactor during approach to criticality to provide a minimum observable SRM neutron count rate attributable to core neutrons of at least 2 counts per second. During plant shutdowns and startups, overlap between the IRM channels and the SRM channels is routinely verified to ensure performance of the SRM channels. There is also more frequent testing, including a COT every 184 days in MODES 1 and 2 and every 31 days in MODES 3, 4, and 5, to verify operation of the electronics for the source range trip. Therefore, any substantial degradation of the SRMs will be evident and long term drift has no impact on the accuracy of this circuit. The results of these analyses will support a 24 month Surveillance interval.

Thermo Gamma-Metrics Neutron Flux Monitors

The function is performed by a Wide Range Detector Assemblies (Gamma Metrics model numbers 200749-103 and 200574-11), Signal Processing Drawers (Gamma Metrics model numbers 900180-101 and 900091-101), a Weschler HX-252 Indicator, and a Tracor Westronics Recorder (Series 4200). These system components were not evaluated for drift but were justified for extension based on engineering judgment. SRMs satisfy their design function if calibration is sufficient to ensure neutron level is observable when the reactor is shutdown. This is verified by CHANNEL CHECKS at least every 12 hours when the reactor is shutdown. The SRMs must be operational in MODE 6. SRM response to reactivity changes is distinctive and well known to plant operators, and SRM response is closely monitored during these reactivity changes. Therefore, any substantial degradation of the SRMs will be evident and long term drift has no impact on the accuracy of this circuit. The results of these analyses will support a 24 month Surveillance interval.

Based on the design of the instrumentation and the qualitative drift evaluations, it is concluded that the impact, if any, from this change on system availability is minimal. A review of the Surveillance test history was performed to validate the above conclusion. Those tests that were classified as failures were evaluated and primarily involved components found with out of tolerance calibration data. The other failures were reviewed and those failures did not invalidate the

DISCUSSION OF CHANGES
ITS 3.9.2, NUCLEAR INSTRUMENTATION

conclusion that the impact, if any, on system availability from this change is minimal. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (30 months) does not invalidate any assumptions in the unit licensing basis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

CTS

3.9 REFUELING OPERATIONS ①

3.9.3 ② Nuclear Instrumentation

LCO 3.9.2 — Two source range neutron flux monitors shall be OPERABLE. ①

3.9.2

AND

One source range audible ~~alarm~~ count rate circuit shall be OPERABLE. ②

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend operations that would cause introduction of coolant into the RCS ~~coolant~~ with boron concentration less than required to meet the boron concentration of LCO 3.9.1 "Boron Concentration."	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

Action a

DOC M.2

② ④ ③ ②

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 ①

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~- REVIEWER'S NOTE - Condition C is included only for plants that assume a boron dilution event is mitigated by operator response to an audible source range indication.~~ DOC L3	C.1 Initiate action to isolate unborated water sources.	Immediately. ③
C. Required source range audible alarm count rate circuit inoperable. ②		

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.9.2.a SR 3.9.~~3~~.1 ②	Perform CHANNEL CHECK.	12 hours ①
4.9.2.b SR 3.9.~~3~~.2	- NOTE - Neutron detectors are excluded from CHANNEL CALIBRATION. Perform CHANNEL CALIBRATION.	~~[18]~~ 24 months ②

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.2, NUCLEAR INSTRUMENTATION

1. CNP has analyzed a boron dilution event in MODE 6. Therefore, ISTS 3.9.2 is not included in the ITS and ISTS 3.9.3 is renumbered as ITS 3.9.2.
2. The brackets are removed and the proper plant specific information/value is provided.
3. Editorial correction to be consistent with the format of the ITS.
4. Changes have been made to be consistent with changes made to another Specification.

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

 1

B 3.9 REFUELING OPERATIONS

B 3.9.~~3~~ 2 Nuclear Instrumentation 1

BASES

BACKGROUND

~~- REVIEWER'S NOTE -~~
~~Bracketed options are provided for source range OPERABILITY requirements to include audible alarm or count rate function. These options apply to plants that assume a boron dilution event that is mitigated by operator response to an audible indication. For plants that isolate all boron dilution paths (per LCO 3.9.2), the source range OPERABILITY includes only a visual monitoring function.~~ 2

INSERT 1

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The ~~installed~~ source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. INSERT 2 3 Westinghouse types of

The ~~installed~~ source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) ~~with a [5]% instrument accuracy~~. The detectors also provide continuous visual indication in the control room ~~[~~and an audible ~~[alarm]~~ ~~[~~count rate~~]~~ to alert operators to a possible dilution accident~~]~~. The NIS is designed in accordance with the criteria presented in Reference 1. 3 4 2

3 INSERT 3

INSERT 4 3

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. ~~[~~The audible count rate from the source range neutron flux monitors provides prompt and definite ~~indiction~~ indication of any boron dilution. The count rate increase is proportional to the subcritical multiplication factor and allows operators to promptly recognize the initiation of a boron dilution event. Prompt recognition of the initiation of a boron dilution event is consistent with the assumptions of the safety analysis and is necessary to assure sufficient time is available for isolation of the primary water makeup source before SHUTDOWN MARGIN is lost (Ref. 2)~~]~~. 2 5 2

3 **INSERT 1**

(i.e., the Westinghouse source range neutron flux monitors and the Thermo Gamma-Metrics neutron flux monitors)

3 **INSERT 2**

The Thermo Gamma-Metrics neutron flux monitors are part of the Thermo Gamma-Metrics Neutron Flux Monitoring System. Both of

3 **INSERT 3**

(selectable between proportional source range neutron flux monitors)

3 **INSERT 4**

There are two Thermo Gamma-Metrics neutron flux monitors. Each monitor includes two fission chamber detectors capable of monitoring a wide range from source level (shutdown) to full power reactor operation. In the source range, the detectors monitor the neutron flux in counts per second and are capable of detecting six decades of neutron flux. The detectors also provide continuous visual indication in the control room of source count rate and a source rate of change.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~- REVIEWER'S NOTE -~~
~~The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."~~ 2

The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that two source range neutron flux monitors INSERT 5 3 be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication ~~in the control room~~. ~~In addition, at least one of~~ ~~the two~~ monitors must provide an OPERABLE audible ~~alarm~~ count rate function to alert the operators to the initiation of a boron dilution event. 2 in the control room INSERT 6 3

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," and LCO 3.3.~~9~~, "~~BDPS~~." as applicable 11 8 Boron Dilution Monitoring Instrumentation (BDMI) 2

ACTIONS

A.1 and A.2 1

With only one required source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

3 **INSERT 5**

(any combination of Westinghouse source range neutron flux and Thermo Gamma-Metrics neutron flux monitors)

3 **INSERT 6**

(which must be a Westinghouse source range neutron flux monitor, since the Thermo Gamma-Metrics neutron flux monitors do not have an audible count rate function)

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BASES

ACTIONS (continued)

B.1 1

With no required source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2 1

With no required source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

C.1 2

With no audible [alarm] [count rate] OPERABLE, prompt and definite indication of a boron dilution event, consistent with the assumptions of the safety analysis, is lost. In this situation, the boron dilution event may not be detected quickly enough to assure sufficient time is available for operators to manually isolate the unborated water source and stop the dilution prior to the loss of SHUTDOWN MARGIN. Therefore, action must be taken to prevent an inadvertent boron dilution event from occurring. This is accomplished by isolating all the unborated water flow paths to the Reactor Coolant System. Isolating these flow paths ensures that an inadvertent dilution of the reactor coolant boron concentration is prevented. The Completion Time of "Immediately" assures a prompt response by operations and requires an operator to initiate actions to isolate an affected flow path immediately. Once actions are initiated, they must be continued until all the necessary flow paths are isolated or the circuit is restored to OPERABLE status. 2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (2)

SR 3.9.3.1 (2) is the performance of a CHANNEL CHECK, which is a Normally (1) comparison of the parameter indicated on one channel to a similar parameter on other channel~~s~~ (an) (1). It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2 (2) (1)

INSERT 7 (7)

SR 3.9.3.2 (2) is the performance of a CHANNEL CALIBRATION every ~~18~~ 24 months. ~~This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION.~~ The CHANNEL CALIBRATION for the source range neutron flux monitors ~~consists of~~ also includes (3) obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. (3) Westinghouse In addition The CHANNEL CALIBRATION ~~also~~ includes verification of the audible ~~(alarm)~~ count rate function (2). The ~~18~~ 24 (1) month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the ~~18~~ month Frequency. for the required monitor (1)

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29~~. UFSAR Section 1.4.5 (6)
2. UFSAR, Section ~~15.2.4~~ 14.1.5. (3) (2)

7 **INSERT 7**

CHANNEL CALIBRATION is a complete check of the instrument loop, except the detector.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.9.2 BASES, NUCLEAR INSTRUMENTATION

1. Changes are made to reflect those changes made to the ISTS. Subsequent requirements are renumbered or revised, where applicable, to reflect the changes.

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

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

4. The specific accuracy of the source range neutron flux monitors is not part of the licensing basis of CNP and has been deleted.

5. Typographical/grammatical error corrected.

6. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A, criteria have been replaced with references to the appropriate section of the UFSAR.

7. Changes are made to be consistent with similar words in other places in the ITS Bases.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.2, NUCLEAR INSTRUMENTATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 3

ITS 3.9.3, Containment Penetrations

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

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

LCO 3.9.3 — 3.9.4 The containment building penetrations shall be in the following status:

LCO 3.9.3.a — a. The equipment door closed and held in place by a minimum of four bolts,

b. The airlock doors are controlled in the following manner:

LCO 3.9.3.b — 1. A minimum of one door in each airlock is closed, or

2. Both airlock doors may be open provided:*

a. One door in each airlock is OPERABLE*, — A.2

b. ~~Refueling cavity level is greater than 23 feet above the fuel, and~~ — A.3

c. ~~A designated individual is available at all times to close the airlock if required.~~ — LA.1

LCO 3.9.3.c — c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or

2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve.

NOTE

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere via the auxiliary building vent may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment. — L.1

ACTION:

ACTION A — With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. ~~The provisions of Specification 3.0.3 are not applicable.~~ — A.4

SURVEILLANCE REQUIREMENTS — L.2

SR 3.9.3.1, SR 3.9.3.2 — 4.9.4 Each of the above required containment building penetrations shall be determined to be in its required status within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by: — L.1 — L.3

* ~~For the purpose of this Specification, an OPERABLE airlock door is a door that is capable of being closed and secured. Cables or hoses transversing the airlock shall be designed to allow for removal in a timely manner (e.g., quick disconnects).~~ — LA.1

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.9.3.1 a. Verifying the penetrations are in the required status, or

SR 3.9.3.2 b. Testing the Containment Purge and Exhaust isolation valves per the applicable portions of Specification 4.6.3.1.2.

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ITS 3.9.3

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.6 CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

See ITS 3.6.3

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

Add proposed SR 3.9.3.2 Note — L.3

SR 3.9.3.2

c. Verifying that on a ~~Containment Purge and Exhaust~~ actual or simulated isolation signal, each Purge and Exhaust valve actuates to its isolation position. — L.4

See ITS 3.6.3

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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ITS 3.9.3

A.1

ITS

REFUELING OPERATIONS

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.9.3.c.2 3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE. L.1

APPLICABILITY: During ~~Core Alterations or~~ movement of irradiated fuel within the containment.

ACTION:

LCO 3.9.3.c.1 With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. ~~The provision of Specification 3.0.3 are not applicable.~~ A.4

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2 4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE ~~within 100 hours prior to the start of and~~ at least ~~once per 7 days~~ ~~during CORE ALTERATIONS~~ by verifying that containment Purge and Exhaust isolation occurs on ~~manual initiation and on a high radiation signal from each of the containment radiation instrumentation monitors.~~ L.2 L.1 L.3

See ITS 3.3.6

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

LCO 3.9.3 — 3.9.4 The containment building penetrations shall be in the following status:

LCO 3.9.3.a — a. The equipment door closed and held in place by a minimum of four bolts,

b. The airlock doors are controlled in the following manner:

LCO 3.9.3.b

1. A minimum of one door in each airlock is closed, or — A.2

2. Both airlock doors may be open provided:

a. One door in each airlock is OPERABLE*,

~~b. Refueling cavity level is greater than 23 feet above the fuel, and~~ — A.3

~~c. A designated individual is available at all times to close the airlock if required.~~ — LA.1

LCO 3.9.3.c — c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or

2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve.

---NOTE---
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere via the auxiliary building vent may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment. — L.1

ACTION:

ACTION A — With the requirements of the above specification not satisfied, immediately suspend ~~all operations involving CORE ALTERATIONS or~~ movement of irradiated fuel in the containment building. ~~The provisions of Specification 3.0.3 are not applicable.~~ — A.4

SURVEILLANCE REQUIREMENTS — L.2

SR 3.9.3.1, SR 3.9.3.2 — 4.9.4 Each of the above required containment building penetrations shall be determined to be in its required status ~~within 100 hours prior to the start of and~~ at least once per 7 days during ~~CORE ALTERATIONS or~~ movement of irradiated fuel in the containment building by: — L.1, L.3

~~* For the purpose of this Specification, an OPERABLE airlock door is a door that is capable of being closed and secured. Cables or hoses transversing the airlock shall be designed to allow for removal in a timely manner (e.g., quick disconnects).~~ — LA.1

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

CONTAINMENT PENETRATIONS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.9.3.1 a. Verifying the penetrations are in the required status, or

SR 3.9.3.2 b. Testing the Containment Purge and Exhaust isolation valves per the applicable portions of Specification 4.6.3.1.2.

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ITS 3.9.3

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.6 CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

See ITS 3.6.3

4.6.3.1.2 Each containment isolation valve specified shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

Add proposed SR 3.9.3.2 Note — L.3

SR 3.9.3.2 c. Verifying that on a Containment Purge and Exhaust actual or simulated isolation signal, each Purge and Exhaust valve actuates to its isolation position. — L.4

See ITS 3.6.3

4.6.3.1.3.1 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

ITS 3.9.3

A.1

ITS

REFUELING OPERATIONS

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.9.3.c.2 3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

L.1

APPLICABILITY: During ~~Core Alterations or~~ movement of irradiated fuel within the containment.

ACTION:

LCO 3.9.3.c.1 With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. ~~The provisions of Specification 3.0.3 are not applicable.~~ A.4

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2 4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE ~~within 100 hours prior to the start of and~~ at least once per 7 days ~~during CORE ALTERATIONS~~ by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

L.2

L.1

See ITS 3.3.6

L.3

DISCUSSION OF CHANGES
ITS 3.9.3, CONTAINMENT PENETRATIONS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 3.9.4.b requires a minimum of one door in each airlock to be closed or allows both airlock doors to be open provided one door in each airlock is OPERABLE, refueling cavity level is greater than 23 feet above the fuel, and a designated individual is available at all times to close the airlock if required. A footnote associated with CTS 3.9.4.b clarifies that for the purpose of this Specification, an OPERABLE air lock door is a door that is capable of being closed and secured. ITS 3.9.3 requires that one door in each air lock is capable of being closed. This changes the CTS by replacing the prescriptive requirements for control of the air lock doors with a more general requirement that the air lock doors must be capable of being closed. Other aspects of this change are discussed in DOC A.3 and DOC LA.1.

This change is acceptable because the CTS requirements have not changed. A door that is closed is a door that is also capable of being closed. The ITS requirements preserve the intent of the CTS. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.9.4.b.2.b allows both airlock doors to be open provided, in part, that the refueling cavity level is greater than 23 feet above the fuel. ITS 3.9.3 does not contain this restriction.

This change is acceptable because the requirement is duplicative of the requirements of ITS LCO 3.9.6, which requires that refueling cavity water level be maintained ≥ 23 feet above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 The CTS 3.9.4 and CTS 3.9.9 Actions state "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.3 does not include this statement. This changes CTS by deleting the Specification 3.0.3 exception.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

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DISCUSSION OF CHANGES
ITS 3.9.3, CONTAINMENT PENETRATIONS

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.9.4.b.2.c allows both doors of each airlock to be open provided, in part, that a designated individual is available at all times to close an airlock door if required. A footnote associated with CTS 3.9.4.b clarifies that for the purpose of this Specification, an OPERABLE airlock door is a door that is capable of being closed and secured. The footnote also states that cables or hoses transversing the airlock shall be designed to allow for removal in a timely manner (e.g., quick disconnects). ITS 3.9.3.b requires that one door in each air lock is capable of being closed, but does not provide the level of description provided in the CTS. This changes the CTS by moving the requirement for a designated individual and the details on cables or hoses that transverse the air lock from the CTS to the Bases.

The removal of these details for compliance with the LCO from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the one door in each air lock be capable of being closed. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.9.4 and CTS 3.9.9 are applicable during CORE ALTERATIONS and movement of irradiated fuel within the containment. ITS 3.9.3 is applicable during movement of irradiated fuel assemblies within containment. References to CORE ALTERATIONS in CTS 3.9.4 are eliminated in the Applicability, Action, and Surveillances. References to CORE ALTERATIONS in CTS 3.9.9 are eliminated in the Applicability and Surveillances. This changes the CTS by eliminating requirements for containment closure and the Containment Purge and Exhaust Isolation System during CORE ALTERATIONS.

The purpose of CTS 3.9.4 is to ensure the containment penetrations are in the condition assumed in the Fuel Handling Accident (FHA) inside containment analysis. The purpose of CTS 3.9.9 is to ensure the containment purge supply and exhaust valves are capable of being closed as assumed in the FHA inside containment 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. There are no accidents postulated to occur during CORE ALTERATIONS that result in significant radioactive release except a FHA. The analysis for a FHA assumes that the accident is initiated only by movement of irradiated fuel. Therefore, imposing requirements during CORE ALTERATIONS in addition to during movement of irradiated fuel is unnecessary. This change is designated as less restrictive because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.9.4 states that specified containment penetration Surveillances shall be performed, in part, "within 100 hours prior to the start of" the specified conditions in the Applicability. ITS SR 3.9.3.1 and ITS SR 3.9.3.2 do not include the "within 100 hours prior to the start of" Frequency. ITS SR 3.0.1 states "SRs shall be met during the MODES or other specified conditions in the Applicability for the individual LCOs, unless otherwise stated in the SR." Therefore, the ITS requires that the Surveillances must be met prior to the initiation of movement of irradiated fuel. This changes the CTS by eliminating the stipulation that the Surveillances be met within 100 hours prior to entering the conditions specified in the Applicability.

The purpose of CTS 4.9.4 is to verify the equipment required to meet the LCO is OPERABLE. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. For CTS 4.9.4, the periodic Surveillance Frequency for verifying containment penetrations are in the required status is acceptable during the conditions specified in the Applicability, and is also acceptable during the period prior to entering the conditions specified in the Applicability. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.3 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.9.4 and CTS 4.9.9 include a Surveillance Frequency of "once per 7 days" during conditions specified in the Applicability for performing Surveillance of the Containment Purge Supply and Exhaust System. The ITS SR 3.9.3.2 Frequency for the same requirement is 24 months. ITS SR 3.9.3.2 is also modified by a Note that states that SR 3.9.3.2 is not required to be met for containment purge supply and exhaust valve(s) in penetrations that are closed to comply with LCO 3.9.3.c.1. This changes the CTS by changing the Surveillance Frequency from 7 days to 24 months and adding the Note that the SR is not required to be met for containment purge supply and exhaust valve(s) in penetrations that are closed to comply with ITS LCO 3.9.3.c.1.

The purpose of CTS 4.9.4 and CTS 4.9.9 is to verify the equipment required to meet the LCO is OPERABLE. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Containment purge supply and exhaust valve testing is still required, but at a Frequency consistent with the testing Frequency for containment isolation valves required in MODES 1, 2, 3, and 4. This Frequency provides an appropriate degree of assurance that the valves are

DISCUSSION OF CHANGES
ITS 3.9.3, CONTAINMENT PENETRATIONS

OPERABLE. When containment purge supply and exhaust valve(s) in penetrations are closed to comply with ITS LCO 3.9.3.c.1, the penetrations are in the expected condition (isolated) to mitigate the effects of a fuel handling accident inside containment. Therefore, there is no need for the actuation signal to reposition the valves to the closed position. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.4 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.6.3.1.2.c requires verification of the automatic actuation of the Containment Purge and Exhaust valves on a Containment Purge and Exhaust isolation signal (i.e., a test signal). ITS SR 3.9.3.2 specifies 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 4.6.3.1.2.c is to ensure that the containment purge and exhaust valves operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment can not 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.

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

CTS

3.9 REFUELING OPERATIONS ①

3.9.~~4~~ 3 Containment Penetrations

LCO 3.9.~~4~~ 3 The containment penetrations shall be in the following status: ①

3.9.4.a a. The equipment is hatch closed and held in place by four bolts, ③ ② ⑥

3.9.4.b b. One door in each air lock is capable of being closed, and ② ⑥

3.9.4.c c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:

3.9.4.c.1, 3.9.9 Action 1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or ③ ⑥

3.9.4.c.2, 3.9.9 2. Capable of being closed by an OPERABLE Containment Purge Supply and Exhaust ~~Isolation~~ System. ⑤

- NOTE -

3.9.4 Note Penetration flow path(s) providing direct access from the containment atmosphere via the auxiliary building vent to the outside atmosphere may be unisolated under administrative controls. ④

APPLICABILITY: During movement of ~~recently~~ irradiated fuel assemblies within containment. ②

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of ~~recently~~ irradiated fuel assemblies within containment.	Immediately

Action ②

 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.4	SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days	1
4.6.3.1.2.c, 4.9.4, 4.9.9	SR 3.9.3.2	- NOTE - Not required to be met for containment purge supply and exhaust valve(s) in penetrations closed to comply with LCO 3.9.3.c.1. Verify each required containment purge supply and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	~~18~~ 24 months	1 5 1 5 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.3, CONTAINMENT PENETRATIONS

1. CNP has analyzed a boron dilution event in MODE 6. Therefore, ISTS 3.9.2 is not included in the ITS and ISTS 3.9.4 is renumbered as ITS 3.9.3.

2. The brackets are removed and the proper plant specific information/value is provided.

3. Typographical/grammatical error corrected.

4. The Note has been modified consistent with the current licensing basis.

5. Changes have been made to be consistent with changes made in another Specification and to be consistent with plant specific nomenclature.

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

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

B 3.9 REFUELING OPERATIONS 1

B 3.9.~~4~~ 3 Containment Penetrations

BASES

BACKGROUND

During movement of [recently] irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required. 2

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of [recently] irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. 2

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of [recently] irradiated fuel assemblies within containment, containment closure is required; therefore, the door 2

BASES

BACKGROUND (continued)

interlock mechanism may remain disabled, but at least one air lock door must always remain ~~[~~capable of being~~]~~ closed. 2 7

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Containment Purge Supply and Exhaust System includes ~~two subsystems. The normal subsystem includes~~ a ~~42~~ 24 inch purge supply penetration and a ~~42~~ 30 inch exhaust penetration. ~~The second subsystem, a minipurge system, includes an 8 inch purge penetration and an 8 inch exhaust penetration.~~ During MODES 1, 2, 3, and 4, the two valves in each of the ~~normal~~ purge supply and exhaust penetrations are ~~secured in the~~ normally maintained closed ~~position~~. ~~The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is~~ The Containment Purge Supply and Exhaust System is not subject to a Specification in MODE 5. 3

In MODE 6, large air exchangers are necessary to conduct refueling operations. The ~~normal 42 inch purge system~~ Containment Purge Supply and Exhaust System is used for this purpose~~, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."~~ .

~~[The minipurge system remains operational in MODE 6, and all four valves are also closed by the ESFAS.~~

~~[or]~~

~~The minipurge system is not used in MODE 6. All four 8 inch valves are secured in the closed position.]~~

The ~~other~~ containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during ~~recently~~ irradiated fuel movements ~~(Ref. 1)~~. 2 3

BASES

APPLICABLE SAFETY ANALYSES

~~During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident [involving handling recently irradiated fuel].~~ The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. ~~2~~ ①). Fuel handling accidents, analyzed in Reference ~~2~~, ~~include~~ involve dropping a single irradiated fuel assembly and handling tool ~~or a heavy object onto other irradiated fuel assemblies~~. The requirements of LCO 3.9.~~7~~ ⑥, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to ~~[~~irradiated fuel movement with containment closure capability ~~or a minimum decay time of [X] days without containment closure capability]~~, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are ~~well within~~ a small fraction of the guideline values specified in 10 CFR 100. ~~Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).~~ ③ ④ ③ ④ ① ② ③

Containment ~~p~~Penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). ③

LCO

~~REVIEWER'S NOTE~~

~~The allowance to have containment personnel airlock doors open and penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on (1) confirmatory dose calculations of a fuel handling accident as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitments from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences) that the open airlock can and will be promptly closed following containment evacuation and that the open penetration(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.~~ ②

This LCO limits the consequences of a fuel handling accident ~~involving handling recently irradiated fuel~~ in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge supply and exhaust penetrations ② ③

BASES

LCO (continued)

~~(~~and the containment personnel air locks~~)~~. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the UFSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit. ② ③ Supply Supply ③ ③

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere via the auxiliary building vent to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident. ① ⑤

The containment personnel air lock doors ~~may~~ be open during movement of irradiated fuel in the containment ~~and during CORE ALTERATIONS~~ provided that one door is capable of being closed in the event of a fuel handling accident INSERT 1. Should a fuel handling accident occur inside containment, one personnel air lock door will be closed following an evacuation of containment. ⑥ ⑤ ③

APPLICABILITY

The containment penetration requirements are applicable during movement of ~~[recently]~~ irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. ~~[Additionally, due to radioactive decay, a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days) will result in doses that are well within the guideline values specified in 10 CFR 100 even without containment closure capability.]~~ Therefore, under these conditions no requirements are placed on containment penetration status. ② ②

3 **INSERT 1**

A designated individual shall be available at all times during movement of irradiated fuel to close an air lock door if required. Cables or hoses transversing the air lock shall be designed to allow for removal in a timely manner (e.g., quick disconnects).

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BASES

APPLICABILITY (continued)

- REVIEWER'S NOTE -

~~The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by analysis that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10CFR100).~~

~~Additionally, licensees adding the term "recently" must make the following commitment which is consistent with draft NUMARC 93-01, Revision 3, Section 11.2.6 "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment - Primary (PWR)/Secondary (BWR)."~~

~~"The following guidelines are included in the assessment of systems removed from service during movement irradiated fuel:~~

~~- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification OPERABILITY amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.~~

~~- A single normal or continency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.~~

~~The purpose of the "prompt methods" mentioned above are to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."~~

②

 1

BASES

ACTIONS

A.1

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge Supply and Exhaust ~~Isolation~~ System not capable of automatic actuation when the purge supply and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of ~~[recently]~~ irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position. 3 3 2

SURVEILLANCE REQUIREMENTS

SR 3.9.~~4~~3.1 1

This Surveillance is in its demonstrates that each of the containment penetrations required status ~~to be in its closed position is in that position~~. ~~The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.~~ INSERT 2 5

The Surveillance is performed every 7 days during movement of ~~[recently]~~ irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. ~~A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance~~ ensures that a postulated fuel handling accident ~~involving handling recently irradiated fuel~~ that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess of ~~those recommended by Standard Review Plan Section 15.7.4 (Reference 3)~~. INSERT 3 2 7 2 3 5

SR 3.9.~~4~~3.2 1

This Surveillance demonstrates that each required containment purge supply and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The ~~18~~ 24 month Frequency maintains consistency with other similar ~~ESFAS instrumentation and~~ valve testing requirements. ~~In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY~~ INSERT 4 3 1 3

5 **INSERT 2**

The LCO 3.9.3.c.2 status requirement, which requires penetrations to be capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System, can be verified by ensuring each required

3 **INSERT 3**

a small fraction of the guideline values specified in 10 CFR 100

3 **INSERT 4**

LCO 3.3.6, "Containment Purge Supply and Exhaust System Isolation Instrumentation," provides additional Surveillance Requirements for the containment purge supply and exhaust valve actuation circuitry.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.~~ These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident ~~involving handling recently irradiated fuel~~ to limit a release of fission product radioactivity from the containment. ③ ②

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic ~~acutation~~ actuation capability. ⑥

REFERENCES

~~1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~ ③

~~2.~~ 1. uFSAR, Section ~~15.4.5~~ 14.2.1.5. ③ ②

~~3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.~~ ③

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.3 BASES, CONTAINMENT PENETRATIONS

1. Changes are made to reflect consistency with or those changes made to the Specification. Subsequent requirements are renumbered or revised, where applicable, to reflect the changes.

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

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

4. The reference to a Fuel Handling Accident being initiated by CORE ALTERATIONS or the dropping of a heavy object onto irradiated fuel assemblies is deleted from the Applicable Safety Analyses section of the Bases. CORE ALTERATIONS other than irradiated fuel movement inside containment and dropping of a heavy object onto irradiated fuel assemblies are not assumed to initiate a Fuel Handling Accident. Only the dropping of an irradiated fuel assembly is assumed to initiate a Fuel Handling Accident.

5. Changes have been made to be consistent with the ISTS.

6. Typographical/grammatical error corrected.

7. Editorial change for clarity.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.3, CONTAINMENT PENETRATIONS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 4

ITS 3.9.4, Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

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

ITS

A.1

REFUELING OPERATIONS

3/4 9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

LCO 3.9.4

3.9.8.1 At least one residual heat removal loop shall be OPERABLE and in operation. (M.1)

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of the reactor vessel flange. (A.2)

ACTION:

ACTION A

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.* Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. (A.3) (L.1) (L.2)

Add proposed Required Action A.3 (M.2)

LCO 3.9.4 Note

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. (M.3)

c. ~~The provisions of Specification 3.0.3 are not applicable.~~ (A.4)

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per ~~24~~ 12 hours. (M.4)

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.7.b.2. (L.1)

A.1

ITS

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

M.1

OPERABLE and

LCO 3.9.4

3.9.8.1 At least one residual heat removal loop shall be in operation.

A.2

APPLICABILITY: MODE 6. with the water level ≥ 23 ft above the top of the reactor vessel flange

ACTION:

A.3

ACTION A

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.* Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

L.1

L.2

Add proposed Required Action A.3

M.2

LCO 3.9.4 Note

b. ~~The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.~~

M.3

c. ~~The provisions of Specification 3.0.3 are not applicable.~~

A.4

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per ~~24~~ 12 hours.

M.4

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.7.b.2.

L.1

DISCUSSION OF CHANGES
ITS 3.9.4, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - HIGH WATER LEVEL

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 3.9.8.1 requires at least one residual heat removal loop to be in operation in MODE 6. ITS 3.9.4 requires one RHR loop to be OPERABLE and in operation in MODE 6 with the water level greater than or equal to 23 feet above the top of the reactor vessel flange. However, ITS 3.9.5 covers the Applicability of MODE 6 with water level less than 23 feet above the top of the reactor vessel flange. This changes the CTS by splitting the requirements associated with CTS 3.9.8.1 into two Applicabilities, one for MODE 6 with water level < 23 feet above the top of the reactor vessel flange, and one for MODE 6 with water level greater than of equal to 23 feet above the reactor vessel flange.

The purpose of CTS 3.9.8.1 is to ensure that adequate decay heat removal capability is in operation and that the coolant is circulated in MODE 6. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. MODE 6 RHR and coolant circulation requirements are governed by ITS 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and ITS 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level." The combination of ITS 3.9.4 and ITS 3.9.5 ensures that the appropriate RHR loops are available in MODE 6 regardless of the water level. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.9.8.1 Action a states, in part, that with less than one RHR loop in operation, suspend all operations involving an increase in the reactor decay heat load of the Reactor Coolant System. ITS 3.9.4 Required Action A.2 states, in part, that with the RHR loop requirements not met, suspend loading irradiated fuel assemblies in the core. This changes the CTS by requiring that the loading of irradiated fuel assemblies be suspended instead of requiring that all operations involving an increase in the reactor decay heat load be suspended.

This change is acceptable because the requirements have not changed. The reactor decay heat load is generated only by irradiated fuel. The only method of increasing the decay heat load of a reactor in MODE 6 is to load additional irradiated fuel assemblies into the core. Therefore, the CTS and ITS requirements are equivalent. This change is designated as administrative because it does not result in technical changes to the CTS.

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DISCUSSION OF CHANGES
ITS 3.9.4, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - HIGH WATER LEVEL

A.4 CTS 3.9.8.1 Action c states "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.4 does not include this statement. This changes CTS by deleting the Specification 3.0.3 exception.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.8.1 requires that at least one residual heat removal loop be in operation. ITS 3.9.4 requires that one RHR loop shall be OPERABLE and in operation. This changes the CTS by requiring the RHR loop to be OPERABLE, instead of just in operation.

The purpose of CTS 3.9.8.1 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. However, the CTS LCO could be interpreted as allowing an RHR loop to be placed in operation that was not OPERABLE. The ITS eliminates this possible misinterpretation. This change is acceptable because the RHR loop must be OPERABLE (i.e., capable of performing its safety function) instead of just being in operation. This change is designated as more restrictive because the ITS contains more specific requirements on a component.

M.2 The CTS 3.9.8.1 Actions do not include an action to immediately initiate action to satisfy the RHR loop requirements in the event the RHR loop requirements are not met. ITS 3.9.4 Required Action A.3 requires that action be immediately initiated to satisfy the RHR loop requirements. This changes the CTS by requiring that action be taken immediately to satisfy the RHR loop requirements.

The purpose of CTS 3.9.8.1 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. Although decay heat is removed from the Reactor Coolant System via natural circulation to the bulk of water contained in the refueling canal, this method of heat transfer can continue for only a discrete amount of time before boiling would occur. This change is acceptable because it requires that action be initiated to restore the RHR loop requirements in order to restore forced coolant flow and heat removal. This change is designated as more restrictive because additional actions will be required in the ITS than are required in the CTS.

M.3 CTS 3.9.8.1 Action b states that the RHR loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. The ITS LCO 3.9.4 Note states that the required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System, coolant with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, "Boron Concentration." This results in two changes to the CTS. First, the allowance to remove RHR from operation is no longer restricted to CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. Second, the use of the allowance in the ITS is predicated on prohibiting operations that would cause introduction into the RCS, coolant with a boron concentration less than that required to meet the boron concentration of LCO 3.9.1.

This change is acceptable because it applies appropriate controls during periods when RHR is not in operation. The ITS requirement prohibiting operations which would cause a reduction in the RCS boron concentration below that required to maintain the required shutdown margin is necessary to avoid unexpected reactivity changes. Under the ITS definition of CORE ALTERATIONS, many activities that would be considered CORE ALTERATIONS in the CTS, such as core mapping, are not considered CORE ALTERATIONS in the ITS. Therefore, the application of the allowance is expanded in the ITS to cover other activities beyond CORE ALTERATIONS. This change is nominally less restrictive, but represents no practical operational change, and the overall change is considered more restrictive. This change is designated as more restrictive because it imposes a new condition to be met when an RHR loop is not in operation.

M.4 CTS 4.9.8.1 requires that a residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per 24 hours. ITS SR 3.9.4.1 requires the same verification every 12 hours. This changes the CTS by requiring that RHR loop operation and reactor coolant flow rate be verified every 12 hours instead of every 24 hours.

The purpose of CTS 4.9.8.1 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. This change is acceptable since it results in an increased Frequency of performance. The 12 hour Frequency is consistent with similar CTS Surveillances in MODES 4 and 5, and with similar SRs in the ITS. This change is designated as more restrictive because the Surveillance will be performed at an increased Frequency in the ITS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.1 Action a states, in part, that with less than one RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System. This

DISCUSSION OF CHANGES
ITS 3.9.4, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - HIGH WATER LEVEL

CTS Action is modified by a footnote which states that addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2 (i.e., 2400 ppm). ITS 3.9.4 Required Action A.1 states that with the RHR loop requirements not met, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1, "Boron Concentration." ITS 3.9.1 requires boron concentration to be within limit. This changes the CTS by allowing coolant with boron concentration less than the RCS boron concentration, but greater than the boron concentration limit in ITS LCO 3.9.1, to be added to the RCS from sources other than the RWST when the RHR requirements are not met.

The purpose of CTS 3.9.8.1 Action a is to ensure that the required SHUTDOWN MARGIN is maintained during periods when the RHR requirements are not met. 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 OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions ensure that the RCS boron concentration is maintained within the limits of ITS LCO 3.9.1, which is sufficient to ensure that adequate SHUTDOWN MARGIN is maintained. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.1 Action a states, in part, that with less than one RHR loop in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. ITS 3.9.4 Required Actions A.4, A.5, and A.6 state that with the RHR loop requirements not met, within 4 hours close and secure the equipment hatch with at least four bolts, close one door in each air lock, and verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System. This changes the CTS Actions by allowing penetrations capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System to remain open when the RHR requirements are not met.

The purpose of CTS 3.9.8.1 Action a is to ensure that radioactive material does not escape the containment should the RHR requirements continue to not be met and boiling occurs in the core. Therefore, containment penetrations are closed to seal the containment. 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 OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions are consistent with the actions taken for containment closure in CTS 3.9.4 and ITS 3.9.3. Penetrations which can be closed by an OPERABLE Containment Purge Supply and Exhaust System do not need to be closed if RHR is inoperable, since the presence of radioactivity in the containment will cause the valves to close automatically, thus performing the isolation function. 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)

CTS

3.9 REFUELING OPERATIONS

3.9.5 4 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level 1

3.9.8.1 LCO 3.9.5 4 One RHR loop shall be OPERABLE and in operation. 1

removed from

\- NOTE -

Action b The required RHR loop may be ~~not in~~ operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System ~~coolant~~ (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1~~.~~, "Boron Concentration." TSTF-438 2 5

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations that would cause introduction of coolant into the RCS ~~coolant~~ with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	

Action a 5

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a	A.4 Close equipment hatch and secure with ~~four~~ bolts.	4 hours 3
	AND	
	A.5 Close one door in each air lock.	4 hours
	AND	
	A.6 ~~.1~~ Verify ~~Close~~ each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent~~.~~ , or	4 hours 4
	~~OR~~	
	~~A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.~~ Supply 5	~~4 hours~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.9.8.1	SR 3.9.5.1 4 Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ ~~2800~~ 2000 gpm. 3	12 hours 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.4, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - HIGH WATER LEVEL

1. CNP has analyzed a boron dilution event in MODE 6. Therefore, ISTS 3.9.2 is not included in the ITS and ISTS 3.9.5 is renumbered as ITS 3.9.4.

2. Editorial correction to be consistent with the format of the ITS.

3. The brackets are removed and the proper plant specific information/value is provided.

4. ISTS 3.9.5 Required Actions A.6.1 and A.6.2 are connected by an "OR" logical connector, such that either one can be performed to meet the requirements of the ACTION. However, the two Required Actions are applicable to all the penetrations; either Required Action A.6.1 or Required Action A.6.2 must be performed for all the penetrations. Thus, this will not allow one penetration to be isolated by use of a manual valve and another penetration to be capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System. This is not the intent of the requirement. The requirement is based on ISTS LCO 3.9.4 (ITS LCO 3.9.3), which requires each penetration to be either: a) closed by a manual or automatic isolation valve, blind flange, or equivalent; or b) capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System. For consistency with the actual LCO requirement, ISTS 3.9.5 Required Actions A.6.1 and A.6.2 have been combined into a single Required Action in ITS 3.9.4 Required Action A.6.

5. Changes have been made to be consistent with changes made in another Specification and to be consistent with plant specific nomenclature.

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

B 3.9 REFUELING OPERATIONS

B 3.9.~~5~~ 4 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level 1

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), ~~as required by GDC 34~~, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. INSERT 1 Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System. 2 4

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One ~~train~~ loop of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. 7 The LCO does permit ~~de-energizing~~ the RHR pump to be removed from operation for short durations, under the condition that the boron concentration is not diluted. This conditional ~~de-energizing~~ stopping of the RHR pump does not result in a challenge to the fission product barrier. TSTF-438

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat ~~or~~ ; and 3

4 **INSERT 1**

, as well as adjustments in Component Cooling Water System temperature and flow

BASES

LCO (continued)

b. Mixing of borated coolant to minimize the possibility of criticality, ~~and~~ . (7)

~~c. Indication of reactor coolant temperature.~~ (7)

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path ~~and to determine the low end temperature~~. The flow path starts in one of the RCS hot legs and is returned to at least one of the RCS cold legs. (7) (4)

The LCO is modified by a Note that allows the required operating RHR loop to ~~not~~ be removed from ~~in~~ operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1, "Boron Concentration." Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity. TSTF-438 (8) (8)

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS) ~~and Section 3.5, Emergency Core Cooling Systems (ECCS)~~. RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level." (1) (5) (1) (6) (6) (5)

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the

BASES

ACTIONS (continued)

minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and A.6.2 (1)

If no RHR is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with ~~four~~ bolts, ~~,~~ and (3) (6) (3)

b. One door in each air lock must be closed, ~~and~~ and (3) (3)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge Supply and Exhaust ~~Isolation~~ System. (1)

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment

BASES

ACTIONS (continued)

penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 4 1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. UFSAR, Section 5.5.7 9.3.2 6

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JUSTIFICATION FOR DEVIATIONS
ITS 3.9.4 BASES, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - HIGH WATER LEVEL

1. Changes are made to reflect those changes made to the ISTS. Subsequent requirements are renumbered or revised, where applicable, to reflect the changes.
2. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A, criteria have been replaced with references to the appropriate section of the UFSAR.
3. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.
4. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
5. The wording has been modified, as Section 3.5 does not provide requirements for the RHR Shutdown Cooling function.
6. The brackets have been removed and the proper plant specific information/value has been provided.
7. Changes have been made to be consistent with the ISTS.
8. Typographical/grammatical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.4, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION – HIGH WATER LEVEL

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 5

ITS 3.9.5, Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

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

ITS 3.9.5

A.1

ITS

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

LCO 3.9.5

3.9.8.1 At least one residual heat removal loop shall be in operation.

Add proposed LCO Note 1 — L.3

APPLICABILITY: MODE 6. with the water level < 23 ft above the top of the reactor vessel flange — A.2

ACTION: — A.3

ACTION B

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.* Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. — L.1, L.2

Add proposed Required Action B.2 — M.1

See ITS 3.9.4

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. ~~The provisions of Specification 3.0.3 are not applicable.~~ — A.4

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per 24 hours.

12 — M.2

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.7.b.2. — L.1

D. C. COOK - UNIT 1 3/4 9-9 AMENDMENT NO. 120

Page 1 of 4

ITS

A.1

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.5

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.* — A.5

Add proposed LCO Note 2 — L.3

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

ACTION A

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

Add proposed Required Action A.2 — A.6

b. The provisions of Specification 3.0.3 are not applicable. — A.4

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5. — L.4

Add proposed SR 3.9.5.2 and Note — M.3

*The normal or emergency power source may be inoperable for each RHR loop. — A.5

D. C. COOK - UNIT 1 3/4 9-9a Amendment No.78

ITS 3.9.5

A.1

ITS

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

L.3

LCO 3.9.5 — 3.9.8.1 At least one residual heat removal loop shall be in operation.

Add proposed LCO Note 1

A.2

APPLICABILITY: MODE 6.

with the water level < 23 ft above the top of the reactor vessel flange

ACTION:

A.3

ACTION B — a. With less than one residual heat removal loop in operation, excep as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.* Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

L.1

L.2

Add proposed Required Action B.2

M.1

See ITS 3.9.4 — b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. ~~The provisions of Specification 3.0.3 are not applicable~~

A.4

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 — 4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per ~~24~~ hours.

12

M.2

L.1

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.7.b.2.

ITS 3.9.5

A.1

ITS

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

A.5

LCO 3.9.5 3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

Add proposed LCO Note 2 — L.3

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

ACTION A

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

Add proposed Required Action A.2 — A.6

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ — A.4

SURVEILLANCE REQUIREMENTS

~~4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.~~ — L.4

Add proposed SR 3.9.5.2 and Note — M.3

~~*The normal or emergency power source may be inoperable for each RHR loop.~~ — A.5

DISCUSSION OF CHANGES
ITS 3.9.5, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - LOW WATER LEVEL

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 3.9.8.1 requires at least one residual heat removal loop to be in operation in MODE 6. ITS 3.9.5 requires two RHR loops to be OPERABLE and one RHR loop to be in operation in MODE 6 with the water level less than 23 feet above the top of the reactor vessel flange. However, ITS 3.9.4 covers the Applicability of MODE 6 with water level greater than or equal to 23 feet above the top of the reactor vessel flange. This changes the CTS by splitting the requirements associated with CTS 3.9.8.1 into two Applicabilities, one for MODE 6 with water level < 23 feet above the top of the reactor vessel flange, and one for MODE 6 with water level $\geq$ 23 feet above the reactor vessel flange.

The purpose of CTS 3.9.8.1 is to ensure that adequate decay heat removal capability is in operation and that the coolant is circulated in MODE 6. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. MODE 6 RHR and coolant circulation requirements are governed by ITS 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and ITS 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level." The combination of ITS 3.9.4 and ITS 3.9.5 ensures that the appropriate RHR loops are available in MODE 6 regardless of the water level. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.9.8.1 Action a states, in part, that with less than one RHR loop in operation, suspend all operations involving an increase in the reactor decay heat load of the Reactor Coolant System. ITS 3.9.5 does not include this requirement. This changes the CTS by eliminating the requirement to suspend operations involving an increase in reactor decay heat load.

This change is acceptable because the requirements have not changed. The reactor decay heat load is generated only by irradiated fuel. The only method of increasing the decay head load of a reactor in MODE 6 is to load additional irradiated fuel assemblies into the core. However, ITS LCO 3.9.6 prohibits loading of fuel assemblies into the reactor when the water level is less than 23 feet over the top of the reactor vessel flange. Therefore, when LCO 3.9.5 is applicable there is no method available to increase the reactor decay heat load, and the requirement can be deleted with no effect on plant operations. This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 3.9.5, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - LOW WATER LEVEL

A.4 CTS 3.9.8.1 Action c and CTS 3.9.8.2 Action c state, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.5 does not include this statement. This changes CTS by deleting the Specification 3.0.3 exception.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

A.5 CTS LCO 3.9.8.2 is modified by footnote *, which states that the normal or emergency power source may be inoperable for each RHR loop. ITS 3.9.5 does not include this statement. This changes the CTS by deleting an allowance already provided in a different portion of the ITS.

This change is acceptable because the ITS definition of OPERABLE contains the necessary requirements for a component to perform its safety function. The ITS definition of OPERABLE states that a component is OPERABLE if either the normal or emergency power source is OPERABLE. This change is designated as administrative because it does not result in technical changes to the CTS.

A.6 CTS 3.9.8.2 Action a states that with less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible. ITS 3.9.5 ACTION A includes the same requirement, but also includes an allowance (Required Action A.2) to immediately initiate action to establish $\geq$ 23 feet of water above the top of reactor vessel flange. This changes the CTS by providing the option to exit the Applicability of the LCO.

This change is acceptable because the requirements have not changed. Exiting the Applicability of LCO is always an option to exit an ACTION. Therefore, stating this option explicitly does not change the requirements of the Specification. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 The CTS 3.9.8.1 Actions do not include an action to immediately initiate action to restore one RHR loop to operation in the event the RHR loop requirements are not met. ITS 3.9.5 Required Action B.2 requires that action be immediately initiated to restore one RHR loop to operation. This changes the CTS by requiring that action be taken immediately to restore one RHR loop to operation.

The purpose of CTS 3.9.8.1 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. Although decay heat is removed from the Reactor Coolant System via natural circulation to the bulk of water contained in the refueling canal, this method of heat transfer can continue for only a discrete amount of time before boiling would occur. This change is acceptable because it requires that action be initiated to restore one RHR loop to operation in order to restore forced coolant flow and heat removal. This change is designated as more restrictive because additional actions will be required in the ITS than are required in the CTS.

M.2 CTS 4.9.8.1 requires that a residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per 24 hours. ITS SR 3.9.5.1 requires the same verification every 12 hours. This changes the CTS by requiring that RHR loop operation and reactor coolant flow rate be verified every 12 hours instead of every 24 hours.

The purpose of CTS 4.9.8.1 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. This change is acceptable since it results in an increased Frequency of performance. The 12 hour Frequency is consistent with similar CTS Surveillances in MODES 4 and 5, and with similar SRs in the ITS. This change is designated as more restrictive because the Surveillance will be performed at an increased Frequency in the ITS.

M.3 CTS 3.9.8.2 requires two independent RHR loops to be OPERABLE and CTS 3.9.8.1 requires at least one RHR loop to be in operation. ITS SR 3.9.5.2 requires verification every seven days of correct breaker alignment and that indicated power is available to the required RHR pump not in operation. A Note states that the Surveillance Requirement is not required to be performed until 24 hours after a required RHR pump is not in operation. This changes the CTS by adding a Surveillance Requirement.

The purpose of ITS 3.9.5 is to require one RHR loop to be in operation and one RHR loop to be held in readiness should it be needed. This change is acceptable because it verifies that the RHR loop that is in standby will be ready should it be needed. This change is designated as more restrictive because it adds a new Surveillance Requirement to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.1 Action a states, in part, that with less than one RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System. This CTS Action is modified by a footnote which states that addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2 (i.e., 2400 ppm). ITS 3.9.5 Required Action B.1 states that with no RHR loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1, "Boron Concentration." ITS 3.9.1 requires boron concentration to be within limit. This changes the CTS by allowing coolant with boron concentration less than the RCS boron concentration, but greater than the boron concentration limit in ITS LCO 3.9.1, to be added to the RCS from sources other than the RWST when the RHR loops are not in operation.

The purpose of CTS 3.9.8.1 Action a is to ensure that the required SHUTDOWN MARGIN is maintained during periods when the RHR requirements are not met. 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 OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions ensure that the RCS boron concentration is maintained within the limits of ITS LCO 3.9.1, which is sufficient to ensure that adequate SHUTDOWN MARGIN is maintained. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.1 Action a states, in part, that with less than one RHR loop in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. ITS 3.9.5 Required Actions B.3, B.4, and B.5 state that with no RHR loop in operation, within 4 hours close and secure the equipment hatch with at least four bolts, close one door in each air lock, and verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System. This changes the CTS Actions by allowing penetrations capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System to remain open when no RHR loop is in operation.

The purpose of CTS 3.9.8.1 Action a is to ensure that radioactive material does not escape the containment should the RHR requirements continue to not be met and boiling occurs in the core. Therefore, containment penetrations are closed to seal the containment. 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 OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA

DISCUSSION OF CHANGES
ITS 3.9.5, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - LOW WATER LEVEL

occurring during the repair period. The Required Actions are consistent with the actions taken for containment closure in CTS 3.9.4 and ITS 3.9.3. Penetrations which can be closed by an OPERABLE Containment Purge Supply and Exhaust System do not need to be closed if RHR is inoperable, since the presence of radioactivity in the containment will cause the valves to close automatically, thus performing the isolation function. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 1 – Relaxation of LCO Requirements)* ITS 3.9.5 is modified by two LCO Notes. Note 1 allows all RHR pumps to be removed from operation for ≤ 15 minutes when switching from one loop to another, provided several conditions are met. Note 2 allows one required RHR loop to be inoperable for up to 2 hours for Surveillance testing, provided that the other loop is OPERABLE and in operation. Neither CTS 3.9.8.1 nor CTS 3.9.8.2 contain these allowances. This changes the CTS by allowing the LCO to not be met under certain situations.

The purpose of CTS 3.9.8.1 and CTS 3.9.8.2 is to ensure sufficient decay heat removal is available in the specified MODES and conditions. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The ITS Notes allow normal operational evolutions, such as pump swapping and surveillance testing, to be performed while in the Applicability of the Specification. These evolutions are necessary to demonstrate RHR OPERABILITY. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L.4 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.9.8.2 requires verification that each RHR loop is OPERABLE per Specification 4.0.5. ITS 3.9.5 does not contain this Surveillance. This changes the CTS by deleting this specific Surveillance.

The purpose of CTS Specification 4.0.5 is to require inservice testing in accordance with 10 CFR 50.55a. The purpose of inservice testing of RHR is to detect gross degradation caused by impeller structural damage or other hydraulic component problems. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed function. This Technical Specification will no longer tie RHR loop OPERABILITY to the Inservice Testing Program. This change is acceptable because it is not necessary to perform inservice testing of an RHR loop to determine if it is OPERABLE, as the system is routinely operated and the RHR loops are instrumented so that degradation can be observed. Significant degradation of the RHR System would be indicated by the RHR System flow and temperature instrumentation in the Control Room. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

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

CTS

3.9 REFUELING OPERATIONS

3.9.6 5 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level ①

3.9.8.1, 3.9.8.2 LCO 3.9.6 5 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation. ①

- NOTES - TSTF-438

Doc L.3

1. All RHR pumps may be ~~de-energized~~ removed from operation for ≤ 15 minutes when switching from one ~~train~~ loop to another provided: ⑤

   a. The core outlet temperature is maintained at least ≥ 10 ~~degrees~~ °F below saturation temperature; ⑨ ⑥ ②

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System (RCS) boron concentration; and ⑥ ②

   c. No draining operations to further reduce RCS water volume are permitted.

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.8.2 Action a — A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	OR	
	A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately

 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
3.9.8.1 Action a — B. No RHR loop in operation.	B.1	Suspend operations that would cause introduction of coolant into the RCS ~~coolant~~ with boron concentration less than required to meet the boron concentration of LCO 3.9.~~1~~ 1, "Boron Concentration."	Immediately (8) (6)
	AND		
	B.2	Initiate action to restore one RHR loop to operation.	Immediately
	AND		
	B.3	Close equipment hatch and secure with ~~four~~ bolts.	4 hours (3)
	AND		
	B.4	Close one door in each air lock.	4 hours
	AND		
	B.5 ~~1~~	~~Close~~ Verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent ~~.~~, or	4 hours (7) (7)
	~~OR~~	Move from next page	(7)

3.9.8.1
Action a

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Move to B.5	B.5.2 ~~Verify each penetration is capable of being closed~~ by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours Supply (8) (7)

SURVEILLANCE REQUIREMENTS

4.9.8.1

Doc M.3

SURVEILLANCE		FREQUENCY
SR 3.9.6.1 (5)	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ ~~2800~~ 2000 gpm.	12 hours (1) (3)
SR 3.9.6.2 (5)	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days (1)

INSERT 1

(4)

4 **INSERT 1**

-NOTE-

Not required to be performed until 24 hours after a required RHR pump is not in operation.

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.5, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - LOW WATER LEVEL

1. CNP has analyzed a boron dilution event in MODE 6. Therefore, ISTS 3.9.2 is not included in the ITS and ISTS 3.9.6 is renumbered as ITS 3.9.5.

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

3. The brackets are removed and the proper plant specific information/value is provided.

4. TSTF-265 was previously approved and incorporated in NUREG-1431, Rev. 2, in similar SRs (e.g., ISTS SRs 3.4.5.3, 3.4.6.3, 3.4.7.3, and 3.4.8.2). Consistent with TSTF-265, a Note is added to ISTS SR 3.9.6.2 that permits the performance of the SR to verify correct breaker alignment and power availability to be delayed until 24 hours after a required pump is not in operation. This provision is required because when pumps are swapped under the current requirements, the Surveillance is immediately not met on the pump taken out of operation. This change avoids entering an Action for a routine operational occurrence. The change is acceptable because adequate assurance exists that the pump is aligned to the correct breaker with power available because, prior to being removed from operation, the applicable pump had been in operation. Allowing 24 hours to perform the breaker alignment verification is acceptable because the pump was in operation, which demonstrated OPERABILITY, and because 24 hours is currently allowed by invoking SR 3.0.3. This is a new Surveillance Requirement not required in CTS 3.9.8.2.

5. Editorial change made to be consistent with the LCO statement.

6. Editorial change made to be consistent with the format of the ITS.

7. ISTS 3.9.6 Required Actions B.5.1 and B.5.2 are connected by an "<u>OR</u>" logical connector, such that either one can be performed to meet the requirements of the ACTION. However, the two Required Actions are applicable to all the penetrations; either Required Action B.5.1 or Required Action B.5.2 must be performed for all the penetrations. Thus, this will not allow one penetration to be isolated by use of a manual valve and another penetration to be capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System. This is not the intent of the requirement. The requirement is based on ISTS LCO 3.9.4 (ITS LCO 3.9.3), which requires each penetration to be either: a) closed by a manual or automatic isolation valve, blind flange, or equivalent; or b) capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System. For consistency with the actual LCO requirement, ISTS 3.9.6 Required Actions B.5.1 and B.5.2 have been combined into a single Required Action in ITS 3.9.5 Required Action B.5.

8. Changes have been made to be consistent with changes made in another Specification and be consistent with plant specific nomenclature.

9. The limit has been changed to be consistent with the same limit provided in Notes to ISTS 3.4.6 and ISTS 3.4.7.

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

B 3.9 REFUELING OPERATIONS 1

B 3.9.6 5 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

BASES

BACKGROUND	The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System. 2 5 INSERT 1
APPLICABLE SAFETY ANALYSES	If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains loops of the RHR System are required to be OPERABLE, and one train loop in operation, in order to prevent this challenge. 9 The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide: a. Removal of decay heat; . 7 b. Mixing of borated coolant to minimize the possibility of criticality; and 9 c. Indication of reactor coolant temperature. 9 This LCO is modified by a Note that permits the RHR pumps to be de-energized removed from operation for ≤ 15 minutes when switching from one train loop to another. TSTF-438

5 **INSERT 1**

, as well as adjustments in Component Cooling Water System temperature and flow

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BASES

LCO (continued)

(1) at least

The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10 degrees F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped. (4) (7)

(NG)

This LCO is modified by a Note that allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible. (3)

time to — potential for RCS — down — and RCS makeup

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. (9) (5)

at least one of

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level." (4) (8) (6) (8) (1)

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An Immediate Completion Time is necessary for an operator to initiate corrective actions. (4) (1)

BASES

ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3, B.4, B.5.1, and B.5.2 (1)

If no RHR is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with four bolts, (3) (4)

b. One door in each air lock must be closed, and (3) (7)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. Supply (7) (1)

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

BASES

SURVEILLANCE REQUIREMENTS 1

SR 3.9.6.1 5

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room. 8

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience. 9

REFERENCES

1. UFSAR, Section ~~5.5.7~~ 9.3.2 5 4

INSERT 2 1

1 **INSERT 2**

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.5 BASES, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - LOW WATER LEVEL

1. Changes are made to reflect those changes made to the ISTS. Subsequent requirements are renumbered or revised, where applicable, to reflect the changes.

2. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A, criteria have been replaced with references to the appropriate section of the UFSAR.

3. The current wording implies specific restrictions not contained in LCO Note 2. Therefore, the words have been modified to provide guidance on what should be considered in determining whether or not to use the Note allowance.

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

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

6. The wording has been modified, as Section 3.5 does not provide requirements for the RHR Shutdown Cooling function.

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

8. Typographical/grammatical error corrected.

9. Changes are made to be consistent with the ISTS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.5, RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION – LOW WATER LEVEL

There are no specific NSHC discussions for this Specification.

ATTACHMENT 6

ITS 3.9.6, Refueling Cavity Water Level

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

ITS 3.9.6

ITS

A.1

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.6

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange. L.1

APPLICABILITY: During movement of irradiated fuel assemblies ~~or control rods~~ ~~within the reactor pressure vessel~~ ~~while in MODE 6~~. L.2 A.2 M.1

ACTION:

ACTION A

With the requirements of the above specification not satisfied, suspend all operations involving movement of irradiated fuel assemblies ~~or control rods~~ ~~within the pressure vessel~~. ~~The provisions of Specification 3.0.3 are not applicable.~~ L.1 L.2 A.3 M.1

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

4.9.10 The water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of and~~ ~~at least once per~~ 24 hours ~~thereafter during movement of fuel assemblies~~ ~~or control rods~~. L.3 L.2

ITS 3.9.6

ITS

A.1

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.6

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange. L.1

APPLICABILITY: During movement of irradiated fuel assemblies ~~or control rods~~ ~~within the reactor pressure vessel~~ ~~while in MODE 6.~~ L.2 A.2 M.1

ACTION:

ACTION A

With the requirements of the above specification not satisfied, suspend all operations involving movement of irradiated fuel assemblies ~~or control rods~~ ~~within the pressure vessel.~~ ~~The provisions of Specification 3.0.3 are not applicable.~~ L.1 L.2 A.3 M.1

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

4.9.10 The water level shall be determined to be at ~~least its minimum~~ required depth ~~within 2 hours prior to the start of and~~ at least once per 24 hours ~~thereafter during movement of fuel assemblies~~ ~~or control rods.~~ L.3 L.2

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DISCUSSION OF CHANGES
ITS 3.9.6, REFUELING CAVITY WATER LEVEL

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 3.9.10 is applicable in MODE 6 during movement of fuel assemblies or control rods within the reactor pressure vessel. ITS 3.9.6 is applicable during movement of irradiated fuel assemblies within containment. This changes the CTS by eliminating the "MODE 6" portion of the Applicability. The change to "irradiated fuel assemblies" from "fuel assemblies" is discussed in DOC L.1. The change from within "the reactor pressure vessel" to within "containment" is discussed in DOC M.1. The change eliminating control rods is discussed in DOC L.2.

This change is acceptable because the technical requirements have not changed. Fuel movement in the containment only occurs in MODE 6. Therefore, specifying MODE 6 during movement of fuel is unnecessary. This change is designated as administrative because the technical requirements of the CTS have not changed.

A.3 The CTS 3.9.10 Action states "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.6 does not include this statement. This changes the CTS by deleting the Specification 3.0.3 exception.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.10 is applicable during movement of fuel assemblies or control rods within the "reactor pressure vessel" while in MODE 6. The CTS 3.9.10 Action states that with the reactor vessel water level not within limit, suspend movement of fuel assemblies or control rods within the "pressure vessel." The ITS 3.9.6 Applicability is during movement of irradiated fuel assemblies within "containment." ITS 3.9.6 ACTION A states that with the refueling cavity water level not within limit, suspend movement of irradiated fuel assemblies within "containment." This changes the CTS by expanding the suspension of movement of fuel assemblies from within the "reactor pressure vessel" to within the "containment." The change to "irradiated fuel assemblies" from "fuel assemblies" is discussed in DOC L.1. The change eliminating MODE 6 is discussed in DOC A.2. The change eliminating control rods is discussed in DOC L.2.

DISCUSSION OF CHANGES
ITS 3.9.6, REFUELING CAVITY WATER LEVEL

The purpose of CTS 3.9.10 is to ensure the refueling cavity water level is greater than or equal to that assumed in the fuel handling accident analysis. This change is acceptable because the fuel handling accident analysis assumes an irradiated fuel assembly is damaged within the containment, not only within the reactor vessel. In order to protect the initial assumptions of the fuel handling accident analysis, prohibition of irradiated fuel movement within the containment is required. This change is designated as more restrictive because it will prohibit operations that are not prohibited in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.9.10 states that at least 23 feet of water must be maintained over the reactor pressure vessel flange in MODE 6 during movement of fuel assemblies or control rods within the reactor pressure vessel. The CTS 3.9.10 Action requires suspension of movement of fuel assemblies or control rods within the pressure vessel if the water level requirement is not met. ITS 3.9.6 states the refueling cavity water level shall be maintained ≥ 23 feet above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. ITS 3.9.6 Required Action A.1 requires the suspension of movement of irradiated fuel assemblies within containment. This changes the CTS restricting the Applicability and ACTIONS from movement of any "fuel assemblies" within the reactor pressure vessel to movement of "irradiated fuel assemblies" within containment. The change eliminating MODE 6 is discussed in DOC A.2. The change from within "the reactor pressure vessel" to within "containment" is discussed in DOC M.1. The change eliminating control rods is discussed in DOC L.2.

The purpose of CTS 3.9.10 is to ensure that the refueling cavity water level is greater than or equal to that assumed in the fuel handling accident analysis. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The fuel handling accident analysis is based on damaging a single irradiated fuel assembly. An unirradiated fuel assembly does not contain the radioactive materials generated by fission and does not result in significant offsite doses if damaged. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.2 *(Category 2 – Relaxation of Applicability)* CTS 3.9.10 requires the refueling cavity water level to be maintained at least 23 feet over the top of the reactor pressure vessel flange during movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6. The CTS 3.9.10 Action requires suspension of all operations involving movement of the fuel assemblies or control rods within the pressure vessel in the event the LCO is not met. CTS 4.9.10 requires a determination of the refueling canal water level during the movement of fuel assemblies or control rods. ITS 3.9.6 requires the refueling cavity water level to be maintained greater than or equal to 23 feet above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. This changes the CTS by deleting the requirement that the LCO, ACTIONS, and Surveillance is applicable during control rod movement. The change to "irradiated fuel assemblies" from "fuel assemblies" is discussed in DOC L.1. The change eliminating MODE 6 is discussed in DOC A.2. The change from within "the reactor pressure vessel" to within "containment" is discussed in DOC M.1.

The purpose of CTS 3.9.10 is to ensure that the refueling cavity water level is greater than or equal to that assumed in the fuel handling accident analysis. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The fuel handling accident is based on damaging a single irradiated fuel assembly. Movement of control rods is not assumed to result in a fuel handling accident. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.3 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.9.10 requires the refueling cavity water level to be determined to be within limit "within 2 hours prior to the start of" and at least once per 24 hours thereafter during movement of fuel assemblies or control rods. ITS SR 3.9.6.1 requires verification that the refueling cavity water level is within limit every 24 hours. This changes the CTS by reducing the Frequency for verifying refueling cavity water level from 2 hours before entering the Applicability of the LCO to 24 hours before entering the Applicability of the LCO.

The purpose of CTS 4.9.10 is to ensure that the refueling cavity water level is greater than or equal to that assumed in the fuel handling accident analysis. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The Frequency of 24 hours is sufficient during the movement of fuel assemblies, therefore it is sufficient before fuel assemblies are moved. ITS SR 3.0.1 requires the SR to be met during the MODES or other specified conditions in the Applicability. Therefore, the water level must be met when fuel assemblies are moved or fuel assembly movement must be suspended immediately (thereby exiting the Applicability of the Specification). Therefore, changing the Frequency from 2 hours before moving fuel assemblies to within 24 hours before moving fuel assemblies has no effect on plant safety. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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

CTS

Refueling Cavity Water Level
3.9.7 6 (1)

3.9 REFUELING OPERATIONS (1)

3.9.7 6 Refueling Cavity Water Level

3.9.10 LCO 3.9.7 6 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange. (1)

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

Action

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7 6.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

4.9.10 (1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.6, REFUELING CAVITY WATER LEVEL

1. CNP has analyzed a boron dilution event in MODE 6. Therefore, ISTS 3.9.2 is not included in the ITS and ISTS 3.9.7 is renumbered as ITS 3.9.6.

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

 (1)

(1)

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level (6)

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~≤ 25%~~ a small fraction of 10 CFR 100 limits~~, as provided by the guidance of Reference 3~~. (2)
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of ~~[ ]~~ 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref~~s~~. 4 ~~and 5~~) (3). (3) (2) Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits~~, as provided by the guidance of Reference 3~~. (2)

BASES 6 1

APPLICABILITY

LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not ~~present~~ being moved in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.~~15~~ 14 "Fuel Storage Pool Water Level." 5 1

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving ~~or~~ movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. 4

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 6 1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. U FSAR, Section ~~15.4.5~~ 14.2.1.5 3
3. ~~NUREG-0800, Section 15.7.4~~ 2
3 ~~4~~. 10 CFR 100.10. 2

 (1)

BASES

REFERENCES (continued)

5. Malinowski, D. D., ~~Bell, M. J.~~, Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December 1971. (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.6 BASES, REFUELING CAVITY WATER LEVEL

1. Changes are made to reflect those changes made to the ISTS. Subsequent requirements are renumbered or revised, where applicable, to reflect the changes.

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

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

4. Typographical/grammatical error corrected.

5. Changes are made to be consistent with the ISTS.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.6, REFUELING CAVITY WATER LEVEL

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 7

Relocated/Deleted Current Technical Specifications (CTS)

CTS 3/4.9.3, Decay Time

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

LA.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least

a. 100 hours

b. 148 hours

APPLICABILITY: Specification 3.9.3.a - From September 15 through June 15, during movement of irradiated fuel in the reactor pressure vessel

Specification 3.9.3.b - From June 16 through September 14, during movement of irradiated fuel in the reactor pressure vessel

ACTION:

With the reactor subcritical for less than the required time, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical as required by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CTS 3/4.9.3

LA.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least:

a. 100 hours

b. 148 hours

APPLICABILITY: Specification 3.9.3.a - From September 15 through June 15, during movement of irradiated fuel in the reactor pressure vessel.

Specification 3.9.3.b - From June 16 through September 14, during movement of irradiated fuel in the reactor pressure vessel

ACTION:

With the reactor subcritical for less than the required time, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical as required by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

DISCUSSION OF CHANGES
CTS 3/4.9.3, DECAY TIME

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Category 6 – Relocation of LCO or SR to the TRM)* CTS LCO 3.9.3 requires the reactor to be subcritical for a required period of time (100 hours from September 15 through June 15 and 148 hours from June 16 through September 14) prior to movement of irradiated fuel in the reactor pressure vessel. ITS 3.9 does not include the requirements for decay time. This changes the CTS by moving the explicit decay time requirements from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of CTS LCO 3.9.3 to ensure that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products in the irradiated fuel consistent with the assumptions used in the fuel handling accident analysis. Additionally, two time limits are currently provided to account for decay heat load capacity of the spent fuel storage pool. Although CTS LCO 3.9.3 satisfies Criterion 2 of the Technical Specifications Selection Criteria in 10 CFR 50.36 (c)(2)(ii) (for the radioactive decay assumptions in the fuel handling accident), the requirements for decay time following subcriticality will always be met for a refueling outage because of the operations required prior to moving irradiated fuel in the reactor vessel (e.g., containment entry, removal of vessel head, removal of vessel internals, etc.). Also, this change is acceptable because the removed information will be adequately controlled in 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 change because a requirement is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.9.3, DECAY TIME

There are no specific NSHC discussions for this Specification.

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CTS 3/4.9.5, Communications

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

CTS 3/4.9.5

R.1

REFUELING OPERATIONS

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

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CTS 3/4.9.5

R.1

REFUELING OPERATIONS

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: *During CORE ALTERATIONS.*

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

DISCUSSION OF CHANGES
CTS 3/4.9.5, COMMUNICATIONS

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3.9.5 states that direct communications shall be maintained between the control room and personnel at the refueling station during CORE ALTERATIONS. This ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS. The prompt notification of the control room of a fuel handling accident is not an assumption in the fuel handling accident analysis. While notification is necessary to ensure that the control room is isolated to meet the control room operator dose limits in General Design Criteria 19, the fuel handling accident analysis does not take credit for direct communications between the refueling station and the control room (30 minutes is assumed before control room operator actions are taken). This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual (TRM).

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. Communications are not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The Communications Specification does not satisfy criterion 1.

2. Communications are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The Communications Specification does not satisfy criterion 2.

3. Communications are part of the primary success path and are assumed in the mitigation of a DBA which assumes the failure of a fission product barrier. However, communications are not a structure, system or component. The Communications Specification does not satisfy criterion 3.

4. Communications are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. As discussed in Section 4.0, (Appendix A, page A-67) and Table 1 of WCAP-11618, communications was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation,

DISCUSSION OF CHANGES
CTS 3/4.9.5, COMMUNICATIONS

considers it applicable to CNP Units 1 and 2, and concurs with this assessment. The Communications Specification does not meet criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the communications LCO and associated Surveillances may be relocated out of the Technical Specifications. The communications specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as a relocation because the LCO did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.9.5, COMMUNICATIONS

There are no specific NSHC discussions for this Specification.

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CTS 3/4.9.13, Spent Fuel Cask Movement

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

CTS 3/4.9.13

LA.1

REFUELING OPERATIONS

SPENT FUEL CASK MOVEMENT

LIMITING CONDITION FOR OPERATION

3.9.13 Movement of the spent fuel cask above elevation 620 feet shall be done with the spent fuel cask handling crane operating in the Controlled Path Mode of operation.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks which prevent raising the bottom of the spent fuel cask more than 6 inches above the top of the Cask Drop Protection System cyclinder and restrict the crane's movement to the Controlled Path shall be demonstrated OPERABLE within 7 days prior to crane operation in the Controlled Path Mode and at least once per 7 days thereafter during crane operation in the Controlled Path Mode.

CTS 3/4.9.13

LA.1

REFUELING OPERATIONS

SPENT FUEL CASK MOVEMENT

LIMITING CONDITION FOR OPERATION

3.9.13 Movement of the spent fuel cask above elevation 620 feet shall be done with the spent fuel cask handling crane operating in the Controlled Path Mode of operation.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks which prevent raising the bottom of the spent fuel cask more than 6 inches above the top of the Cask Drop Protection System cylinder and restrict the crane's movement to the Controlled Path shall be demonstrated OPERABLE within 7 days prior to crane operation in the Controlled Path Mode and at least once per 7 days thereafter during crane operation in the Controlled Path Mode.

D. C. COOK - UNIT 2 3/4 9-16

Page 2 of 2

DISCUSSION OF CHANGES
CTS 3/4.9.13, SPENT FUEL CASK MOVEMENT

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 6 – Relocation of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPP, or IIP)* CTS LCO 3.9.13 requires the movement of the spent fuel cask above elevation 620 feet to be done with the spent fuel cask handling crane operating in the Controlled Path Mode of operation. The ITS does not include the requirements for the movement of the spent fuel cask above elevation 620 feet. This changes the CTS by moving the explicit requirements for movement of the spent fuel cask above elevation 620 feet, including the Action and Surveillance Requirement, from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of CTS LCO 3.9.13 to ensure that, during insertion or removal of spent fuel casks from the spent fuel pool; fuel cask movement will be constrained to the path and lift height assumed in the Cask Drop Protection System safety analysis. Restricting the spent fuel cask movement within these requirements provides protection for the spent fuel pool and stored fuel from the effects of a fuel cask drop accident. These requirements are proposed to be relocated to the TRM since the movement of loads other than fuel assemblies is controlled based on the heavy loads analysis. The bounding design basis fuel handling accident in the auxiliary building assumes a single irradiated fuel assembly is damaged. In addition, as stated in the NRC Safety Evaluation for License Amendments 197 (Unit 1) and 182 (Unit 2), dated July 12, 1995, the controls in place ensure that the potential for other, more severe events that could occur, such as a heavy load drop on irradiated fuel, need not be postulated and analyzed. This change is acceptable because the removed information will be adequately controlled in 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 change because a requirement is being removed from the Technical Specifications.

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DISCUSSION OF CHANGES
CTS 3/4.9.13, SPENT FUEL CASK MOVEMENT

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.9.13, SPENT FUEL CASK MOVEMENT

There are no specific NSHC discussions for this Specification.

CTS 3/4.9.14, Spent Fuel Cask Drop Protection System

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

CTS 3/4.9.14

LA.1

REFUELING OPERATIONS

SPENT FUEL CASK DROP PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.14 The maximum weight of a spent fuel cask used with the Cask Drop Protection System shall be limited to 110 tons (nominal).

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14 The weight of a spent fuel cask shall be verified to be ≤ 110 tons (nominal) prior to its use with the Cask Drop Protection System.

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CTS 3/4.9.14

LA.1

REFUELING OPERATIONS

SPENT FUEL CASK DROP PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.14 The maximum weight of a spent fuel cask used with the Cask Drop Protection System shall be limited to 110 tons (nominal).

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14 The weight of a spent fuel cask shall be verified to be ≤ 110 tons (nominal) prior to its use with the Cask Drop Protection System.

DISCUSSION OF CHANGES
CTS 3/4.9.14, SPENT FUEL CASK DROP PROTECTION SYSTEM

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 6 – Relocation of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPP, or IIP)* CTS LCO 3.9.14 specifies that the maximum weight of a spent fuel cask used with the Cask Drop Protection System be limited to 110 tons (nominal). The ITS does not include this spent fuel cask weight limitation associated with the Cask Drop Protection System. This changes the CTS by moving the explicit spent fuel cask weight limitation associated with the Cask Drop Protection System, including the Action and Surveillance Requirement, from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of CTS LCO 3.9.14 is to ensure that limitations on the use of spent fuel casks weighing in excess of 110 tons (nominal) are in effect to provide assurance that the spent fuel pool would not be damaged by a dropped fuel cask since this weight is consistent with the assumptions used in the safety analyses for the performance of the Cask Drop Protections System. These requirements are proposed to be relocated to the TRM since the movement of loads other than fuel assemblies is controlled based on the heavy loads analysis. The bounding design basis fuel handling accident in the auxiliary building assumes a single irradiated fuel assembly is damaged. In addition, as stated in the NRC Safety Evaluation for License Amendments 197 (Unit 1) and 182 (Unit 2), dated July 12, 1995, the controls in place ensure that the potential for other, more severe events that could occur, such as a heavy load drop on irradiated fuel, need not be postulated and analyzed. This change is acceptable because the removed information will be adequately controlled in 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 change because a requirement is being removed from the Technical Specifications.

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DISCUSSION OF CHANGES
CTS 3/4.9.14, SPENT FUEL CASK DROP PROTECTION SYSTEM

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.9.14, SPENT FUEL CASK DROP PROTECTION SYSTEM

There are no specific NSHC discussions for this Specification.

ATTACHMENT 8

Improved Standard Technical Specifications (ISTS) not adopted in the CNP ITS

ISTS 3.9.2, Unborated Water Source Isolation Valves

ISTS 3.9.2 Markup and Justification for Deviations (JFDs)

1

3.9 REFUELING OPERATIONS

3.9.2 [Unborated Water Source Isolation Valves]

- REVIEWER'S NOTE -
This Technical Specification is not required for units that have analyzed a boron dilution event in MODE 6. It is required for those units that have not analyzed a boron dilution event in MODE 6. For units which have not analyzed a boron dilution event in MODE 6, the isolation of all unborated water sources is required to preclude this event from occurring.

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. - NOTE - Required Action A.3 must be completed whenever Condition A is entered. One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS. AND A.2 Initiate actions to secure valve in closed position. AND A.3 Perform SR 3.9.1.1.	Immediately Immediately 4 hours

(1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify each valve that isolates unborated water sources is secured in the closed position.	31 days

JUSTIFICATION FOR DEVIATIONS
ISTS 3.9.2, UNBORATED WATER SOURCE ISOLATION VALVES

1. CNP has analyzed a boron dilution event in MODE 6. Isolation of all unborated water sources in MODE 6 is not required. Therefore, ISTS 3.9.2 is not included in the ITS.

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ISTS 3.9.2 Bases Markup and Justification for Deviations (JFDs)

1

B 3.9 REFUELING OPERATIONS

B 3.9.2 [Unborated Water Source Isolation Valves]

BASES

BACKGROUND	During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.
APPLICABLE SAFETY ANALYSES	The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6. The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.
APPLICABILITY	In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS. For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.

1

BASES

ACTIONS

The ACTIONS Table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. FSAR, Section [15.2.4].
2. NUREG-0800, Section 15.4.6.

JUSTIFICATION FOR DEVIATIONS
ISTS 3.9.2 BASES, UNBORATED WATER SOURCE ISOLATION VALVES

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

SUMMARY OF CHANGES
ITS CHAPTER 4.0

Change Description	Affected Pages
A self-identified change to ITS 4.3.1.2.a has been made. This change makes an editorial revision to ITS 4.3.1.2.a.	Page 38 of 45.

VOLUME 15

CNP UNITS 1 AND 2
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 4.0
DESIGN FEATURES

Revision 1

LIST OF ATTACHMENTS

1. ITS Chapter 4.0

ATTACHMENT 1

ITS Chapter 4.0, Design Features

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

A.1

ITS

4.0 5.0 DESIGN FEATURES

4.1 5.1 SITE

EXCLUSION AREA

4.1.1 5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

4.1.2 5.1.2 The low population zone shall be ~~as shown in Figure 5.1-2~~. all the land within a circle centered on the reactor containment structures and a radius of 2 miles A.2

Site Boundary For Gaseous and Liquid Effluents

4.1.1 5.1.3 The SITE BOUNDARY for gaseous and liquid effluents shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 160 feet.*

c. Minimum thickness of concrete walls = 3'6".

d. Minimum thickness of concrete roof = 2'6".

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner, side and dome = 3/8 inches.

g. Nominal thickness of steel liner, bottom = 1/4 inch.

h. Net free volume = 1.24×10^6 cubic feet.

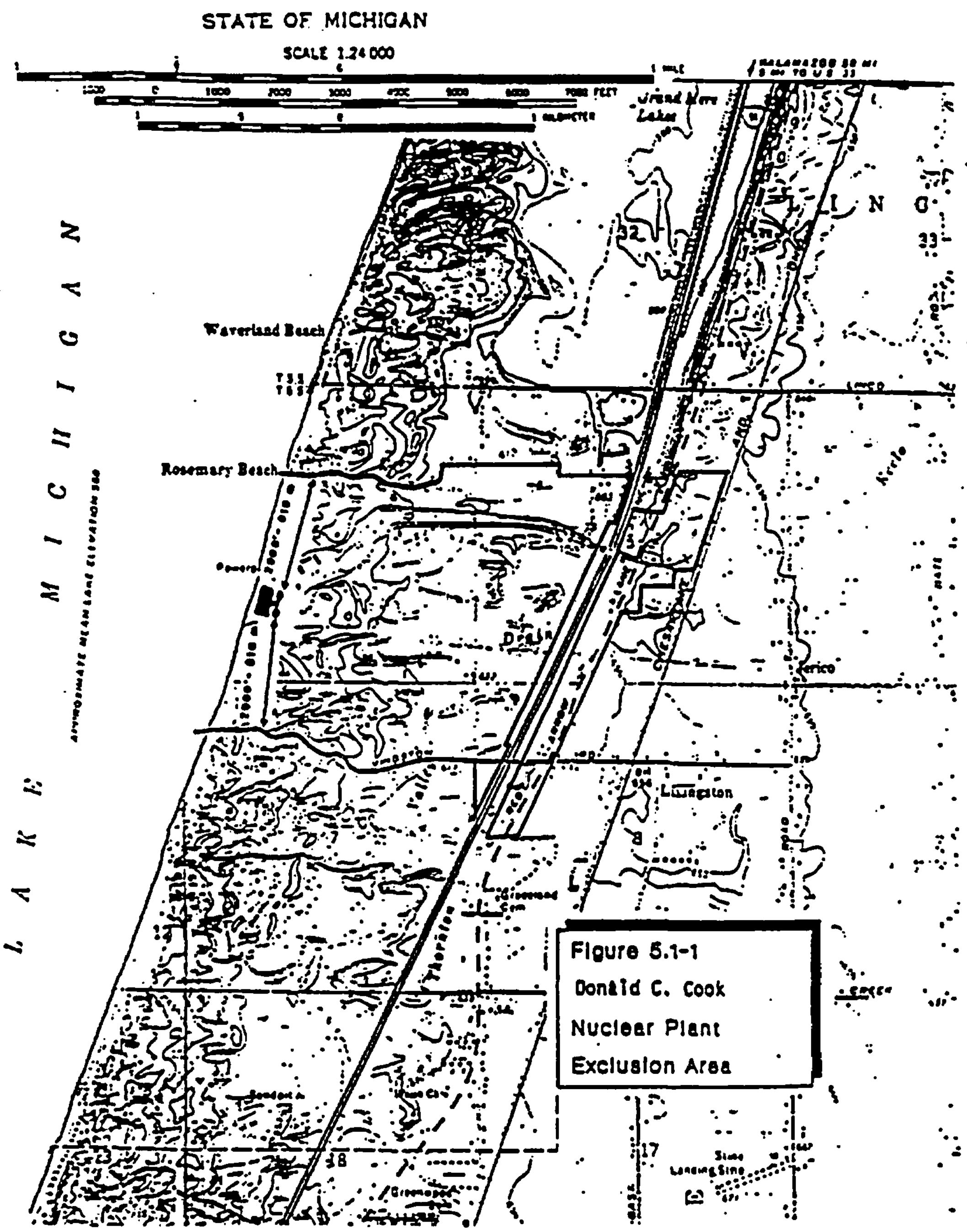
LA.1

*From grade (Elev. 608') to inside of dome.

A.1

ITS

Figure 4.1-1

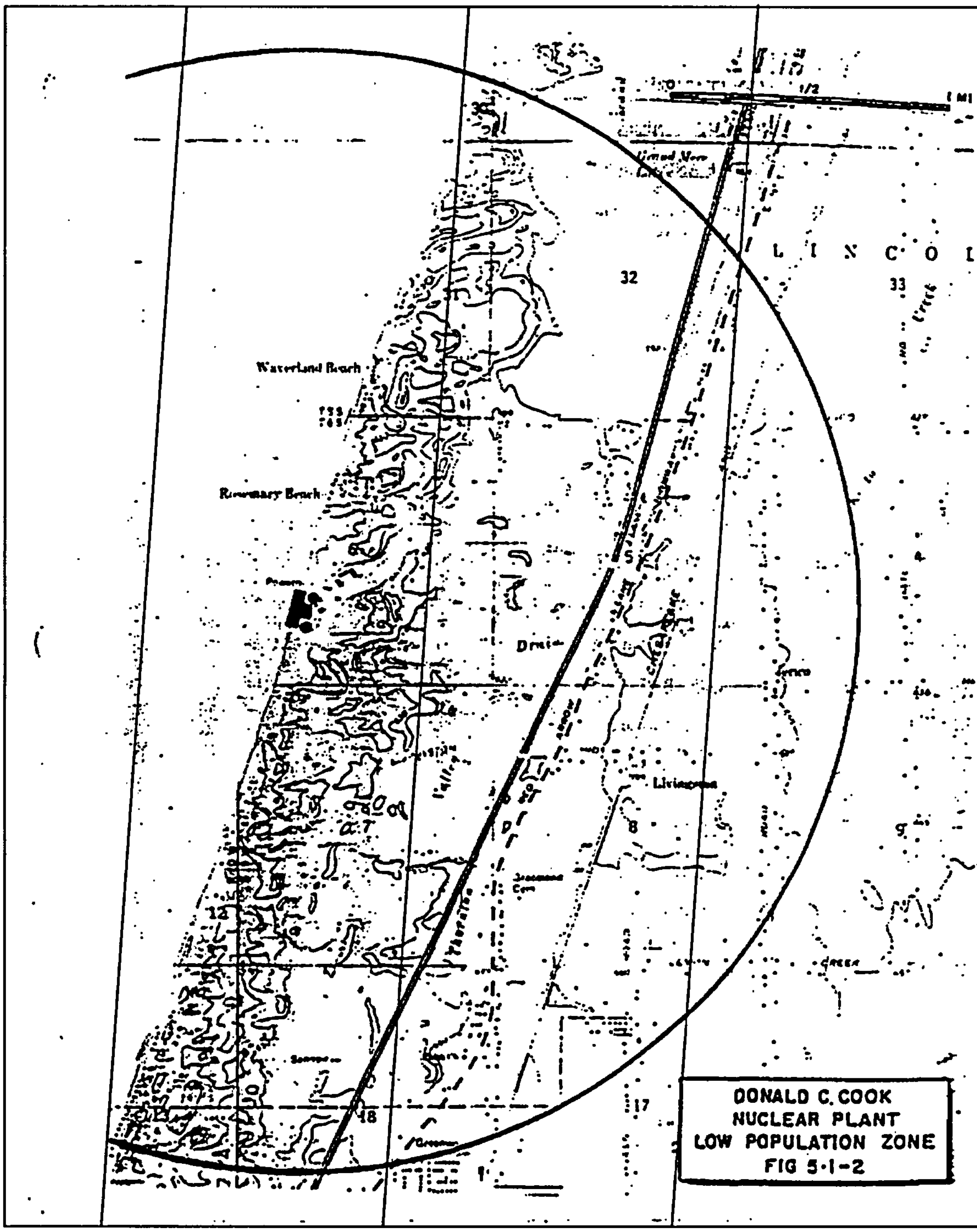


COOK NUCLEAR PLANT - UNIT 1 5-2 AMENDMENT NO. 155

ITS 4.0

A.1

ITS



A.2

A.1

ITS 4.0

ITS

5.0 DESIGN FEATURES

~~5.2 CONTAINMENT (Continued)~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.2.2 The reactor containment building is designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.2 of the FSAR.~~

~~PENETRATIONS~~

~~5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 5.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.~~

LA.1

4.2 5.3 REACTOR CORE

FUEL ASSEMBLIES

consisting of a matrix of

4.2.1 5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly ~~containing 204~~ fuel ~~rods~~ clad with Zircaloy-4 or ZIRLO, except that limited substitutions of zirconium alloy or stainless steel filler rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analysis to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. ~~Each fuel rod shall have a nominal active fuel length of 144 inches.~~ ~~The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235.~~ Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 4.95 weight percent U-235.

fuel rods with an initial composition of natural or slightly enriched UO_2 as fuel material

4.3.1.1.a

LA.2

CONTROL ROD ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. ~~The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.~~

The control material shall be silver indium cadmium, as approved by the NRC.

LA.3

A.1

ITS

5.0 DESIGN FEATURES

~~5.4 REACTOR COOLANT SYSTEM~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 The reactor coolant system is designed and shall be maintained:~~

~~a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,~~

~~b. For a pressure of 2485 psig, and~~

~~c. For a temperature of 650°F, except for the pressurizer which is 680°F.~~

LA.4

~~5.5 EMERGENCY CORE COOLING SYSTEMS~~

~~5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.~~

LA.5

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY – SPENT FUEL

4.3.1.1 5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

4.3.1.1.b a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water.

4.3.1.1.c b. A nominal 8.97 inch center-to-center distance between fuel assemblies placed in the storage racks.

See ITS 3.7.16

4.3.1.1.d, 4.3.1.1.e c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

See ITS 3.7.16

A.1

ITS

4.3.1.1.a, 4.3.1.1.d

1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

4.3.1.1.e

2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

See ITS 3.7.16

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

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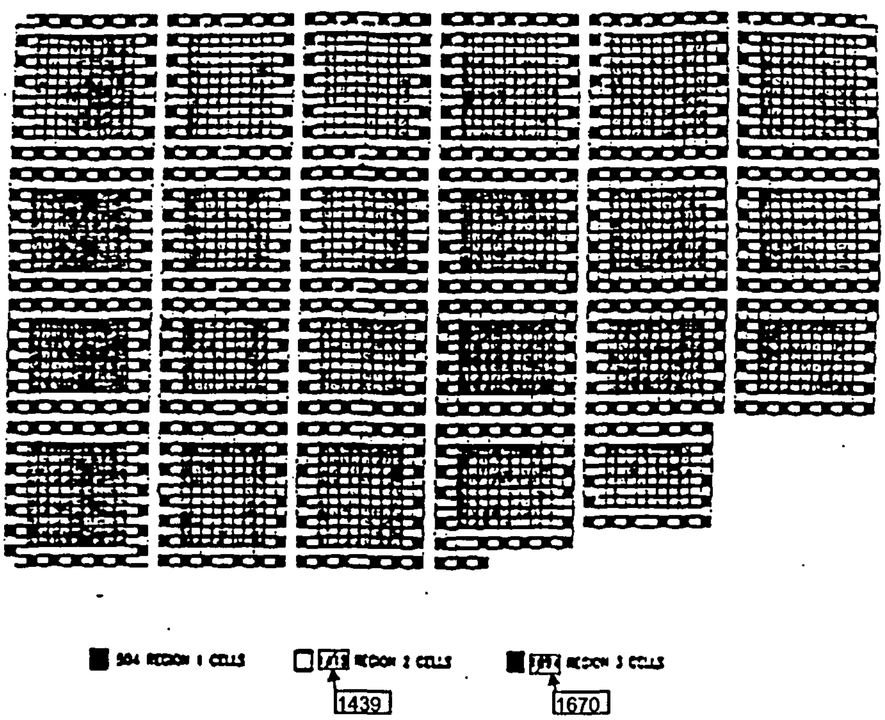
ITS 4.0

A.1

ITS

Figure 4.3-1

FIGURE 5.6-1: Normal Storage Pattern (Mixed Three Zone)



A.3

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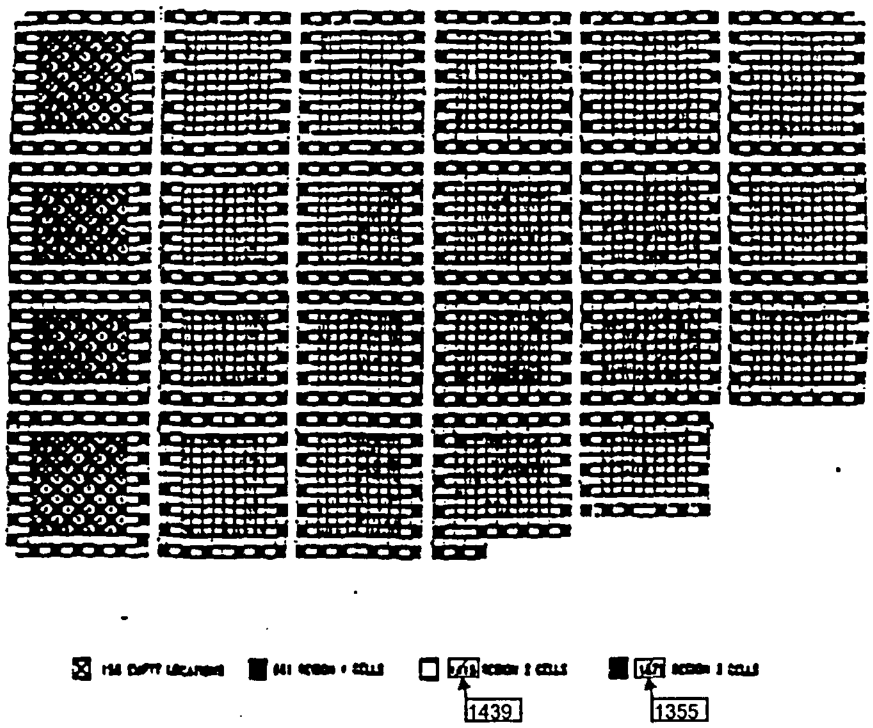
ITS 4.0

A.1

ITS

Figure 4.3-2

Figure 5.4-2: Interim Storage Pattern (Checkerboard)



A.3

COOK NUCLEAR PLANT - UNIT 1 5-7a AMENDMENT NO. 169

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ITS 4.0

A.1

ITS

5.0 DESIGN FEATURES

Figure 5.6-3 Intentionally deleted.

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A.1

ITS

5.0 DESIGN FEATURES

5.6 FUEL STORAGE (Continued)

CRITICALITY - NEW FUEL

4.3.1.2 5.6.2 The new fuel storage racks are designed and shall be maintained with:

4.3.1.2.a a. Westinghouse fuel assemblies having either a maximum enrichment of 4.55 weight % U-235, or an enrichment between 4.55 and 4.95 weight % U-235 with greater than or equal to the minimum number of integral fuel burnable absorber pins as shown on Figure 5.6-4(interpolation of the Boron-10 loading between 1.0X and 1.5X and between 1.5X and 2.0X is acceptable);

4.3.1.2.b b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR;

4.3.1.2.c c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR; and

4.3.1.2.d d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

DRAINAGE

4.3.2 5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629'4".

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ITS 4.0

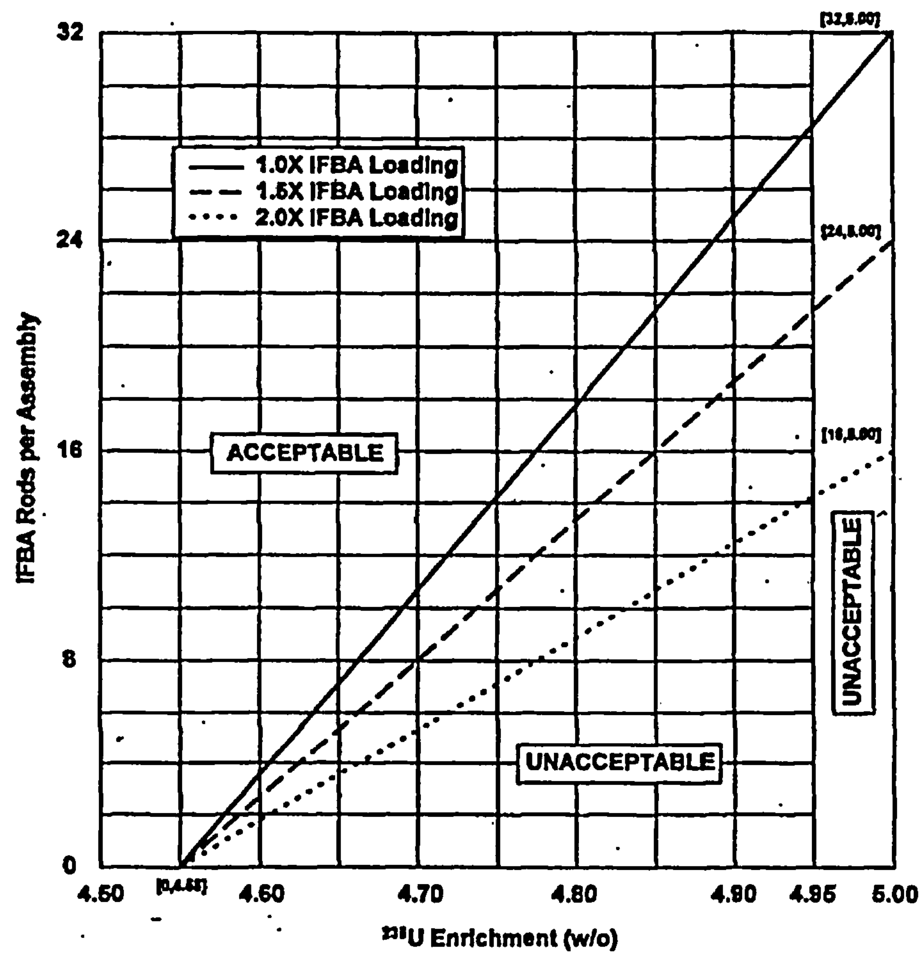
A.1

<u>ITS</u>

Figure 4.3-3

5.0 DESIGN FEATURES

Figure 5.6-4: New Fuel Storage Rack Integral Fuel Burnable Absorber (IFBA) Requirements



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A.1

ITS

5.0 DESIGN FEATURES

5.6 FUEL STORAGE (Continued)

CAPACITY

4.3.3 5.6.4 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3613 fuel assemblies.

~~5.7 SEISMIC CLASSIFICATION~~

~~5.7.1 Those structures, systems and components identified as Category I Items in the FSAR shall be designed and maintained to the original design provisions contained in the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.~~ LA.6

~~5.8 METEOROLOGICAL TOWER LOCATION~~

~~5.8.1 The meteorological tower shall be located as shown on Figure 5.1-3.~~ LA.7

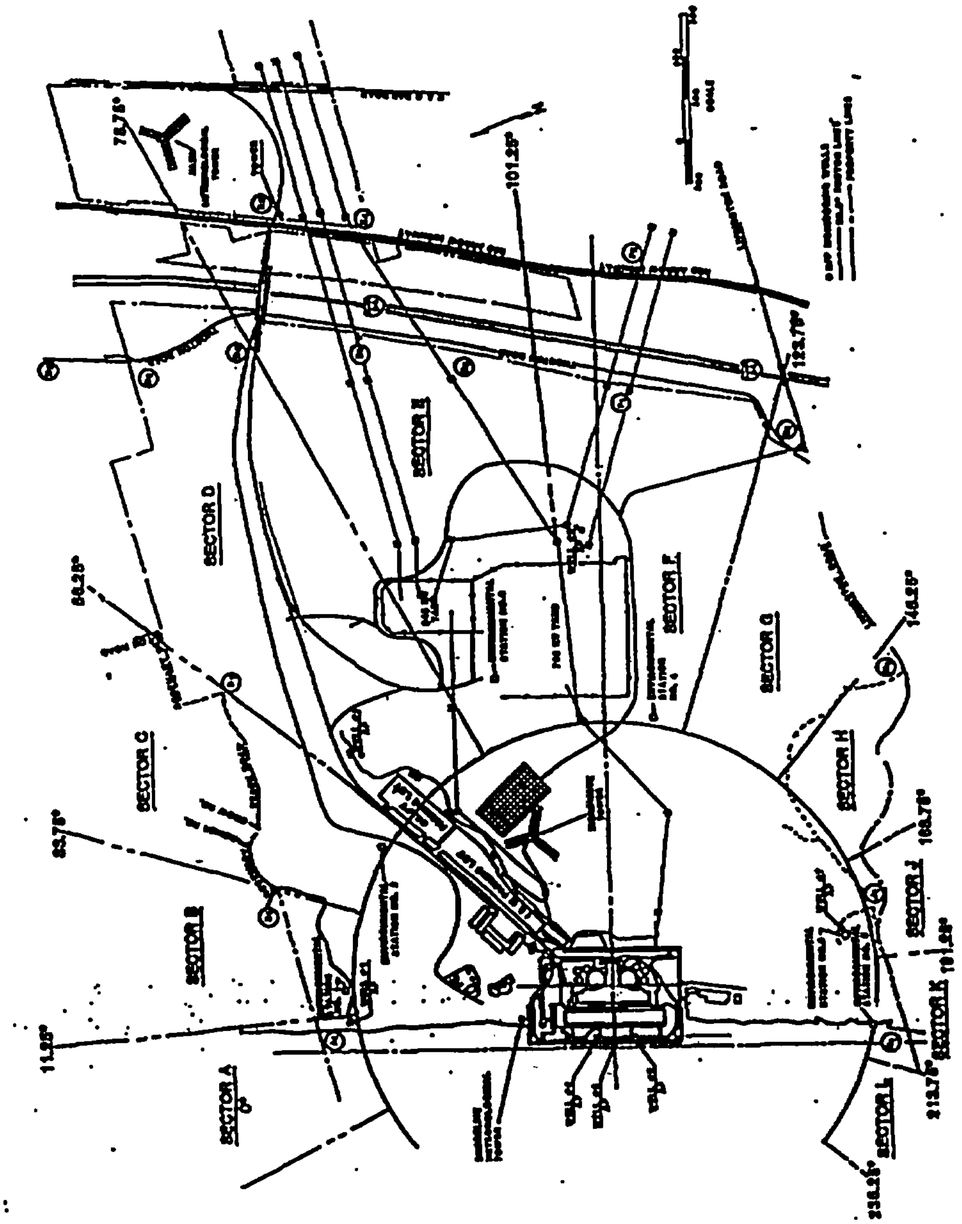
A.1

ITS

Figure 4.1-1

5.0 - DESIGN FEATURES

FIGURE 5.1-3: SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS



A.1

ITS

4.0 5.0 DESIGN FEATURES

4.1 5.1 SITE

Exclusion Area

4.1.1 5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

Low Population Zone

4.1.2 5.1.2 The low population zone shall be ~~as shown in Figure 5.1-2~~. all the land within a circle centered on the reactor containment structures and a radius of 2 miles A.2

Site Boundary For Gaseous and Liquid Effluents

4.1.1 5.1.3 The SITE BOUNDARY for gaseous and liquid effluents shall be as shown in Figure 5.1-3.

~~5.2 CONTAINMENT~~

~~CONFIGURATION~~

~~5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:~~

~~a. Nominal inside diameter = 115 feet.~~

~~b. Nominal inside height = 160 feet.~~

~~c. Minimum thickness of concrete walls = 3'6".~~

~~d. Minimum thickness of concrete roof = 2'6".~~

~~e. Minimum thickness of concrete floor pad = 10 feet.~~

~~f. Nominal thickness of steel liner = 3/8 inches.~~

~~g. Net free volume = 1.24×10^6 cubic feet.~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.2.2 The reactor containment building is designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.2 of the FSAR.~~

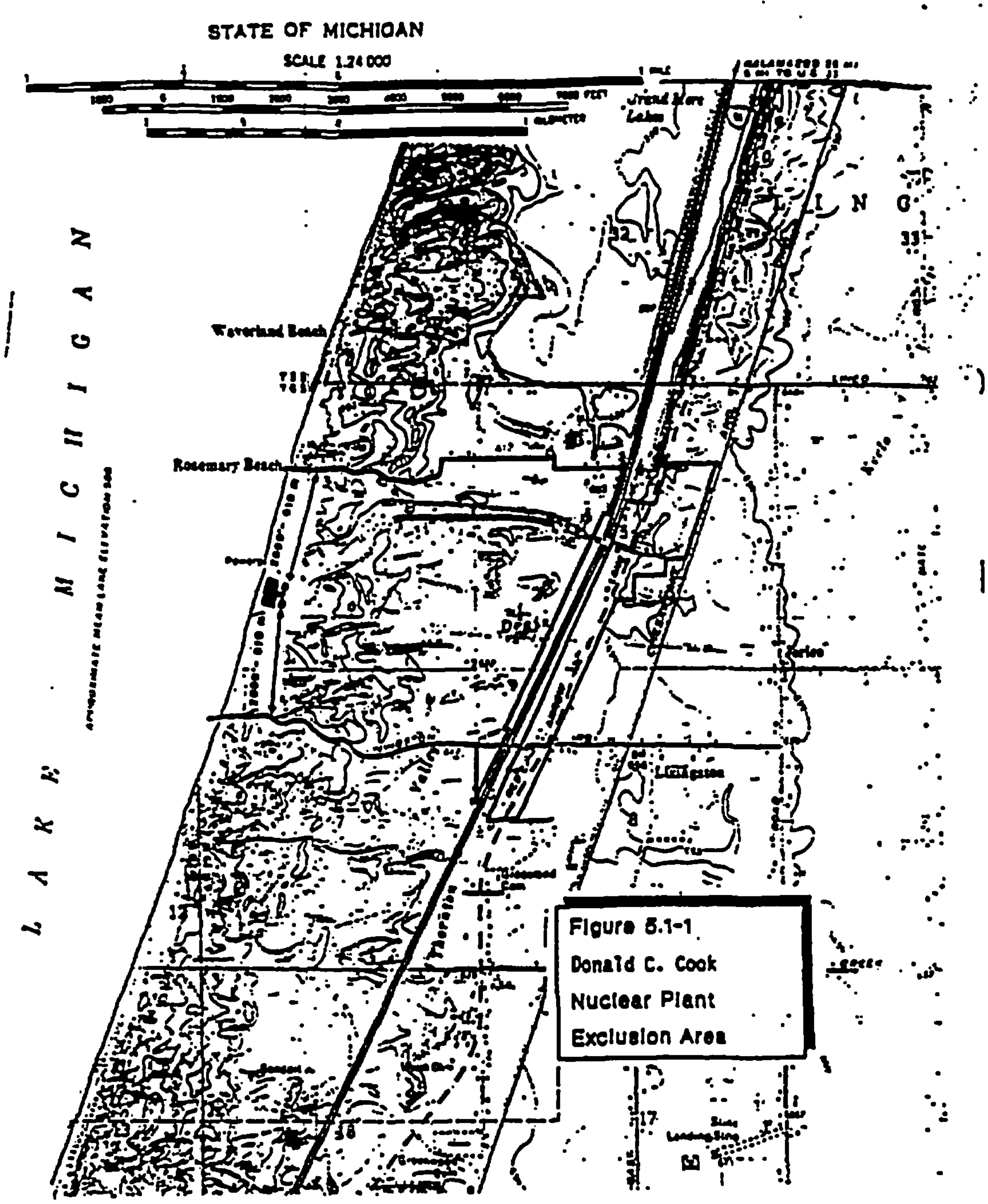
LA.1

A.1

ITS 4.0

ITS

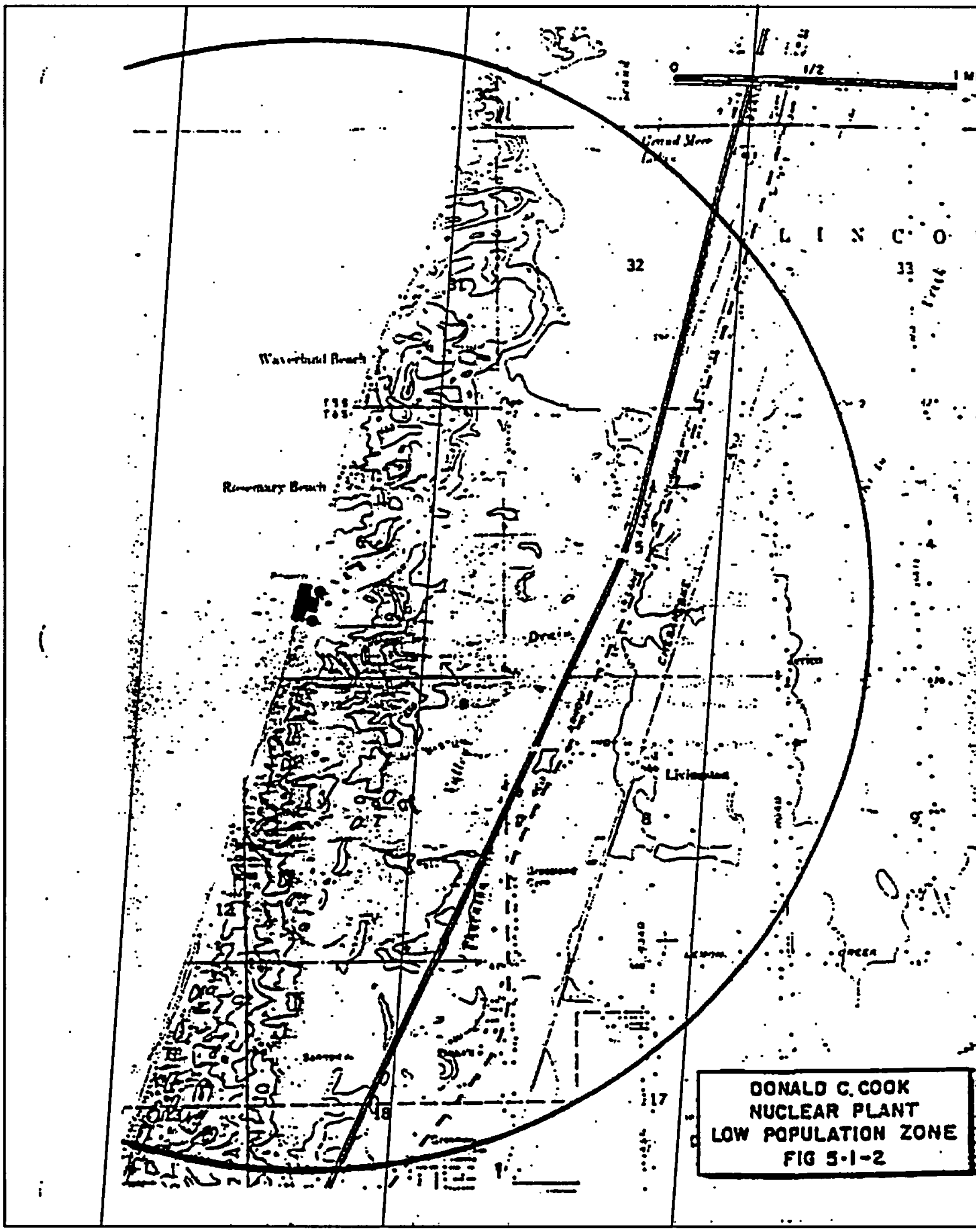
Figure 4.1-1



ITS 4.0

A.1

ITS



DONALD C. COOK NUCLEAR PLANT LOW POPULATION ZONE FIG 5-1-2

A.2

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A.1

ITS

5.0 DESIGN FEATURES

4.2 5.3 REACTOR CORE

FUEL ASSEMBLIES

consisting of a matrix of

4.2.1 5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly ~~containing 264 fuel rods~~ clad ~~with~~ Zircaloy ~~-4~~ or ZIRLO, except that limited substitutions of zirconium alloy or stainless steel filler rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. ~~Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.3 weight percent U-235.~~ Reload fuel shall be similar in physical design to the initial core loading and may be nominally enriched up to 4.95 weight percent U-235.

fuel rods with an initial composition of natural or slightly enriched UO_2 as fuel material

4.3.1.1.a

LA.2

CONTROL ROD ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. ~~The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.~~

The control material shall be silver indium cadmium, as approved by the NRC.

LA.3

~~5.4 REACTOR COOLANT SYSTEM~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 The reactor coolant system is designed and shall be maintained:~~

~~a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.~~

~~b. For a pressure of 2485 psig, and~~

~~c. For a temperature of 650°F, except for the pressurizer which is 680°F.~~

LA.4

A.1

ITS

5.0 DESIGN FEATURES

~~5.5 METEOROLOGICAL TOWER LOCATION~~

~~5.5.1 The meteorological tower shall be located as shown on Figure 5.1-3.~~ LA.7

4.3 5.6 FUEL STORAGE

4.3.1 CRITICALITY - SPENT FUEL

4.3.1.1 5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

4.3.1.1.b a. A K_{eff} equivalent to less than 0.95 when flooded with unborated water,

4.3.1.1.c b. A nominal 8.97-inch center-to-center distance between fuel assemblies, placed in the storage racks.

See ITS 3.7.16

4.3.1.1.d, 4.3.1.1.e c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

See ITS 3.7.16

4.3.1.1.a, 4.3.1.1.d 1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

4.3.1.1.e 2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MTU, or fuel of other enrichments with equivalent reactivity.

3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MTU, or fuel of other enrichments with equivalent reactivity.

See ITS 3.7.16

ITS 4.0

A.1

ITS

5.0 DESIGN FEATURES

4.3.1.1.e

5.6 FUEL STORAGE (Continued)

CRITICALITY - SPENT FUEL (Continued)

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

See ITS 3.7.16

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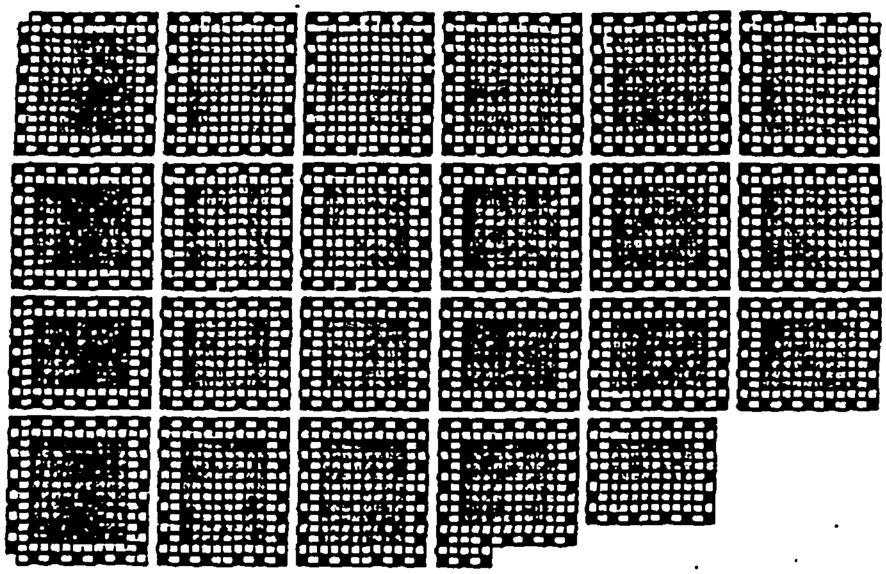
ITS 4.0

A.1

ITS

Figure 4.3-1

FIGURE 5.6-1: Normal Storage Pattern (Mixed Three Zone)



304 REGION 1 CELLS   1439 REGION 2 CELLS   1670 REGION 3 CELLS

A.3

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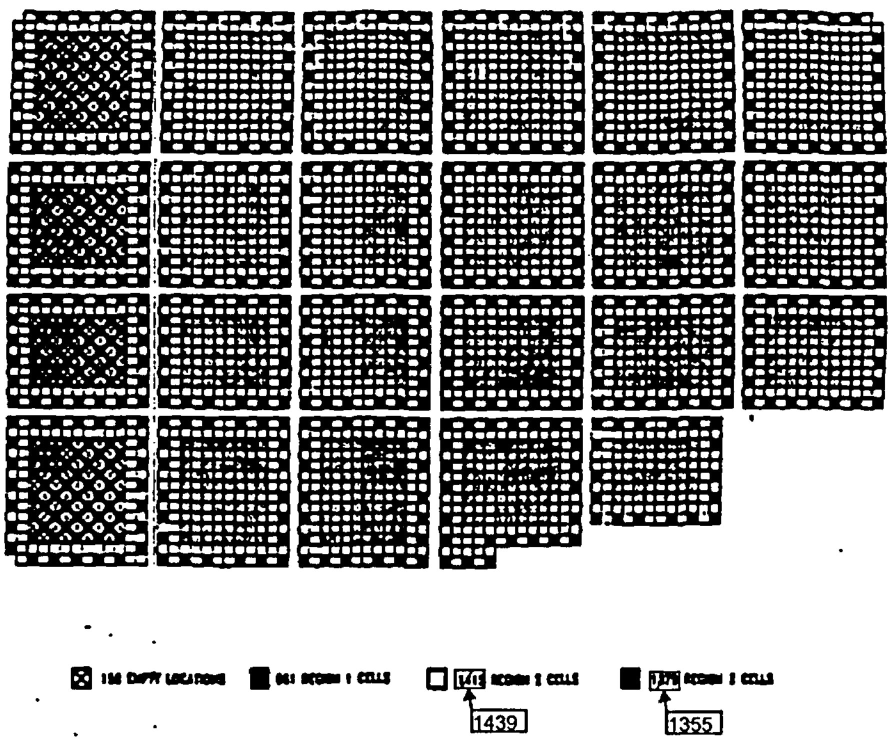
ITS 4.0

A.1

ITS

Figure 4.3-2

Figure 5.6-2: Interim Storage Pattern (Checkerboard)



A.3

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A.1

ITS

5.0 DESIGN FEATURES

Figure 5.6-3 intentionally deleted.

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ITS 4.0

A.1

ITS

5.0 ADMINISTRATIVE CONTROLS

5.6 FUEL STORAGE (Continued)

4.3.1.2 5.6.2 The new fuel storage racks are designed and shall be maintained with:

4.3.1.2.a a. Westinghouse fuel assemblies having either a maximum enrichment of 4.55 weight % U-235, or an enrichment between 4.55 and 4.95 weight % U-235 with the minimum number of integral fuel burnable absorber pins as shown on Figure 5.6-4 (interpolation of the Boron-10 loading between 1.0X and 1.5X and 2.0X is acceptable);

4.3.1.2.b b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR;

4.3.1.2.c c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR; and

4.3.1.2.d d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

DRAINAGE

4.3.2 5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629'4".

CAPACITY

4.3.3 5.6.4 The spent fuel storage pool is designed and shall me maintained with a storage capacity limited to no ı than 3613 fuel assemblies.

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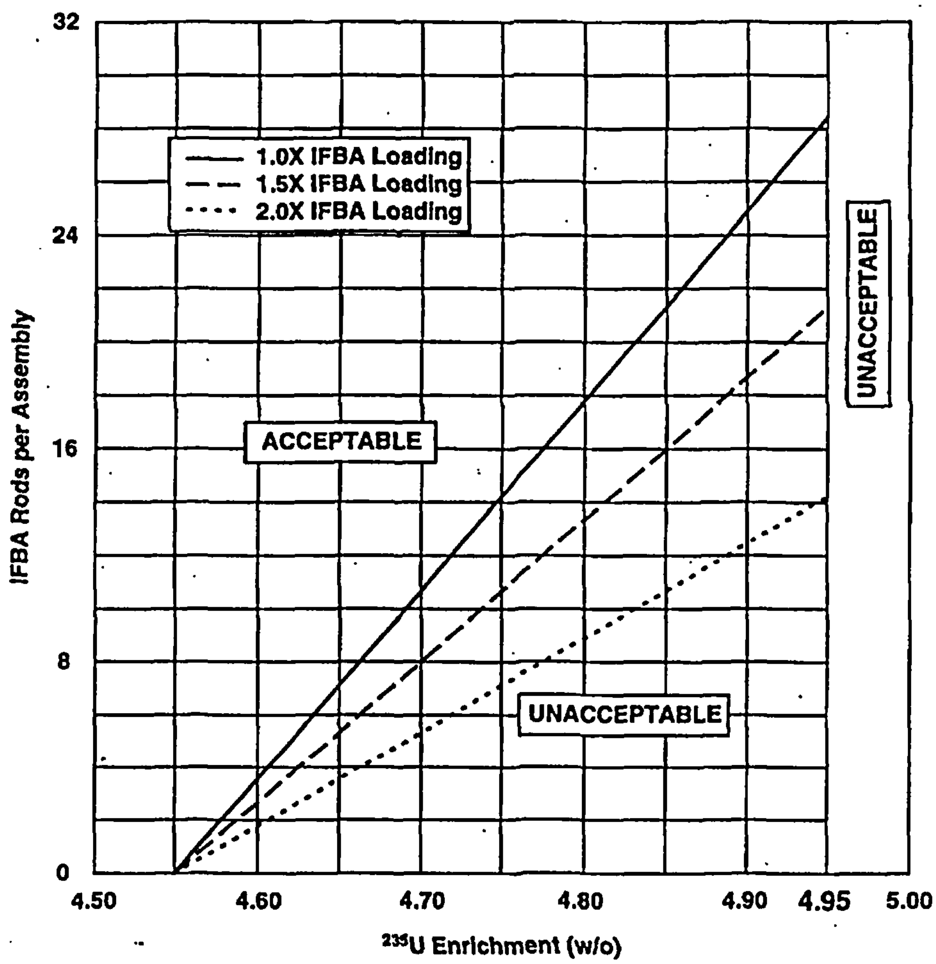
A.1

ITS

Figure 4.3-3

Figure 5.6-4: New Fuel Storage Rack Integral Fuel Burnable Absorber (IFBA) Requirements



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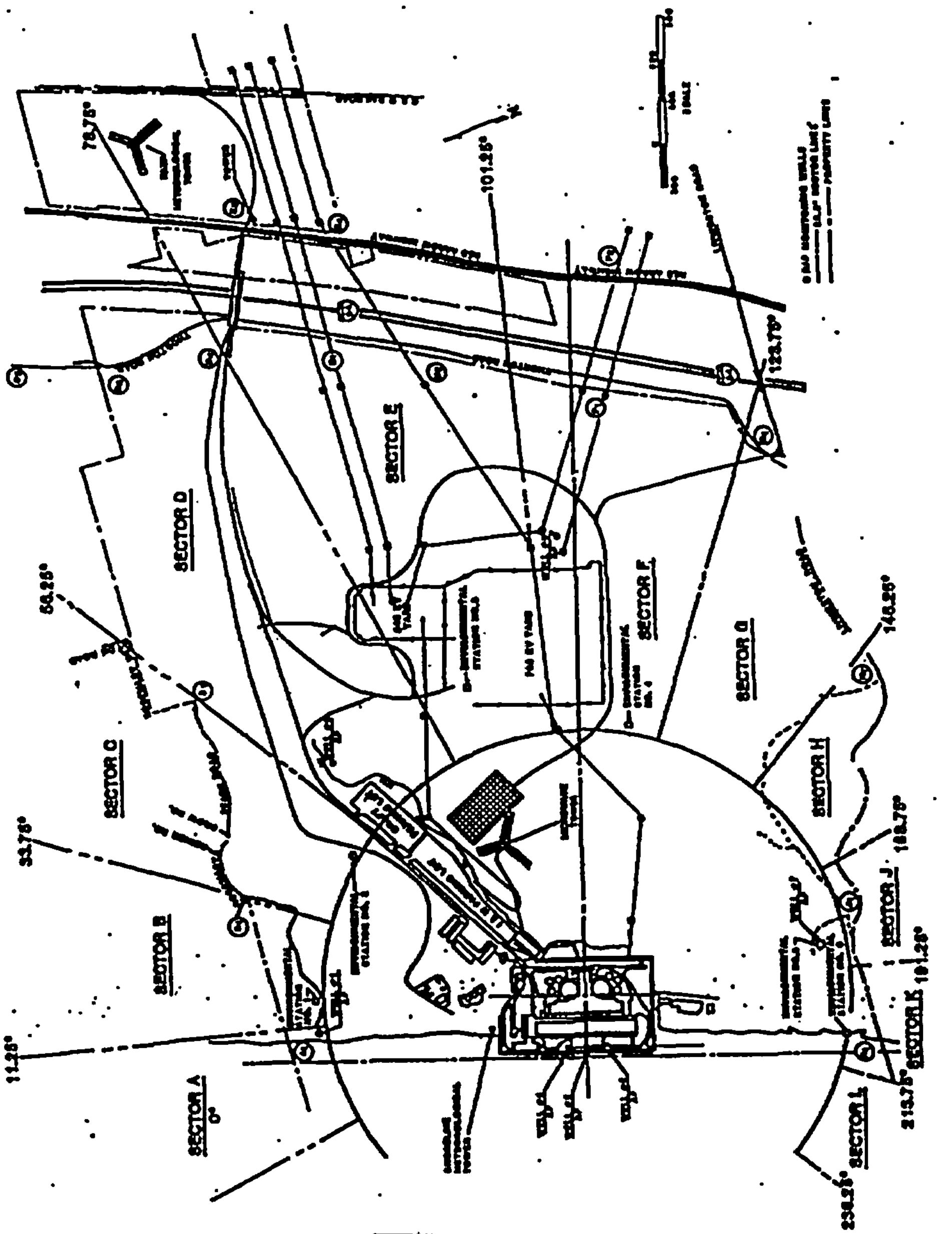
A.1

ITS

Figure 4.1-1

5.0 DESIGN FEATURES

FIGURE 5.1-3: SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS



DISCUSSION OF CHANGES
ITS CHAPTER 4.0, DESIGN FEATURES

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 5.1.2 states "The low population zone shall be as shown in Figure 5.1-2." CTS Figure 5.1-2 provides a map depicting the low population zone. ITS 4.1.2 provides a description of the low population zone; a figure is not provided. This changes the CTS by providing a word description of the low population zone instead of a map.

This change is acceptable since it does not change the current requirements. A description is provided consistent with the current map in the figure. This change is designated as administrative because it does not result in a technical change to the Technical Specifications.

A.3 CTS Figures 5.6-1 and 5.6-2 provide drawings that depict the various regions of the spent fuel storage pool racks for a normal storage pattern (mixed three zone) and for an interim storage pattern (checkerboard). The key at the bottom of the figures identifies the total number of cells for the various regions. The CTS Figure 5.6-1 key identifies, in part, that there are 1415 Region 2 cells and 1694 Region 3 cells, and the CTS Figure 5.6-2 key identifies, in part, that there are 1415 Region 2 cells and 1379 Region 3 cells. The ITS Figure 4.3-1 key identifies that there are 1439 Region 2 cells and 1670 Region 3 cells, and the ITS Figure 4.3-2 key identifies that there are 1439 Region 2 cells and 1355 Region 3 cells. This changes the keys to clearly identify the actual number of cells depicted in each region.

This change is acceptable since it does not change the current requirements. The number of cells listed in the keys for the ITS Figures is consistent with the actual number of cells depicted by the Figures. This change is considered administrative because it does not result in a technical change to the Technical Specifications.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS CHAPTER 4.0, DESIGN FEATURES

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 5.2 describes the various design features of the reactor containment building. The ITS does not contain this information. This changes the CTS by moving the description of the reactor containment building to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements on containment OPERABILITY in ITS 3.6.1. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), 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.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 5.3.1 contains details of fuel assembly design, such as number of fuel rods per fuel assembly, the fuel rod nominal active fuel length, and the initial core loading maximum enrichment. The ITS does not contain these details, but provides a general statement that, "Each assembly shall consist of a matrix of Zircaloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material." This changes the CTS by moving the detailed description of the fuel assemblies to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements on fuel assembly enrichment in ITS 4.2.1. In addition, core power distribution requirements, which are dependant upon fuel assembly design, are described in ITS Section 3.2. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), 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.

LA.3 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 5.3.2 contains details of control rod design, such as the nominal length of absorber material, percentage of each absorber material, and control rod cladding material. The ITS does not contain these details, but provides a general statement that, "The control material shall be silver indium cadmium, as approved by the NRC." This changes the CTS by moving the detailed description of the control rod assemblies to the UFSAR.

DISCUSSION OF CHANGES
ITS CHAPTER 4.0, DESIGN FEATURES

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements on the control rod material in ITS 4.2.2 and on control rod OPERABILITY in ITS Section 3.1. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), 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.

LA.4 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 5.4 describes the Reactor Coolant System. The ITS does not contain this information. This changes the CTS by moving the description of the Reactor Coolant System to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements on Reactor Coolant System OPERABILITY in ITS Section 3.4. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), 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.

LA.5 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* (Unit 1 only) Unit 1 CTS 5.5 describes the Emergency Core Cooling Systems (ECCS). The ITS does not contain this information. This changes the Unit 1 CTS by moving the description of the ECCS to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements on ECCS OPERABILITY in ITS Section 3.5. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), 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.

LA.6 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* (Unit 1 only) Unit 1 CTS 5.7 describes certain general Seismic Classification requirements. The ITS does not contain this information. This changes the Unit 1 CTS by moving the description of these general Seismic Classification requirements to the UFSAR.

DISCUSSION OF CHANGES
ITS CHAPTER 4.0, DESIGN FEATURES

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements for various Category I structures, systems, and components. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), 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.

LA.7 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 5.8.1 (Unit 1) and CTS 5.5.1 (Unit 2) describes the location of the meteorological tower. The ITS does not contain this information. This changes the CTS by moving the location of the meteorological tower to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. 10 CFR 50.36(c)(4) states "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of this section." These paragraphs provide the criteria for safety limits, limiting safety system settings, and limiting control settings; limiting conditions for operation; and surveillance requirements to be included in the Technical Specifications, respectively. The location of the meteorological tower does not meet any of these requirements. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), 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

None

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

CTS

4.0 DESIGN FEATURES

4.1 Site Location

5.1.1, 5.1.2, 5.1.3 ~~[Text description of site location.]~~ INSERT 1 (1)

4.2 Reactor Core

5.3.1 4.2.1 Fuel Assemblies (1)

The reactor shall contain ~~157~~ 193 fuel assemblies. Each assembly shall consist of a matrix of [Zircalloy or ZIRLO] fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. (1)

5.3.2 4.2.2 [Control Rod] Assemblies (1)

The reactor core shall contain ~~[48]~~ 53 full length [control rod] assemblies. The control material shall be [silver indium cadmium, ~~boron carbide, or hafnium metal~~] as approved by the NRC. (1)

4.3 Fuel Storage

5.6

5.6.1.1 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

5.3.1, 5.6.1.1.c.1 a. Fuel Assemblies having a maximum Nominal U-235 enrichment of ~~[4.5]~~ 4.95 weight percent; (1) (2)(3)

5.6.1.1a b. $k_{eff} \le 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1] 4 7.2 of the FSAR; (2)(1)

5.6.1.1.b [c. A nominal ~~[9.15]~~ 8.97 inch center to center distance between fuel assemblies placed in [the ~~high density~~ fuel storage racks] ; (1) (2)

~~[d. A nominal [10.95] inch center to center distance between fuel assemblies placed in [low density fuel storage racks].]~~ (4)

1 **INSERT 1**

4.1.1 Site and Exclusion Area Boundaries

The site area and exclusion area boundaries are as shown in Figure 4.1-1.

4.1.2 Low Population Zone

The low population zone is all the land within a circle centered on the reactor containment structures and a radius of 2 miles.

CTS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

5.6.1.1.c, 5.6.1.1.c.1 — d. New or partially spent fuel assemblies with any discharge burnup ~~in the "acceptable range" of Figure [3.7.17-1]~~ may be allowed unrestricted storage in ~~[either]~~ ~~fuel storage rack(s)~~; and INSERT 2 — 1 4 2

5.6.1.1.c, 5.6.1.1.c.2, 5.6.1.1.c.3 — e. ~~New or~~ partially spent fuel assemblies ~~with a discharge burnup in the "unacceptable range" of Figure [3.7.17-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].~~ INSERT 3 — 4 1

5.6.2 — 4.3.1.2 The new fuel storage racks are designed and shall be maintained with: Westinghouse either — 3

5.6.2.a — a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 4.55 weight percent, INSERT 4 — 1 3

5.6.2.b — b. $k_{eff} \le 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.[ ] of the FSAR; — u 7 — 1 2

5.6.2.c — c. $k_{eff} \le 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.[ ] of the FSAR; and — u 7 — 1 2

5.6.2.d — d. A nominal ~~[10.95]~~ 21 inch center to center distance between fuel assemblies placed in the storage racks. — 1

5.6.3 — 4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23 ft]~~ 629 ft 4 inches. — 1

5.6.4 — 4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1737]~~ 3613 fuel assemblies. — 1

INSERT 5 — 1

INSERT 6 — 1

INSERT 7 — 3

1 **INSERT 2**

Region 1 of Figure 4.3-1 or Figure 4.3-2;

1 **INSERT 3**

Regions 2 and 3 of Figure 4.3-1 or Figure 4.3-2 meeting the initial enrichment and burnup requirements of LCO 3.7.16, "Spent Fuel Pool Storage."

3 **INSERT 4**

or a maximum U-235 enrichment within the Acceptable Region of Figure 4.3-3 not to exceed 4.95 weight percent. Linear interpolation of the Boron-10 integral fuel burnable absorber (IFBA) loading curves between 1.0X and 1.5X and between 1.5X and 2.0X is acceptable;

1 INSERT 5

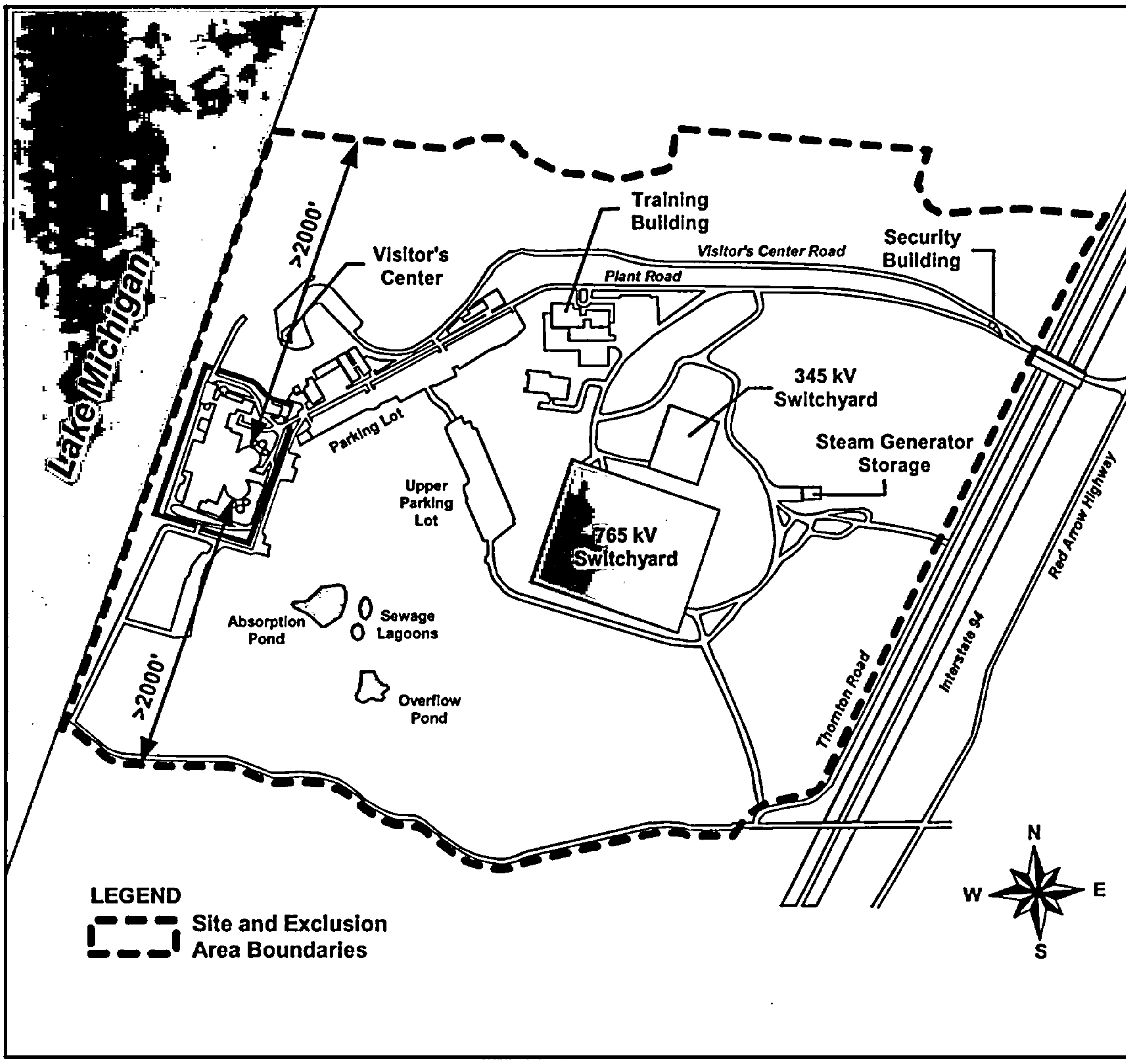
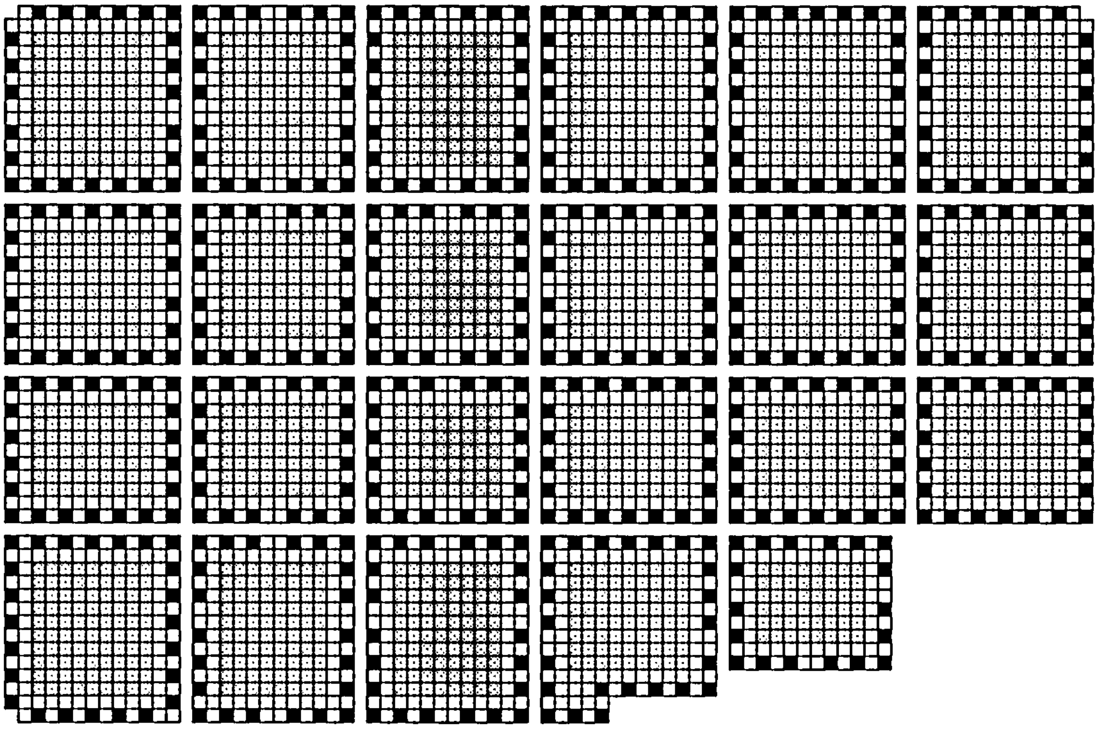


Figure 4.1-1 (Page 1 of 1)
Site and Exclusion Area Boundaries

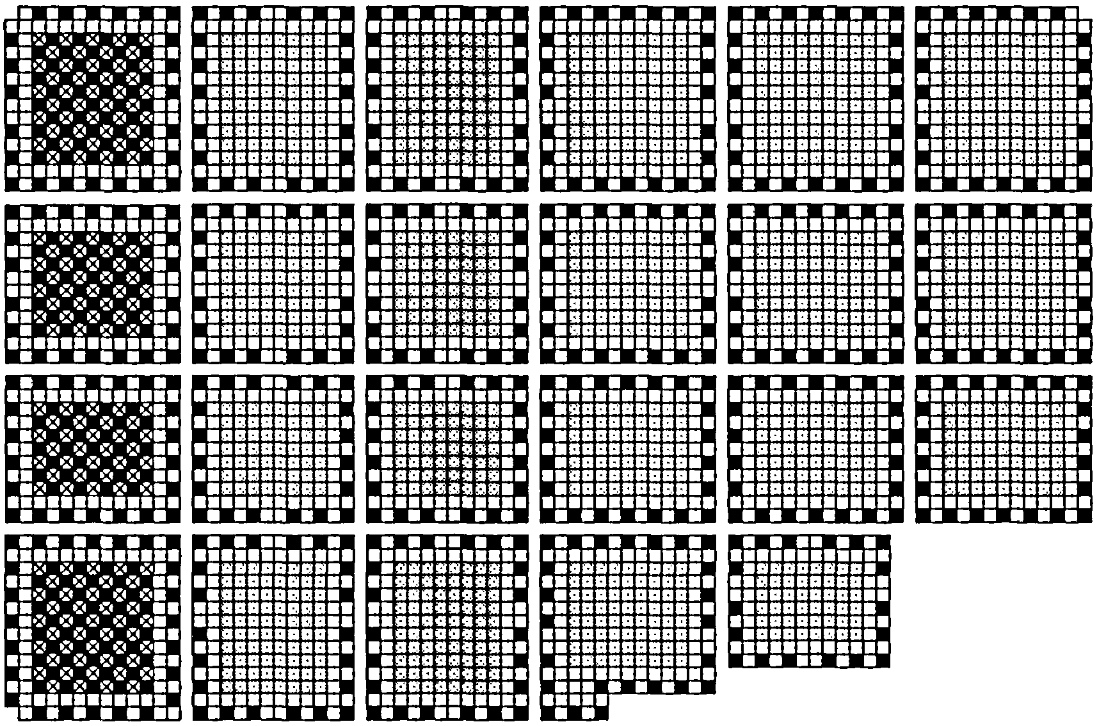
1 **INSERT 6**



■ 504 REGION 1 CELLS □ 1439 REGION 2 CELLS ⊡ 1670 REGION 3 CELLS

Figure 4.3-1 (Page 1 of 1)
Normal Storage Pattern (Mixed Three Zone)

1 **INSERT 6 (continued)**



⊠ 158 EMPTY LOCATIONS ■ 661 REGION 1 CELLS □ 1439 REGION 2 CELLS ⊡ 1355 REGION 3 CELLS

Figure 4.3-2 (Page 1 of 1)
Interim Storage Pattern (Checkerboard)

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3

INSERT 7

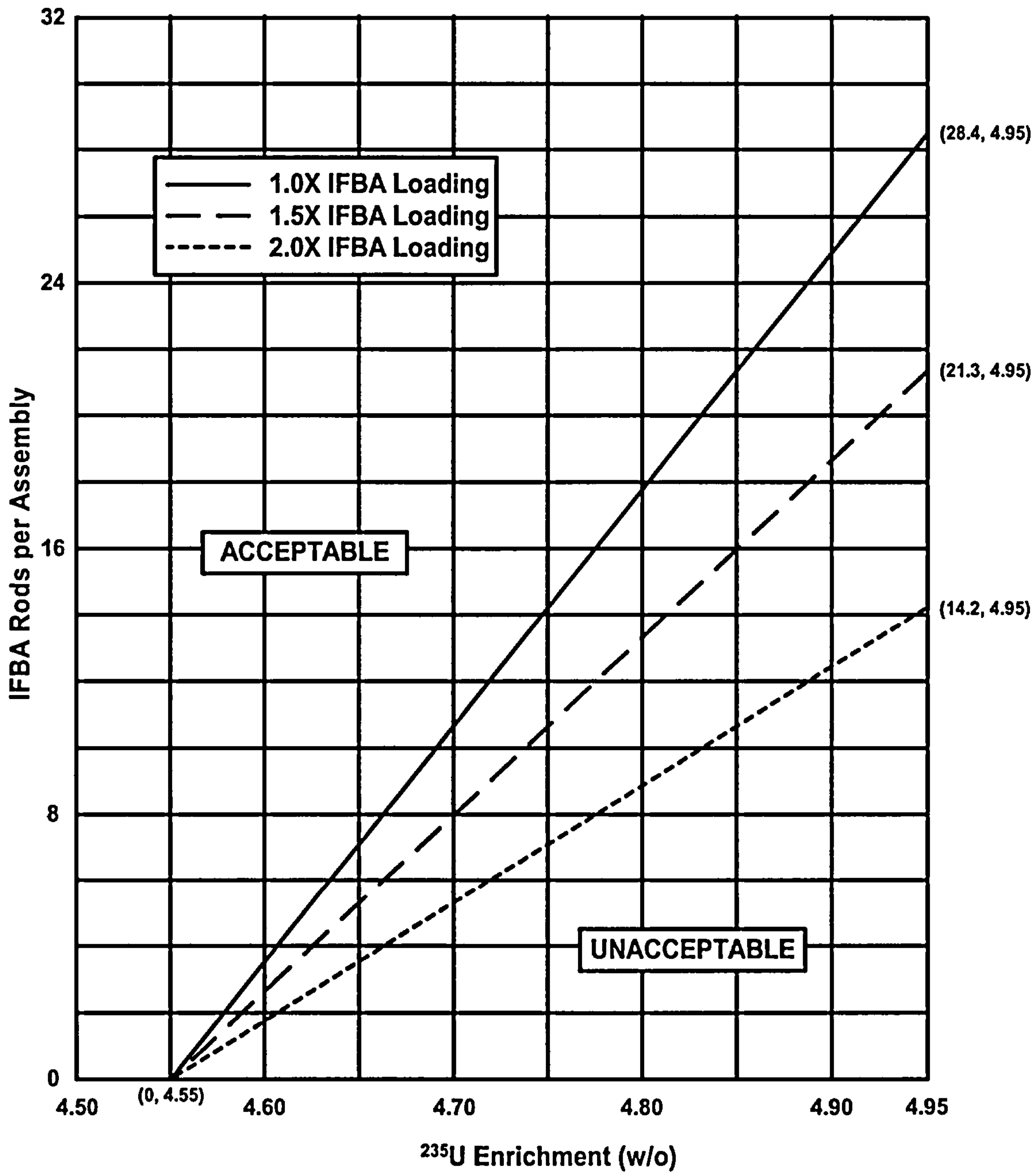

Figure 4.3-3 (Page 1 of 1)
New Fuel Storage Rack Integral Fuel Burnable Absorber (IFBA) Requirements

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JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 4.0, DESIGN FEATURES

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

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

3. 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.

4. ISTS 4.3.1.1.d, a bracketed requirement, has not been included in the ITS because low density fuel racks are not used in the CNP spent fuel storage pool. Subsequent Specifications have been renumbered, as appropriate, due to this deletion.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 4.0, DESIGN FEATURES

There are no specific NSHC discussions for this Chapter.

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SUMMARY OF CHANGES
ITS CHAPTER 5.0

Change Description	Affected Pages
A self-identified change for ITS 5.5 and 5.6 has been made. CTS Amendments 281 (Unit 1) and 265 (Unit 2) have been incorporated into the ITS submittal. This CTS change adopted the allowances of TSTF-359 and deletes CTS 3.4.10.1 Action d, modifies CTS 3.9.12 Action b, CTS 3.11.1 Action b, CTS 3.11.2.1 Action c, CTS 3.11.2.2 Action b, and adds new CTS 6.8.5. This change does not affect the ITS.	Pages 69, 74, 82, 93, 97, 98, 99, 101, 104, 108, 116, 127, 131, 132, 133, 135, 141, 142, 176, 190, 196, 198, and 204 of 256.
The change described in the response to Question 200409200946, Question 200409200950, and Question 200409200954 for ITS 5.5.9 Discussion of Changes (DOC) L.3 has been made. This change revises the Frequency for ITS SR 3.7.10.1 and ITS SR 3.7.12.1 to "46 days on a STAGGERED TEST BASIS," and revises the Frequency for ITS SR 3.7.13.1 to "92 days."	Page 149 of 256.
A self-identified change for ITS 5.5.9 has been made. This change administratively corrects a typographical error.	Page 168 of 256.
The change described in the response to Question 200405271640 for ITS 5.5.14 has been made. This change revises ITS 5.5 Justification for Deviations (JFD) 17 to clarify why ITS SR 3.0.2 does not apply to the Frequencies of ITS 5.5.14.	Page 185 of 256.
A self-identified change for ITS 5.6.5.b has been made. This change revises ITS 5.6.5.b to include the document describing the analytical methods for the Overtemperature ΔT and Overpower ΔT Allowable Value parameter values as required by WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."	Pages 194, 202, 207, and 217 of 256.
A self-identified change for ITS 5.6.6 has been made. This change revises the reference to ITS LCO 3.3.3 Condition G to Condition H, because of changes made to the ITS 3.3.3 Conditions in the response to Questions 200406041043 and 200406041123 for ITS 3.3.3.	Pages 219 and 221 of 256.

VOLUME 16

CNP UNITS 1 AND 2 IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 5.0 ADMINISTRATIVE CONTROLS

Revision 1

LIST OF ATTACHMENTS

1. ITS 5.1
2. ITS 5.2
3. ITS 5.3
4. ITS 5.4
5. ITS 5.5
6. ITS 5.6
7. ITS 5.7
8. Relocated/Deleted Current Technical Specifications (CTS)

ATTACHMENT 1

ITS 5.1, Responsibility

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

ITS 5.1

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

5.1 6.1 RESPONSIBILITY

5.1.1 6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence. INSERT 1 LA.1 M.1

5.1.2 6.1.2 The Shift Manager (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Site Vice President shall be reissued to all station personnel on an annual basis. M.2 A.2

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management level through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the UFSAR and updated in accordance with 10 CFR 50.71(e).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. See ITS 5.2

c. The Senior Vice President - Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

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ITS

M.1

INSERT 1

5.1.1 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affects nuclear safety.

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A.1

ITS 5.1

ITS

6.0 ADMINISTRATIVE CONTROLS

5.1 6.1 RESPONSIBILITY

5.1.1 6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence. INSERT 1 LA.1 M.1

5.1.2 6.1.2 The Shift Manager (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Site Vice President shall be reissued to all station personnel on an annual basis. M.2 A.2

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management level through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the UFSAR and updated in accordance with 10 CFR 50.71(e).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Senior Vice President - Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

See ITS 5.2

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ITS

M.1

INSERT 1

5.1.1 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affects nuclear safety.

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DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 6.1.2 requires a management directive regarding delegation of the control room command function to be signed by the Site Vice President and issued to all station personnel on an annual basis. ITS 5.1.2 does not include this requirement. This changes the CTS by deleting the requirement to issue this management directive annually.

The purpose of CTS 6.1.2 is to specify the plant specific means of implementing the requirement to notify employees of the responsibilities of the Shift Manager. This change is acceptable because CTS 6.1.2 and ITS 5.1.2 state who is responsible for the control room command function. This requirement appears to serve only as a reminder to personnel as to who is in charge. No where else in the CTS or the ITS is a management directive required to remind personnel of a Technical Specification requirement. In addition, this requirement is not considered to be one of the more important requirements since it does not directly impact safety. The Technical Specification control room command function requirement is not being changed. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 ITS 5.1.1 requires that the plant manager or his designee approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affects nuclear safety. The CTS does not include this requirement. This changes the CTS by adding an approved requirement for the plant manager or his designee.

The purpose of the ITS 5.1.1 requirement is to provide additional assurance that the plant manager has direct responsibility for overall unit operation. This change is acceptable because having the plant manager or his designee approve actions affecting nuclear safety is consistent with the CTS 6.2.1.b (ITS 5.2.1.b) requirement that the plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. This change is designated more restrictive because it adds a requirement for the plant manager or his designee to the CTS.

M.2 CTS 6.1.2 allows a designated individual to assume the responsibility for the control room command function when the Shift Manager is absent from the control room complex. ITS 5.1.2 provides the allowance for the designated individual to assume the responsibility for the control room command function, but provides additional requirements for the designated individual. In MODE 1, 2, 3, or 4, ITS 5.1.2 requires the designated individual hold an active Senior Operator license. In MODE 5 or 6, ITS 5.1.2 requires the designated individual hold an active Senior Operator license or Operator license. This changes the CTS by adding qualification requirements for the designated individual that assumes the control room command function.

The purpose of the ITS 5.1.2 requirement is to ensure that the control room command function is maintained. This change is acceptable because the additional requirements ensure that the designated individual assuming the control room command functions meets the appropriate qualification requirements. This change is designated as more restrictive because it adds qualification requirements for the designated individual that assumes the control room command function to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.1.1 uses the title "Plant Manager" and CTS 6.1.2 uses the title "Shift Manager." ITS 5.1.1 uses the generic title "plant manager" and ITS 5.1.2 uses the generic title "shift manager." This changes the CTS by moving the specific CNP organizational titles to the UFSAR and replacing them with generic titles.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific CNP organizational titles is out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the plant manager and shift manager are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

- REVIEWER'S NOTES -

1. ~~Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~

2. ~~The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

6.1.1 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

DOC M.1 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect 5 nuclear safety. 2

manager shift manager

6.1.2 5.1.2 The Shift ~~Supervisor (SS)~~ shall be responsible for the control room command function. During any absence of the ~~SS~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior ~~Reactor~~ Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~SS~~ from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or ~~Reactor~~ Operator license shall be designated to assume the control room command function. 1 3 8 complex 1 4 3

shift manager Senior Operator

JUSTIFICATION FOR DEVIATIONS
ITS 5.1, RESPONSIBILITY

1. The brackets are removed and the proper plant specific information/value is provided.

2. Grammatical error corrected.

3. Typographical error corrected. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator."

4. The term "control room" in ISTS 5.1.2 has been changed to "control room complex" to be consistent with the current licensing basis.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.1, RESPONSIBILITY

There are no specific NSHC discussions for this Specification.

ATTACHMENT 2

ITS 5.2, Organization

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

A.1

ITS 5.2

ITS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Site Vice President shall be reissued to all station personnel on an annual basis.

See ITS 5.1

5.2 6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

5.2.1 6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility, and communication shall be established and defined for the highest management level through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the UFSAR and updated in accordance with 10 CFR 50.71(e). INSERT 1 M.1 A.2

5.2.1.b b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. LA.1

A specified corporate officer

5.2.1.c c. The Senior Vice President - Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. LA.1

5.2.1.d d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

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ITS

M.1 **INSERT 1**

5.2.1.a requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications

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A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.2 ORGANIZATION (Continued)

FACILITY STAFF

5.2.2 6.2.2 The Facility organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1. LA.2

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in Mode 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room. A.3

5.2.2.c c. An individual* qualified in radiation protection procedures shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be directly supervised by a licensed Senior Operator trained or qualified in refueling and CORE ALTERATIONS (SO-CA) who has no other concurrent responsibilities during this operation. A.3

5.2.2.d e. The amount of overtime worked by plant staff members performing safety-related functions must be limited in accordance with NRC Policy Statement on working hours (Generic Letter 82-12). LA.3

f. The Shift Manager and Unit Supervisor shall hold a Senior Operator License.

operations manager LA.1

5.2.2.e g. The ~~Operations Director~~ must hold or have held a Senior Operator License at Cook Nuclear Plant or a similar reactor, or have been certified for equivalent senior operator knowledge. If the ~~Operations Director~~ does not hold a Senior Operator License, then a line ~~(v. staff)~~ operations middle manager shall hold a Senior Operator License for the purposes of directing operational activities. A.1

operations manager LA.1

5.2.2.c * The unexpected absence, for a period of time not to exceed 2 hours, of the on-site individual qualified in radiation protection procedures is permitted provided immediate action is taken to fill the required position.

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ITS 5.2

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION*

	LICENSE CATEGORY	APPLICABLE MODES 1,2,3 & 4	APPLICABLE MODES 5 & 6
	~~SM~~	~~1**~~	1**#
	~~SOL~~	~~1~~	None
	~~OL~~	~~2~~	1
5.2.2.a	Non-Licensed	2	1
5.2.2.f	~~Shift Technical Adv.~~ INSERT 2	1**	None

LA.2

M.2

~~Does~~ not include the licensed ~~Senior~~ Operator - CA ~~supervising~~ CORE ALTERATIONS. LA.2

5.2.2.b * Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

5.2.2.f ** Shared with Cook Nuclear Plant Unit 2.

ITS

M.2 INSERT 2

5.2.2.f An individual shall provide advisory technical support to unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit.

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ITS 5.2

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Director, who must be qualified as specified in Section 6.2.2.g.

5.2.2.f

See ITS 5.3

A.4

See ITS 5.3

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

See CTS 6.0

6.5 DELETED

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A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Site Vice President shall be reissued to all station personnel on an annual basis.

See ITS 5.1

5.2 6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

5.2.1 6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a a. Lines of authority, responsibility, and communication shall be established and defined for the highest management level through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These ~~organizational charts~~ will be documented in the UFSAR ~~and updated in accordance with 10 CFR 50.71(e).~~

INSERT 1

M.1

A.2

5.2.1.b b. The ~~Plant Manager~~ shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

LA.1

A specified corporate officer

5.2.1.c c. ~~The Senior Vice President - Nuclear Operations~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

LA.1

5.2.1.d d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

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ITS

M.1 **INSERT 1**

5.2.1.a requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications

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A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.2 ORGANIZATION (Continued)

FACILITY STAFF

5.2.2 6.2.2 The Facility organization shall be subject to the following:

a. ~~Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.~~ LA.2

b. ~~At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in Mode 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room.~~ A.3

5.2.2.c c. An individual* qualified in radiation protection procedures shall be on site when fuel is in the reactor.

d. ~~All CORE ALTERATIONS shall be directly supervised by a licensed Senior Operator trained or qualified in refueling and CORE ALTERATIONS (SO-CA) who has no other concurrent responsibilities during this operation.~~ A.3

5.2.2.d e. The amount of overtime worked by plant staff members performing safety-related functions must be limited in accordance with NRC Policy Statement on working hours (Generic Letter 82-12).

f. ~~The Shift Manager and Unit Supervisor shall hold a Senior Operator License.~~ LA.3

5.2.2.e g. The ~~Operations Director~~ operations manager must hold or have held a Senior Operator License at Cook Nuclear Plant or a similar reactor, or have been certified for equivalent senior operator knowledge. If the ~~Operations Director~~ operations manager does not hold a Senior Operator License, then a line ~~(v. staff)~~ operations middle manager shall hold a Senior Operator License for the purposes of directing operational activities. LA.1 A.1 LA.1

5.2.2.c * The unexpected absence, for a period of time not to exceed 2 hours, of the on-site individual qualified in radiation protection procedures is permitted provided immediate action is taken to fill the required position.

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A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION*

ITS	LICENSE CATEGORY	APPLICABLE MODES 1, 2, 3 & 4	APPLICABLE MODES 5 & 6
	SM	1**	1***
	SOL	1	None
	OL	2	1
5.2.2.a	Non-Licensed	2	1
5.2.2.f	~~Shift Technical Adv.~~	1**	None

INSERT 2

LA.2

M.2

~~# Does not include the licensed Senior Operator - CA supervising CORE ALTERATIONS.~~

LA.2

5.2.2.b * Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

5.2.2.f ** Shared with Cook Nuclear Plant Unit 1

ITS

M.2 **INSERT 2**

5.2.2.f An individual shall provide advisory technical support to unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit.

ITS 5.2

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.3 FACILITY STAFF QUALIFICATIONS

See ITS 5.3

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation ~~Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2)~~ the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with ~~specific training in plant design, and response and analysis of the plant for transients and accidents and,~~ (3) the Operations Director, who must be qualified as specified in Section 6.2.2.g.

5.2.2.f

A.4

See ITS 5.3

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

See CTS 6.0

6.5 DELETED

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DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 6.2.1.a states, in part, "These organizational charts will be documented in the UFSAR and updated in accordance with 10 CFR 50.71(e)." The ITS does not include the requirement associated with updating the UFSAR in accordance with 10 CFR 50.71(e). This changes the CTS by deleting these requirements for updating the UFSAR.

10 CFR 50.71(e) provides requirements for periodically updating the UFSAR. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.71(e). This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 6.2.2.b states "At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in Mode 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room." CTS 6.2.2.d requires all CORE ALTERATIONS to be directly supervised by a licensed Senior Operator trained or qualified in refueling and CORE ALTERATIONS who has no other concurrent responsibilities during this operation. The ITS does not include these requirements. This changes the CTS by deleting these requirements.

10 CFR 50.54(m)(2)(iii) states "When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by a unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be at the controls at all times." 10 CFR 50.54(m)(2)(iv) states "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person." This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.54(m)(2)(iii) and 10 CFR 50.54(m)(2)(iv). This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 6.3.1 provides, in part, qualification requirements for the Shift Technical Advisor (STA), and requires the STA to have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. ITS 5.2.2.f requires this individual to meet the qualification requirements of the Commission

Policy Statement on Engineering Expertise on Shift. This changes the CTS by referencing the Commission Policy Statement on Engineering Expertise on Shift for qualification requirements instead of listing the specific qualification requirements.

The purpose of the CTS 6.3.1 STA requirements is to specify the minimum qualification requirements for the STA. This change is acceptable because the qualification requirements included in the Commission Policy Statement on Engineering Expertise on Shift encompass the current STA qualification requirements. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 6.2.1.a, regarding documentation and updating of the relationships between operating organization positions, requires the organizational charts to be documented in the UFSAR. ITS 5.2.1.a states "These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the UFSAR." This changes the CTS by requiring that the specific CNP organizational titles be specified in the UFSAR.

This change is acceptable because specifying the relationship of the specific CNP organizational titles to the generic titles used in the Technical Specifications and industry standards in the UFSAR continues to ensure that organizational positions and associated responsibilities will be maintained. This change adds this requirement to the Technical Specifications. This change is designated as more restrictive because it requires additional information be maintained in the UFSAR.

M.2 CTS Table 6.2-1 requires the minimum shift crew to include one STA (shared between Units 1 and 2) when the unit is in MODE 1, 2, 3, or 4. ITS 5.2.2.f requires, in part, that an individual (shared between Units 1 and 2) provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit, when the unit is in MODE 1, 2, 3, or 4. This changes the CTS by detailing the specific responsibilities of the STA.

The purpose of the CTS Table 6.2-1 STA requirements is to ensure that appropriate engineering expertise is available on shift. This change is acceptable because it clarifies STA requirements consistent with Commission Policy Statement on Engineering Expertise on Shift. This change is designated as more restrictive because it provides specific details of the responsibilities of the STA.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.2.1.b uses the title "Plant Manager," CTS 6.2.1.c uses the title "Senior Vice President - Nuclear Operations," and CTS 6.2.2.g uses the title "Operations Director." ITS 5.2.1.b uses the generic title "plant manager," ITS 5.2.1.c uses the generic title "A specified corporate officer," and ITS 5.2.2.e uses the generic title "operations manager." This changes the CTS by moving the specific CNP organizational titles to the UFSAR and replacing them with generic titles.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific CNP organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the plant manager, the specified corporate officer, and the operations manager are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.2.2 and Table 6.2-1, including footnote #, provide minimum shift crew composition requirements. ITS 5.2.2 only includes the minimum shift crew composition requirements that are not already included in 10 CFR 50.54. This changes the CTS by moving the minimum shift crew composition requirements addressed by 10 CFR 50.54 to the Technical Requirements Manual (TRM).

The removal of these details, which are related to meeting Technical Specification 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 minimum shift crew composition requirements for licensed operators and senior operators are also contained in 10 CFR 50.54(k), (l), and (m) and do not need to be repeated in the Technical Specifications. The minimum shift crew composition requirements for non-licensed operators are transferred from CTS Table 6.2-1 to ITS 5.2.2.a and the minimum shift crew composition requirements for the STA are transferred from CTS Table 6.2-1 to ITS 5.2.2.f. The relocation of the details of the minimum shift crew composition requirements to the TRM is acceptable considering the controls provided by regulations and the remaining requirements in the Technical Specifications. Also, this change is acceptable because these details 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

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LA.3 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.2.2.f requires the Shift Manager and Unit Supervisor to hold a Senior Operator license. ITS 5.2.2 does not contain this requirement. This changes the CTS by moving the requirement for the Shift Manager and Unit Supervisor to hold a Senior Operator license to the TRM.

The removal of these details, which are related to meeting Technical Specification 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 requirement for shift supervision to hold Senior Operator licenses is contained in 10 CFR 50.54(m), and does not need to be repeated in the Technical Specifications. The relocation of the details of the shift supervision personnel that are required to hold Senior Operator licenses to the TRM is acceptable considering the controls provided by regulations. Also, this change is acceptable because these details 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 details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

6.2.1.a
a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, ~~functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation~~. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the FSAR ~~QA Plan~~. (1) (3) (3) (2)

6.2.1.b
b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. (3)

6.2.1.c
c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety ~~and~~. (3)

6.2.1.d
d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

Table 6.2-1
a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. (3) (9)

~~- REVIEWER'S NOTE -~~
~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~ (4)

CTS

5.2 Organization

5.2.2 Unit Staff (continued)

Specifications (8)

Table 6.2-1 Note * — b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements; (9)

6.2.2.c and Note * — c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position; (9)

6.2.2.e — d. ~~Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., [licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physicists, auxiliary operators, and key maintenance personnel]).~~ (5)

~~The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.~~ INSERT 1

~~Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.~~

~~*Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not be assigned.*~~ (6)

6.2.2.g — e. The operations manager ~~or assistant operations~~ manager shall hold an SRO license; and INSERT 2 (9)

Table 6.2-1 and Note **, 6.3.1 — f. An ~~individual~~ shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. INSERT 3 (7)

5 **INSERT 1**

The amount of overtime worked by unit staff members performing safety related functions must be limited in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12);

6 **INSERT 2**

must hold or have held a Senior Operator license at Cook Nuclear Plant or a similar reactor, or have been certified for equivalent Senior Operator knowledge. If the operations manager does not hold an Senior Operator license, then a line operations middle manager shall hold a Senior Operator license for the purposes of directing operational activities.

7 **INSERT 3**

In MODE 1, 2, 3, or 4, an individual (shared with Unit 2 (Unit 1) and Unit 1 (Unit 2))

JUSTIFICATION FOR DEVIATIONS
ITS 5.2, ORGANIZATION

1. ISTS 5.2.1.a is revised to reflect the CNP CTS with respect to documentation and updating of the relationships between operating organization positions. Specifically, the ISTS 5.2.1.a requirement for including these relationships in functional descriptions of departmental responsibilities and relationships, and job descriptions of key personnel positions, or in equivalent forms of documentation is not included in ITS 5.2.1.a. This change is made to achieve consistency with CTS 6.2.1.a, which was approved by the NRC in License Amendments 132 (Unit 1) and 117 (Unit 2), dated March 9, 1990.

2. The brackets are removed and the proper plant specific information/value is provided.

3. Grammatical/typographical error corrected.

4. The ISTS Reviewer's Note has been deleted since it is not intended to be included in the ITS. The requirements for non-licensed operators for two unit sites addressed in the ISTS Reviewer's Note are not adopted. This change is consistent with the CNP CTS.

5. ISTS 5.2.2.d provides requirements for working hour limitations. These requirements are revised in ITS 5.2.2.d to reflect the CNP CTS 6.2.2.e requirements, which were approved by the NRC in License Amendments 77 (Unit 1) and 58 (Unit 2), dated November 23, 1983.

6. ISTS 5.2.2.e provides a requirement for the operations manager or the assistant operations manager to hold a Senior Operator license. This requirement is revised in ITS 5.2.2.e to reflect the CNP CTS 6.2.2.g requirements. The CTS 6.2.2.g requirements were approved by the NRC in License Amendments 212 (Unit 1) and 197 (Unit 2), dated November 13, 1996.

7. ISTS 5.2.2.f provides requirements for the Shift Technical Advisor (STA). These requirements are revised in ITS 5.2.2.f to reflect the CNP CTS Table 6.2-1 requirements for the STA. The CTS Table 6.2-1 STA requirements were approved by the NRC in License Amendments 49 (Unit 1) and 34 (Unit 2), dated August 25, 1981.

8. The referenced requirements are Specifications, not Code of Federal Regulations (CFR) requirements. Therefore, the word "Specifications" has been added to clearly state that 5.5.2.a and 5.5.2.f are Specifications.

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

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.2, ORGANIZATION

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 3

ITS 5.3, Unit Staff Qualifications

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

A.1

ITS 5.3

ITS

6.0 ADMINISTRATIVE CONTROLS

5.3 6.3 FACILITY STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the ~~Plant~~ Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations ~~Director~~ manager who must be qualified as specified in Section 6.2.2.g.

LA.1

See ITS 5.2

LA.1

Add proposed Specification 5.3.2

A.2

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

See CTS 6.0

6.5 DELETED

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A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

5.3 6.3 FACILITY STAFF QUALIFICATIONS

LA.1

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Director, who must be qualified as specified in Section 6.2.2.g.

manager

See ITS 5.2

LA.1

Add proposed Specification 5.3.2

A.2

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

See CTS 6.0

6.5 DELETED

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DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 ITS 5.3.2 states "For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed Operator are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m)." The CTS does not include such a statement. This changes the CTS by clarifying that these individuals must meet all of the qualification requirements referenced in 10 CFR 55.4, ITS 5.3.1, and 10 CFR 50.54(m).

This change is acceptable because it clarifies the existing relationship between the Technical Specifications and regulations regarding licensed Senior Operator and Operator qualification requirements. 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

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.3.1 uses the titles "Plant Radiation Protection Manager" and "Operations Director." ITS 5.3.1 uses the generic titles "radiation protection manager" and "operations manager." This changes the CTS by moving the specific CNP organizational titles to the UFSAR and replacing them with generic titles.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific CNP organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the radiation protection manager and the operations

DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

manager are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.3 5.3 Unit Staff Qualifications

~~- REVIEWER'S NOTE -~~
~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~ (1)

6.3.1 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff]~~. INSERT 1 (2)

Doc A.2 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of ~~TS~~ Specification 5.3.1, perform the functions described in 10 CFR 50.54(m). (3) (4)

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2

INSERT 1

ANSI N18.1-1971 for comparable positions, except for the radiation protection manager and the operations manager. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The operations manager shall be qualified as required by Specification 5.2.2.e.

JUSTIFICATION FOR DEVIATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

1. The ISTS Reviewer's Note has been deleted since it is not intended to be included in the ITS.

2. The brackets are removed and the proper plant specific information/value is provided.

3. Grammatical/typographical error corrected. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator."

4. Change made for consistency with the terminology used in other Specifications.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 4

ITS 5.4, Procedures

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

A.1

ITS 5.4

ITS

6.0 ADMINISTRATIVE CONTROLS

5.4 6.8 PROCEDURES AND PROGRAMS — See ITS 5.5

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Rev. 2, February 1978.

Add proposed Specification 5.4.1.b — M.1

b. Deleted.

c. Deleted.

d. ~~PROCESS CONTROL PROGRAM implementation.~~ — LA.1

5.4.1.e e. OFFSITE DOSE CALCULATION MANUAL ~~implementation~~. — A.2

5.4.1.c f. Quality Assurance Program for effluent and environmental monitoring ~~using the guidance in Regulatory Guide 1.21, Rev. 1, June 1974, and Regulatory Guide 4.1, Rev. 1, April 1975.~~ — LA.2

5.4.1.e g. ~~Component Cyclic or Transient Limits program, which provides controls to track the UFSAR, Section 4.1, cyclic and transient occurrences to ensure that components are maintained within the limits.~~ — A.2

5.4.1.d h. Fire Protection Program implementation.

~~6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes, shall be reviewed prior to implementation as set forth in Quality Assurance Program Description, Appendix C, Section 6.5.~~ — LA.3

6.8.3 Deleted.

Add proposed Specification 5.4.1.e — M.2

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A.1

ITS 5.4

ITS

6.0 ADMINISTRATIVE CONTROLS

5.4 6.8 PROCEDURES AND PROGRAMS — See ITS 5.5

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Rev. 2, February 1978.

Add proposed Specification 5.4.1.b — M.1

b. Deleted.

c. Deleted.

d. PROCESS CONTROL PROGRAM implementation. — LA.1

5.4.1.e e. OFFSITE DOSE CALCULATION MANUAL implementation. — A.2

5.4.1.c f. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Rev. 1, June 1974, and Regulatory Guide 4.1, Rev. 1, April 1975. — LA.2

5.4.1.e g. Component Cyclic or Transient Limits program, which provides controls to track the UFSAR, Section 4.1, cyclic and transient occurrences to ensure that components are maintained within the limits. — A.2

5.4.1.d h. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes, shall be reviewed prior to implementation as set forth in Qualification Assurance Program Description, Appendix C, Section 6.5. — LA.3

6.8.3 Deleted.

Add proposed Specification 5.4.1.e — M.2

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 6.8.1.e requires procedures for implementation of the OFFSITE DOSE CALCULATION MANUAL (ODCM) and CTS 6.8.1.g requires procedures for the implementation of the Component Cyclic or Transient Limits Program. ITS 5.4.1 requires procedures for various activities, but does not specifically list the ODCM and the Component Cyclic or Transient Limits Program. This changes the CTS by removing the explicit requirements for written procedures for implementation of the ODCM and the Component Cyclic or Transient Limits Program.

This change is acceptable because implementing procedures for the ODCM and the Component Cyclic or Transient Limits Program are required by ITS 5.4.1.e. ITS 5.4.1.e (added as described in DOC M.2) requires that written procedures be established, implemented, and maintained for all programs and manuals in ITS 5.5 (including the ODCM and Component Cyclic or Transient Limits Program). Therefore, it is not necessary to specifically identify each program in ITS 5.4.1. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 ITS 5.4.1.b requires that written procedures shall be established, implemented, and maintained for the emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. The CTS does not include this requirement. This changes the CTS by adopting a new requirement for emergency operating procedures.

The purpose of ITS 5.4.1.b is to ensure that written procedures are established, implemented, and maintained covering the emergency operating procedures to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. This change is acceptable because it is consistent with an existing requirement to comply with NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33, for emergency operating procedures. This change is designated more restrictive because it imposes a new requirement for procedures within the Technical Specifications.

M.2 ITS 5.4.1.e requires that written procedures shall be established, implemented, and maintained for all programs specified in Specification 5.5. The CTS does not include this requirement for any program except the ODCM and the Component

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

Cyclic or Transient Limits Program. This changes the CTS by adopting a new requirement for procedures to address all programs described in ITS 5.5.

The purpose of ITS 5.4.1.e is to ensure that written procedures are established, implemented, and maintained covering all programs specified in ITS 5.5. This change is considered acceptable because it requires written procedures, including proper procedure control to address programs required by ITS 5.5. This change is designated more restrictive because it imposes new requirements for procedures within the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.8.1.d requires that written procedures for the PROCESS CONTROL PROGRAM (PCP) be established, implemented, and maintained. The ITS does not include these requirements. This changes the CTS by moving the requirements to the UFSAR.

The removal of these details, which are related to meeting Technical Specification 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 PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Compliance with these regulations is required by the CNP Units 1 and 2 Operating Licenses, and written procedures are necessary to ensure compliance with the program. Regulations provide an adequate level of control for the affected requirements, and inclusion of this requirement in the Technical Specifications is not necessary. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.8.1.f requires written procedures be established, implemented and maintained covering the Quality Assurance Program for effluent and environmental monitoring, "using the guidance in Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975." ITS 5.4.1.c does not include the Regulatory Guide references. This changes the CTS by moving the references to the Regulatory Guides to the Quality Assurance Program Description (QAPD).

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable

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DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for written procedures covering quality assurance for effluent and environmental monitoring. Also, this change is acceptable because these types of procedural details will be adequately controlled in the QAPD. Any changes to the QAPD are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.3 *(Type 3 – Removing Procedural Details for Meeting TS or Reporting Requirements)* CTS 6.8.2 requires that each procedure and administrative policy of Specification 6.8.1, and changes to these documents, including temporary changes, be reviewed prior to implementation in accordance with the QAPD. ITS 5.4 does not include this requirement. This changes the CTS by moving these details of procedure and administrative policy reviews to the QAPD.

The removal of these details, which are related to meeting Technical Specification 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. ITS 5.4.1 still retains the requirement for written procedures required by the Technical Specifications to be established, implemented, and maintained. Regulations provide an adequate level of control for the affected review requirement. The requirements for establishment, maintenance, and implementation of procedures related to activities affecting quality are contained in 10 CFR 50, Appendix B, Criterion II and Criterion V and ANSI N18.7-1976 (ANS 3.2-1976). In accordance with these requirements, the QAPD includes adequate detail with respect to administrative control of procedures related to activities affecting quality and nuclear safety, including the review requirements associated with maintenance of these procedures. Also, this change is acceptable because these types of procedural details will be adequately controlled in the QAPD. Any changes to the QAPD are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.8 5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; 2

DOC M.1 b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33; 3 1 2

6.8.1.f c. Quality assurance for effluent and environmental monitoring; 2

6.8.1.h d. Fire Protection Program implementation; and 2

DOC M.2, 6.8.1.d, 6.8.1.g e. All programs specified in Specification 5.5.

JUSTIFICATION FOR DEVIATIONS
ITS 5.4, PROCEDURES

1. The brackets are removed and the proper plant specific information/value is provided.

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

3. Grammatical errors corrected.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.4, PROCEDURES

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 5

ITS 5.5, Programs and Manuals

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

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5 6.8.4 The following programs shall be established, implemented, and maintained:

5.5.3 a. Radioactive Effluent Controls Program

5.5.3 A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

5.5.3.a 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR 20.1001-20.2402, Appendix B, Table 2, Column 2,

5.5.3.c 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

5.5.3.f 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

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ITS 5.5

ITS

6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY shall be limited to the following:

a) For noble gases: Less than or equal to a dose rate of 500 mrem/year to the total body and less than or equal to a dose rate of 3000 mrem/year to the skin, and

b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

5.5.3.h 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

5.5.3.j 10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and ~~to radiation from uranium~~ fuel cycle sources conforming to 40 CFR Part 190.

A.2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Control Program Surveillance Frequencies.

b. ~~Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~

LA.1

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

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6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5.12 6.8.5 Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

5.5.12.a a. Changes to the Bases of the Technical Specification shall be made under appropriate administrative controls and reviews.

5.5.12.b b. Licensees may make changes to Bases without prior Nuclear Regulatory Commission approval provided the changes do not require either of the following:

1. A changes in the Technical Specification incorporated in the license or

2. A change to the Updated Final Safety Analysis Report or Bases that requires Nuclear Regulatory Commission approval pursuant to 10 CFR 50.59.

5.5.12.c c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the Updated Final Safety Analysis Report.

5.5.12.d d. Proposed changes that meet the criteria of Specification 6.8.5.b above shall be reviewed and approved by the Nuclear Regulatory Commission prior to implementation. Changes to the Bases implemented without prior Nuclear Regulatory Commission approval shall be provided to the Nuclear Regulatory Commission on a frequency consistent with 10 CFR 50.71(e).

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator unless otherwise noted.

STARTUP REPORT

See ITS 5.6

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 28 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Less than Four Loop Operation

The licensee shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this license) with less than four reactor coolant loops in operation until (a) safety analyses for less than four loop operation have been submitted, and (b) approval for less than four loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

(4) Indiana Michigan Power Company shall implement and maintain, in effect, all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility and as approved in the SERs dated December 12, 1977, July 31, 1979, January 30, 1981, February 7, 1983, November 22, 1983, December 23, 1983, March 16, 1984, August 27, 1985, June 30, 1986, January 28, 1987, May 26, 1987, June 16, 1988, June 17, 1988, June 7, 1989, February 1, 1990, February 9, 1990, March 26, 1990, April 26, 1990, March 31, 1993, April 8, 1993, December 14, 1994, January 24, 1995, April 19, 1995, June 8, 1995, and March 11, 1996, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(5) Deleted by Amendment No. 279

(6) Deleted by Amendment No. 80

5.5.8 (7) Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program ~~shall be described in the station chemistry manual and~~ shall include: L.6

5.5.8.a 1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

5.5.8.b 2. Identification of the procedures used to measure the values of the critical parameters;

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A.1

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-4-

5.5.8.c 3. Identification of process sampling points;

5.5.8.d 4. Procedure for the recording and management of data;

5.5.8.e 5. Procedures defining corrective actions for off control point chemistry conditions; and

5.5.8.f 6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

(8) Deleted by Amendment No. 279

(9) Deleted by Amendment No. 279

(10) Deleted by Amendment No. 279

(11) Deleted by Amendment No. 279

D. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Donald C. Cook Nuclear Plant Security Plan," with revisions submitted through July 21, 1988; "Donald C. Cook Nuclear Plant Training and Qualification Plan," with revisions submitted through December 19, 1986; and "Donald C. Cook Nuclear Plant Safeguards Contingency Plan," with revisions submitted through June 10, 1988. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

M.1

E. Deleted by Amendment No. 80

F. Deleted by Amendment No. 80

Add proposed Systems list

G. In all places of this license, the reference to the Indiana and Michigan Electric Company is amended to read Indiana Michigan Power Company.

5.5.2 H. System Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low a practical levels. The program shall include the following:

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ITS 5.5

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5.5.2 1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency ~~not to exceed refueling cycle intervals~~. 24 months L.1

The provisions of SR 3.0.2 are applicable.

~~I. Iodine Monitoring~~

~~The licensee shall implement a program which will ensure the capability to accurately determine the airborne concentration in vital areas under accident conditions. This program shall include the following:~~

~~1. Training of Personnel,~~
~~2. Procedures for monitoring, and~~
~~3. Provisions for maintenance of sampling and analysis equipment.~~

LA.2

J. The licensee is authorized to use digital signal processing instrumentation in the reactor protection system.

3. This amended license is effective as of the date of issuance and shall expire at midnight October 25, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A – Technical Specifications

Date of Issuance: March 30, 1976

Amendment No. 279

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DEFINITIONS

PROCESS CONTROL PROGRAM (PCP)

1.28 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste. — See CTS 6.0

1.29 Deleted.

5.5.1 OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a 1.30 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. 5.5.1.b The ODCM shall contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment. — See ITS Chapter 1.0

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.0 APPLICABILITY

This provision shall not prevent entry into OPERATIONAL 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. See ITS Section 3.0

5.5.6 4.0.5 Surveillance Requirements for inservice ~~inspection/and~~ testing of ASME Code Class 1, 2, and 3 ~~components~~ pumps and valves shall be applicable as follows: LA.3

a. ~~Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a.~~ LA.4

SURVEILLANCE REQUIREMENTS A.15

5.5.6.a b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice ~~inspection/and~~ testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: LA.3

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice ~~inspection/and~~ testing criteria	Required frequencies for performing inservice ~~inspection/and~~ testing activities
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
Quarterly or every 3 months	At least once per 92 days
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

A.15 A.3

5.5.6.b c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice ~~inspection/and~~ testing activities. LA.3

~~d. Performance of the above inservice inspection/and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.4

5.5.6.d e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification. A.15

Add proposed ITS 5.5.6.c A.5

4.0.6 Deleted

4.0.7 Deleted

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ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

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

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

See ITS 3.4.13

5.5.7 4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

Add proposed ITS 5.5.7 generic program description

A.6

5.5.7 4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each

5.5.7.a inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

5.5.7.a.1 a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

5.5.7.a.2 b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

5.5.7.a.2.a) 1. All tubes that previously had detectable wall penetrations (greater than or equal to 20%) that have not been plugged.

* This Specification does not apply in Mode 4 while performing crevice flushing as long as Limiting Conditions for Operation for Specification 3.4.1.3 are maintained.

See ITS 3.4.13

A.1

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

5.5.7.a.2.b) 2. Tubes in those areas where experience has indicated potential problems.

5.5.7.a.2.c) 3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5.5.7.a.3 c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

5.5.7.a.3.a) 1. The tubes selected for the samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

5.5.7.a.3.b) 2. The inspections include those portions of the tubes where imperfections were previously found.

5.5.7.b The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

5.5.7.b Note: In all inspections, previously degraded tubes must exhibit significant (greater than or equal to 10%) further wall penetrations to be included in the above percentage calculations.

5.5.7.c 4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

5.5.7.c.1 a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality or replacement of steam generators. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

5.5.7.c.2 b. If the results of inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

5.5.7.c.3 c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

5.5.7.c.3.a) 1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

5.5.7.c.3.b) 2. A seismic occurrence greater than the Operating Basis Earthquake.

5.5.7 c.3.c) 3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

5.5.7.c.3.d) 4. A main steam line or feedwater line break.

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ITS 5.5

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

5.5.7.d 4.4.5.4 Acceptance Criteria

5.5.7.d.1 a. As used in this Specification:

5.5.7.d.1.a) 1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

5.5.7.d.1.b) 2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

5.5.7.d.1.c) 3. Degraded Tube means an imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation.

5.5.7 d.1.d) 4. Percent Degradation means the amount of the tube wall thickness affected or removed by degradation.

5.5.7.d.1.e) 5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

5.5.7.d.1.f) 6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service. Any tube which, upon inspection, exhibits tube wall degradation of 40 percent or more of the nominal tube wall thickness shall be plugged prior to returning the steam generator to service.

5.5.7.d.1.g) 7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

5.5.7.d.1.h) 8. Inspection determines the condition of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

Add proposed ITS 5.5.7.d.1.i) A.14

ITS 5.5

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

5.5.7.d.2 b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Program test Frequencies. A.6

A.1

ITS

REACTOR COOLANT SYSTEMS

Table 5.5.7-1

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

Table Notation:

Table 5.5.7-1 Footnote (a)

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

Table 5.5.7-1 Footnote (a)

2. The third and fourth steam generators not inspected during the first inservice inspection shall be inspected during the second and third inspections, respectively. The fourth and subsequent inspections shall follow the instructions described in 1 above.

Table 5.5.7-2

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes, and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specification 6.9.1	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

S=3(N/n)% N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

A.1

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

See CTS 3/4.4.10.1

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of the ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

5.5.5 4.4.10.1 ~~In addition to the requirements of Specification 4.0.5,~~ each reactor coolant pump flywheel shall be inspected by either qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels once every 10 years.

Add proposed ITS 5.5.5 generic program statement

A.13

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency.

A.1

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ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.6 CONTAINMENT SYSTEMS

See ITS 3.6.1

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.14 and 5.5.14.a

3.6.1.2 Containment leakage rates shall be limited to:

A.7

5.5.14.b, 5.5.14.c, 5.5.14.d.1 a. ~~An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 12.0 psig, and~~

5.5.14.d.1 b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Types B and C tests when pressurized to P_a.

L.2

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

See ITS 3.6.1

5.5.14.d.1 With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, restore the overall integrated leakage rate to $\leq 0.75\ L_a$ and the combined leakage rate for all penetrations and valves subject to Types B and C tests to $\leq 0.60\ L_a$ ~~prior to increasing the Reactor Coolant System temperature above 200°F.~~

L.2

SURVEILLANCE REQUIREMENTS

5.5.14.a 4.6.1.2 Perform leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.163, dated September 1995. See Notes 1 and 2.

a. Each containment air lock shall be verified to be in compliance with the requirements of Specification 3.6.1.3.

See ITS 3.6.1

~~b. The provisions of Specification 4.0.2 are not applicable.~~

A.7

Add proposed ITS 5.5.14.e

A.7

Notes:

5.5.14.a.2 1 A one-time exception to the requirement to perform post-modification Type A testing is allowed for the steam generators and associated piping, as components of the containment barrier. For this case, ASME Section XI leak testing will be used to verify the leak tightness of the repaired or modified portions of the containment barrier. Entry into MODES 3 and 4 following the extended outage that commenced in 1997 may be made to perform this testing.

5.5.14.a.1 2 The Type A testing frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "...at least once per 10 years based on acceptable performance history" is modified to be "...at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in October 1992.

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A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.6 CONTAINMENT SYSTEMS

See ITS 3.6.2

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

5.5.14.d.2.a), 5.5.14.b b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a, 12 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.6.2

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

5.5.14.a a. In accordance with 10 CFR 50 Appendix J Option B and Regulatory Guide 1.163, dated September 1995, and

b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

See ITS 3.6.2

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ITS 5.5

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

Add proposed ITS 5.5.9 generic program statement — A.9

4.7.5.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. Deleted

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

See ITS 3.7.10

L.3 A.8

5.5.9 c. At least once per ~~18~~ 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system ~~, by~~:

while it is in operation that could adversely affect the filter bank or charcoal adsorber capability

5.5.9.b 1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

5.5.9.a 2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

LA.5

5.5.9.c 3. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 1.0% radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

5.5.9.a, 5.5.9.b 4. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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ITS 5.5

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9 d. After every 720 hours of charcoal adsorber operation by either: LA.5

5.5.9.c 1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H; or LA.5

5.5.9.c 2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H; and the samples are prepared by either:

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:~~

~~a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%, and~~

~~b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.~~

L.4

A.1

ITS 5.5

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

L.3

24

5.5.9 e. At least once per 18 months by:

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

See ITS 3.3.7 and ITS 3.7.10

3. Verifying that the system maintains the control room envelope/pressure boundary at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10%, with a makeup air flow rate of ≤ 1000 cfm.

See ITS 3.7.10

5.5.9
5.5.9.a f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

5.5.9
5.5.9.b g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

A.9

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

A.1

ITS 5.5

ITS

PLANT SYSTEMS

3/4.7.6 ESF VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent ESF ventilation system exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ESF ventilation system exhaust air filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.7.12

SURVEILLANCE REQUIREMENTS

Add proposed ITS 5.5.9 generic program statement A.9

4.7.6.1 Each ESF ventilation system exhaust air filter train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

See ITS 3.7.12

L.3

A.8

5.5.9 b. At least once per ~~18~~ 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, while it is in operation that could adversely affect the filter bank or charcoal adsorber capability by:

1. Deleted.

5.5.9.b 2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm ± 10%.

5.5.9.a 3. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm ± 10%.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

LA.5

5.5.9.c 4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.~~ L.4

5.5.9.a, 5.5.9.b 5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.9 c. After every 720 hours of charcoal adsorber operation by either:

5.5.9.c 1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity; or LA.5

5.5.9.c 2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity and the samples are prepared by either: LA.5

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

ITS 5.5

A.1

ITS

PLANT SYSTEM
SURVEILLANCE REQUIREMENTS (Continued)

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%:~~ L.4

5.5.9 d. At least once per ~~18~~ 24 months by: L.3

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that the standby fan starts automatically on a Containment Pressure--High-High Signal and directs its exhaust flow through the HEPA filters and charcoal adsorber banks on a Containment Pressure--High-High Signal. See ITS 3.7.12

5.5.9 e. ~~After each complete or partial replacement of~~ HEPA filter band by
5.5.9.a ~~verifying that the HEPA filter banks remove~~ greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5.5.9 f. ~~After each complete or partial replacement of a~~ charcoal adsorber bank
5.5.9.b ~~by verifying that the charcoal adsorbers remove~~ greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

A.9

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

A.1

ITS 5.5

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.8 ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE: — See ITS 3.8.1 and ITS 3.8.3

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank, — See ITS 3.8.1

2. Verifying the fuel level in the fuel storage tank, — See ITS 3.8.3

3. Verifying that the fuel transfer pump can be started and that it transfers fuel from the storage system to the day tank,

4. Verifying that the diesel starts from standby conditions and achieves in less than or equal to 10 seconds, voltage = 4160±420 V, and frequency = 60±1.2 Hz,* — See ITS 3.8.1

5. Verifying the diesel is synchronized and loaded and operates for greater than or equal to 60 minutes at a load of 3500 kw**, and

6. Verifying that the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By removing accumulated water*** — See ITS 3.8.1 and ITS 3.8.3

1) From the day tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and — See ITS 3.8.1

2) From the storage tanks at least once per 31 days. — See ITS 3.8.3

5.5.11.a c. By sampling new fuel oil*** in accordance with the applicable guidelines of ASTM D4057/81 prior to adding new fuel to the storage tanks and — LA.6

5.5.11.a 1) By verifying, in accordance with the tests specified in ASTM D975-81 and prior to adding the new fuel to the storage tanks, that the sample has:

Add proposed ITS 5.5.11 generic program statement — A.10

* The diesel generator start (10 seconds) from standby conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing and compensatory action may be at reduced acceleration rates as recommended by the manufacturer so that mechanical stress and wear on the diesel engine are minimized. — See ITS 3.8.1

** Momentary load transients do not invalidate this test.

*** The actions to be taken should any of the properties be found outside of specified limits are defined in the Bases. — See ITS 3.8.3

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ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.8 ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

within limits

LA.6

5.5.11.a.2 a) A kinematic viscosity ~~of greater than or equal to 1.9 centistokes but less than or equal to 4.1 centistokes at 40°C (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6 but less than or equal to 40.1)~~ if gravity was not determined by comparison with supplier's certification.

5.5.11.a.2 b) A flash point ~~equal to or greater than 125°F~~ within limits

within limits

5.5.11.a.1 2) By verifying ~~in accordance with the test specified in ASTM D1298-80 and~~ prior to adding the new fuel to the storage tanks, that the sample has either an API gravity ~~of greater than or equal to 30 degrees but less than of equal to 40 degrees at 60°F~~ or an absolute specific gravity ~~at 60/60°F of greater than or equal to 0.82 but less than or equal to 0.88~~ or an API gravity ~~of within 0.3 degrees at 60°F when compared to the supplier's certificate~~ or a specific gravity ~~of within 0.0016 at 60/60° when compared to the supplier's certificate~~.

LA.6

of new fuel oil, other than those addressed in Specification 5.5.11.a above.

5.5.11.a.3 3) By verifying ~~in accordance with the test specified in ASTM D4176-82 and~~ prior to adding new fuel to the storage tanks, that the sample has a clear and bright appearance with proper color.

L.5

5.5.11.b 4) By verifying within 31 days ~~of obtaining the sample~~ that the other properties ~~specified in Table 1 of ASTM D975-81~~ are within the appropriate limits ~~when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D2622-82.~~

LA.6

5.5.11.c d. At least once per 31 days ~~by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-83,~~ and verifying that total particulate contamination is less than 10 mg/liter when tested in accordance with ASTM D2276-~~83~~, Method A*.

See ITS 3.8.3

e. At least once per 18 months, during shutdown, by:

1. Subjecting the diesel engine to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

See ITS 3.8.1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

A.10

*The actions to be taken should any of the properties be found outside of the specified limits are defined in the Bases.

See ITS 3.8.3

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE. See ITS 3.7.13

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

Add proposed ITS 5.5.9 generic program statement A.9

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE: See ITS 3.7.13

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

5.5.9 b. At least once per ~~18~~ 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system while it is in operation that could adversely affect the filter bank or charcoal adsorber capability, by: L.3 A.8

1. Deleted

5.5.9.b 2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm ± 10%.

* The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

** Shared system with D.C. COOK - UNIT 2.

See ITS 3.7.13

A.1

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9.a 3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%. LA.5

5.5.9.c 4. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

5.5.9.c.1 (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.~~ L.4

5.5.9.a, 5.5.9.b 5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.9 c. After every 720 hours of charcoal adsorber operation by either:

5.5.9.c 1. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity; or LA.5

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

LA.5

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity and the samples are prepared by either:

5.5.9.c.1 (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed:

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.~~ L.4

5.5.9 d. At least once per ~~18~~ 24 months by: L.3

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than or equal to 6 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

See ITS 3.7.13

A.1

ITS 5.5

ITS

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9 e. After each complete or partial replacement of a HEPA filter bank by
5.5.9.a verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm ± 10%.

5.5.9 f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a
5.5.9.b halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies. A.9

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ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.11 RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement — A.11

5.5.10, 5.5.10.c 3.11.1 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. LA.7

a. Outside temporary tanks.

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~ LA.7

~~b. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.10.c 4.11.1 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at ~~least once per 7 days when radioactive materials are being added to the tank.~~ LA.7

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. A.11

5.5.10.c * Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement — A.11

5.5.10, 5.5.10.a 3.11.2.1 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 3% by volume if the hydrogen in the system is greater than or equal to 4% by volume. LA.7

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 3% by volume but less than or equal to 4% by volume and containing greater than or equal to 4% hydrogen, restore the concentration of oxygen to less than or equal to 3% or reduce the hydrogen concentration to less than 4% within 96 hours. LA.7

b. With the concentration of oxygen in the waste gas holdup system or tank greater than 4% by volume and greater than 4% hydrogen by volume without delay suspend all additions of waste gases to the system or tank and reduce the concentration of oxygen to less than or equal to 3% or the concentration of hydrogen to less than or equal to 4% within 96 hours in the system or tank.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

5.5.10.a 4.11.2.1 The concentration of oxygen in the waste gas holdup system shall be determined to within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitors required OPERABLE by Table 3.3-12 of Specification 3.3.3.9. LA.7

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Radioactivity Monitoring Program Surveillance Frequencies. — A.11

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.11 RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement — A.11

5.5.10, 5.5.10.b 3.11.2.2 The quantity of radioactivity contained in each gas storage tank shall be limited to 43,800 curies noble gas (considered as Xe-133). LA.7

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA.7

SURVEILLANCE REQUIREMENTS

5.5.10.b 4.11.2.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit ~~at least once per 7 days whenever radioactive materials are added to the tank and at least once per 24 hours during primary coolant system degassing operations.~~ LA.7

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. — A.11

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6.0 ADMINISTRATIVE CONTROLS

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Rev. 2, February 1978.

b. Deleted.

c. Deleted.

d. PROCESS CONTROL PROGRAM implementation.

e. OFFSITE DOSE CALCULATION MANUAL implementation.

f. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Rev. 1, June 1974, and Regulatory Guide 4.1, Rev. 1, April 1975.

See ITS 5.4

5.5.4 g. Component Cyclic or Transient Limits program, which provides controls to track the UFSAR, Section 4.1, cyclic and transient occurrences to ensure that components are maintained within the limits. A.12

h. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes, shall be reviewed prior to implementation as set forth in Quality Assurance Program Description, Appendix C, Section 6.5.

6.8.3 Deleted.

See ITS 5.4

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6.0 ADMINISTRATIVE CONTROLS

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations. See CTS 6.0

b. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.c 6.14.1 Changes to the ODCM:

5.5.1.c.1 a. Shall be documented and records of reviews performed shall be retained ~~as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n.~~ This documentation shall contain: LA.8

5.5.1.c.1.a) 1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.c.1.b) 2. A determination that the change will maintain the level of radioactive effluent control pursuant to 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

LA.8

5.5.1.c.2 b. Shall become effective after ~~review and acceptance by the PORC~~ and the approval of the Plant Manager. LA.9

5.5.1.c.3 c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

Add proposed ITS 5.5.13 and ITS 5.5.15 M.2

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6.8 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5 6.8.4 The following programs shall be established, implemented, and maintained:

5.5.3 a. Radioactive Effluent Controls Program

5.5.3 A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

5.5.3.a 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR 20.1001-20.2402, Appendix B, Table 2, Column 2,

5.5.3.c 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

5.5.3.f 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

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6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY shall be limited to the following:

a) For noble gases: Less than or equal to a dose rate of 500 mrem/year to the total body and less than or equal to a dose rate of 3000 mrem/year to the skin, and

b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

5.5.3.h 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

5.5.3.j 10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

A.2

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Control Program Surveillance Frequencies.

~~b. Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

LA.1

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6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5.12 6.8.5 Technical Specification Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

5.5.12.a a. Changes to the Bases of the Technical Specification shall be made under appropriate administrative controls and reviews.

5.5.12.b b. Licensees may make changes to Bases without prior Nuclear Regulatory Commission approval provided the changes do not require either of the following:

1. A change in the Technical Specification incorporated in the license or

2. A change to the Updated Final Safety Analysis Report of Bases that requires Nuclear Regulatory Commission approval pursuant to 10 CFR 50.59.

5.5.12.c c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the Updated Final Safety Analysis Report.

5.5.12.d d. Proposed changes that meet the criteria of Specification 6.8.5.b above shall be reviewed and approved by the Nuclear Regulatory Commission prior to implementation. Changes to the Bases implemented without prior Nuclear Regulatory Commission approval shall be provided to the Nuclear Regulatory Commission on a frequency consistent with 10 CFR 50.71(e).

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. See ITS 5.6

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

ITS 5.5

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-5-

(s) Deleted by Amendment No. 261

(t) Deleted by Amendment 63

(u) Deleted by Amendment No. 261

5.5.8 (v) Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program ~~shall be described in the station chemistry manual and~~ shall include: L.6

5.5.8.a 1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

5.5.8.b 2. Identification of the procedures used to measure the values of the critical parameters;

5.5.8.c 3. Identification of process sampling points;

5.5.8.d 4. Procedure for the recording and management of data;

5.5.8.e 5. Procedures defining corrective actions for off control point chemistry conditions; and

5.5.8.f 6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

(w) Deleted by Amendment No. 261

(x) Deleted by Amendment No. 261

(y) Deleted by Amendment No. 261

(z) The 72-hour allowed outage time of Technical Specification 3.8.1.1 Action "b" which was entered at 0923, on December 7, 2003, may be extended one time by an additional 72 hours to complete repair and testing of the 2 AB diesel generator.

D. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Donald C. Cook Nuclear Plant Security Plan," with revisions submitted

Amendment No. ~~261~~, 264

A.1

through July 21, 1988; "Donald C. Cook Nuclear Plant Training and Qualification Plan," with revisions submitted through December 19, 1986; and "Donald C. Cook Nuclear Plant Safeguards Contingency Plan," with revisions submitted through June 10, 1988. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

M.1

E. Deleted by Amendment No. 63

F. In all places of this license, the reference to the Indiana and Michigan Electric Company is amended to read Indiana Michigan Power Company.

Add proposed Systems list

5.5.2 G. System Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

24 months

L.1

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency ~~not to exceed refueling cycle intervals~~.

The provisions of SR 3.0.2 are applicable.

~~H. Iodine Monitoring~~

~~The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

~~1. Training of personnel,~~

~~2. Procedures for monitoring, and~~

~~3. Provisions for maintenance of sampling and analysis equipment.~~

LA.2

I. Deleted by Amendment No. 261

(1) Deleted by Amendment No. 261

(2) Deleted by Amendment No. 261

J. The licensee is authorized to use digital signal processing instrumentation in the reactor protection system.

Amendment No. 261

ITS 5.5

A.1

DEFINITIONS

1.29 Deleted.

5.5.1 OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a 1.30 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

5.5.1.b The ODCM shall contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

See ITS Chapter 1.0

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.0 APPLICABILITY

4.0.4 Entry into an OPERATIONAL MODE or other specified condition in the Applicability of a Limiting Condition for Operation shall only be made when the Limiting Condition for Operation's Surveillances have been met within their specified frequency, except as provided by Specification 4.0.3. When a Limiting Condition for Operation is not met due to Surveillances not having been met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4.

This provision shall not prevent entry into OPERATIONAL 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.

See ITS Section 3.0

5.5.6 4.0.5 Surveillance Requirements for inservice ~~inspection/and~~ testing of ASME Code Class 1, 2, and 3 ~~components~~ shall be applicable as follows: pumps and valves LA.3

a. ~~Inservice inspection of ASME Code Class 1, 2, and 3 components and~~ inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a. LA.4 A.15

5.5.6.a b. Surveillance Intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice ~~inspection/and~~ testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: LA.3

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice ~~inspection/and~~ testing criteria	Required frequencies for performing inservice ~~inspection/and~~ testing activities
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
Quarterly or every 3 months	At least once per 92 days
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

A.15 A.3

5.5.6.b c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice ~~inspection/and~~ testing activities. LA.3

d. ~~Performance of the above inservice inspection/and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.4

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification. A.15

Add proposed ITS 5.5.6.c A.5

4.0.6 Deleted

4.0.7 Deleted

ITS 5.5

ITS

A.1

REACTOR COOLANT SYSTEM

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.* See ITS 3.4.13

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

5.5.7 4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

Add proposed ITS 5.5.7 generic program description A.6

5.5.7 4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4.

5.5.7.a The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

5.5.7.a.1 a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

5.5.7.a.2 b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

* This Specification does not apply in Mode 4 while performing crevice flushing as long as Limiting Conditions For Operation for Specification 3.4.1.3 are maintained. See ITS 3.4.13

D.C. COOK - UNIT 2 3/4 4-7 Amendment No. 89

ITS 5.5

ITS

A.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.a.2.a) 1. All nonplugged tubes that previously had detectable wall penetrations (>20%).

5.5.7.a.2.b) 2. Tubes in those areas where experience has indicated potential problems.

5.5.7.a.2.c) 3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5.5.7.a.3 c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

5.5.7.a.3.a) 1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

5.5.7.a.3.b) 2. The inspections include those portions of the tubes where imperfections were previously found.

5.5.7.b The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

ITS

A.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.c 4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

5.5.7.c.1 a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

5.5.7.c.2 b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

5.5.7.c.3 c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

5.5.7.c.3.a) 1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

5.5.7.c.3.b) 2. A seismic occurrence greater than the Operating Basis Earthquake.

5.5.7.c.3.c) 3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

5.5.7.c.3.d) 4. A main steam line or feedwater line break.

ITS

A.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.d 4.4.5.4 Acceptance Criteria

5.5.7.d.1 a. As used in this Specification:

5.5.7.d.1.a) 1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

5.5.7.d.1.b) 2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

5.5.7.d.1.c) 3. Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

5.5.7.d.1.d) 4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5.5.7.d.1.e) 5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

5.5.7.d.1.f) 6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

5.5.7.d.1.g) 7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

5.5.7.d.1.h) 8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

ITS

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REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5.5.7.d.1.l) 9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

5.5.7.d.2 b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Program test Frequencies. A.6

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

See ITS 5.6

D. C. Cook - Unit 2

Table 5.5.7-1

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[2]

Table Notation:

Table 5.5.7-1 Footnote (a)

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

Table 5.5.7-1 Footnote (a)

...he third and fourth steam generators not inspected during the first inservice inspection ...all be inspected during the second and third inspections, respectively. The fourth and ...bsequent inspections shall follow the instructions described in 1 above.

A.1

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A.1

Table 5.5.7-2

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specification 6.9.1	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3.	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1.	N/A	N/A

S = 3(N/n)%   Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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ITS 5.5

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.4 REACTOR COOLANT SYSTEM

See CTS 3/4.4.10.1

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

5.5.5 4.4.10.1 ~~In addition to the requirements of Specification 4.0.5,~~ each reactor coolant pump flywheel shall be inspected by either qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels once every 10 years.

Add proposed ITS 5.5.5 generic program statement

A.13

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency.

ITS 5.5

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.6 CONTAINMENT SYSTEMS

See ITS 3.6.1

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.14 and 5.5.14.a

3.6.1.2 Containment leakage rates shall be limited to:

A.7

5.5.14.b, 5.5.14.c, 5.5.14.d.1 a. An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 12 psig, and

5.5.14.d.1 b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Types B and C tests when pressurized to P_a.

L.2

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

See ITS 3.6.1

5.5.14.d.1 With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, restore the overall integrated leakage rate to $\leq 0.75\ L_a$ and the combined leakage rate for all penetrations and valves subject to Types B and C tests to $\leq 0.60\ L_a$ ~~prior to increasing the Reactor Coolant System temperature above 200°F.~~

L.2

SURVEILLANCE REQUIREMENTS

5.5.14.a 4.6.1.2 Perform leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.163, dated September 1995. See Note 1.

a. Each containment air lock shall be verified to be in compliance with the requirements of Specification 3.6.1.3.

See ITS 3.6.1

b. ~~The provisions of Specification 4.0.2 are not applicable.~~

A.7

Add proposed ITS 5.5.14.e

A.7

Notes:

5.5.14.a.1 1 The Type A testing frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "...at least once per 10 years based on acceptable performance history" is modified to be "...at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in May 1992.

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ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.6 CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

See ITS 3.6.2

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

5.5.14.d.2.a), 5.5.14.b b. An overall air lock leakage rate of ≤ 0.05 L_a at P_a, 12.0 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, maintain at least one door closed; restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.6.2

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

5.5.14.a a. In accordance with 10 CFR 50 Appendix J Option B and Regulatory Guide 1.163, dated September 1995, and

b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

See ITS 3.6.2

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

Add proposed ITS 5.5.9 generic program statement — A.9

4.7.5.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. Deleted

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the system operates for at least 15 minutes.

See ITS 3.7.10

L.3 A.8

5.5.9 c. At least once per ~~18~~ 24 months or (1) after ~~any~~ structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system while it is in operation that could adversely affect the filter bank or charcoal adsorber capability, by:

5.5.9.b 1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

5.5.9.a 2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.

LA.5

5.5.9.c 3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. The carbon samples not obtained from test canisters shall be prepared by either:

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

5.5.9.a, 5.5.9.b 4. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9 d. After every 720 hours of charcoal adsorber operation by either: LA.5

5.5.9.c 1. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H; or LA.5

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 1.0% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. and the samples are prepared by either:

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:~~

~~a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%, and~~

~~b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm ± 10%.~~ L.4

ITS 5.5

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

L.3

5.5.9 e. At least once per ~~18~~ 24 months by:

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

See ITS 3.3.7 and ITS 3.7.10

3. Verifying that the system maintains the control room envelope/pressure boundary at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10% with a makeup air flow rate of ≤ 1000 cfm.

See ITS 3.7.10

5.5.9 / 5.5.9.a f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

5.5.9 / 5.5.9.b g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

A.9

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

ITS 5.5

A.1

ITS

PLANT SYSTEMS

3/4.7.6 ESF VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent ESF ventilation system exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ESF ventilation system exhaust air filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.7.12

SURVEILLANCE REQUIREMENTS

Add proposed ITS 5.5.9 generic program statement

A.9

4.7.6.1 Each ESF ventilation system exhaust air filter train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

See ITS 3.7.12

L.3

A.8

24 months

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

while it is in operation that could adversely affect the filter bank or charcoal adsorber capability

1. Deleted.

5.5.9.b 2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm ± 10%.

5.5.9.a 3. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm ± 10%.

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

LA.5

5.5.9.c 4. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.~~

L.4

5.5.9.a, 5.5.9.b 5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.9 c. After every 720 hours of charcoal adsorber operation by either:

LA.5

5.5.9.c 1. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample obtained from a test shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity; or

LA.5

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity and the samples are prepared by either:

5.5.9.c.1 a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

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ITS

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9.c.2 b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N310-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.~~ L.4

5.5.9 d. At least once per ~~18~~ 24 months by: L.3

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that the standby fan starts automatically on a Containment Pressure--High-High Signal and directs its exhaust flow through the HEPA filters and charcoal adsorber banks on a Containment Pressure--High-High Signal.† See ITS 3.7.12

5.5.9 e. After each complete or partial replacement of a HEPA filter bank by

5.5.9.a verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N310-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5.5.9 f. After each complete or partial replacement of a charcoal adsorber

5.5.9.b bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N310-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

A.9

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

† The provisions of Technical Specification 4.0.8 are applicable. See ITS 3.7.12

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ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.8 ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE: See ITS 3.8.1 and ITS 3.8.3

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank, See ITS 3.8.1

2. Verifying the fuel level in the fuel storage tank, See ITS 3.8.3

3. Verifying that the fuel transfer pump can be started and that it transfers fuel from the storage system to the day tank,

4. Verifying that the diesel starts from standby conditions and achieves in less than or equal to 10 seconds, voltage = 4160±420 V, and frequency = 60±1.2 Hz,* See ITS 3.8.1

5. Verifying the diesel is synchronized and loaded and operates for greater than or equal to 60 minutes at a load of 3500 kw**, and See ITS 3.8.1 and ITS 3.8.3

6. Verifying that the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By removing accumulated water***:

1) From the day tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and See ITS 3.8.1

2) From the storage tanks at least once per 31 days. See ITS 3.8.3

5.5.11.a c. By sampling new fuel oil[redacted] in accordance with the applicable guidelines of ASTM D4057-81 prior to adding new fuel to the storage tanks and

5.5.11.a 1) By verifying, in accordance with the tests specified in ASTM D975-81 and prior to adding the new fuel to the storage tanks, that the sample has: LA.6

Add proposed ITS 5.5.11 generic program statement A.10

* The diesel generator start (10 seconds) from standby conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing and compensatory action may be at reduced acceleration rates as recommended by the manufacturer so that mechanical stress and wear on the diesel engine are minimized. See ITS 3.8.1

** Momentary load transients do not invalidate this test.

*** The actions to be taken should any of the properties be found outside of specified limits are defined in the Bases. See ITS 3.8.3

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.8 ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

within limits

LA.6

5.5.11.a.2 a) A kinematic viscosity ~~of greater than or equal to 1.9 centistokes but less than or equal to 4.1 centistokes at 40°C~~ (alternatively, Saybolt viscosity, ~~SUS at 100°F of greater than or equal to 32.6 but less than or equal to 40.1),~~ ~~if~~ gravity was not determined by comparison with supplier's certification.

within limits

5.5.11.a.2 b) A flash point ~~equal to or greater than 125°F~~

within limits

LA.6

5.5.11.a.1 2) By verifying, ~~in accordance with the test specified in ASTM D1298-80 and~~ prior to adding the new fuel to the storage tanks, that the sample has either an API gravity ~~of greater than or equal to 30 degrees but less than or equal to 40 degrees at 60°F~~ or an absolute specific gravity ~~at 60/60°F of greater than or equal to 0.82 but less than or equal to 0.88,~~ or an API gravity ~~of within 0.3 degrees at 60°F when compared to the supplier's certificate~~ or a specific gravity ~~of within 0.0016 at 60/60°F when compared to the supplier's certificate.~~

of new fuel oil, other than those addressed in Specification 5.5.11.a above,

5.5.11.a.3 3) By verifying, ~~in accordance with the test specified in ASTM D4176-82 and~~ prior to adding new fuel to the storage tanks, that the sample has a clear and bright appearance with proper color.

L.5

5.5.11.b 4) By verifying within 31 days ~~of obtaining the sample~~ that the other properties ~~specified in Table 1 of ASTM D975-81~~ are within the appropriate limits ~~when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D2622-82.~~

LA.6

5.5.11c d. At least once per 31 days ~~by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-83~~ and verifying that total particulate contamination is less than 10 mg/liter when tested in accordance with ASTM D2276-~~83~~, Method A~~*~~

See ITS 3.8.3

e. At least once per 18 months, during shutdown, by:

1. Subjecting the diesel engine to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

See ITS 3.8.1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

A.10

* The actions to be taken should any of the properties be found outside of the specified limits are defined in the Bases.

See ITS 3.8.3

ITS 5.5

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE. See ITS 3.7.13

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specification 3.0.3 are not applicable.

Add proposed ITS 5.5.9 generic program statement A.9

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE: See ITS 3.7.13

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

L.3

5.5.9 b. At least once per ~~18~~ 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system while it is in operation that could adversely affect the filter bank or charcoal adsorber capability, by: A.8

1. Deleted.

5.5.9.b 2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm ± 10%.

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* The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool. See ITS 3.7.13

** Shared system with D. C. COOK - UNIT 1.

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9.a — 3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

LA.5

5.5.9.c — 4. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

5.5.9.c.1 — (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 — (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.~~

L.4

5.5.9.a, 5.5.9.b — 5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

5.5.9 — c. After every 720 hours of charcoal adsorber operation by either:

LA.5

5.5.9.c — 1. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity.

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ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.9 REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

LA.5

5.5.9.c 2. Verifying ~~within 31 days after removal~~ that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 *fpm face* velocity and the samples are prepared by either:

5.5.9.c.1 (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.9.c.2 (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested *in-place* in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.~~ L.4

5.5.9 d. At least once per ~~18~~ 24 months by: L.3

5.5.9.d 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than or equal to 6 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

See ITS 3.7.13

A.1

ITS

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.9
5.5.9.a

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm ± 10%.

5.5.9
5.5.9.b

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies. A.9

ITS 5.5

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.11 RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement — A.11

5.5.10, 5.5.10.c 3.11.1 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. — LA.7

a. Outside temporary tanks.

APPLICABILITY: At all times.

ACTION: — LA.7

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

5.5.10.c 4.11.1 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank. — LA.7

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. — A.11

5.5.10.c * Tanks included in this Specifications are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tanks contents and that do not have tank over flows and surrounding area drains connected to the liquid radwaste treatment system.

ITS 5.5

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement — A.11

5.5.10, 5.5.10.a 3.11.2.1 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 3% by volume if the hydrogen in the system is greater than or equal to 4% by volume. LA.7

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the concentration of oxygen in the waste gas holdup system greater than 3% by volume but less than or equal to 4% by volume and containing greater than or equal to 4% hydrogen, restore the concentration of oxygen to less than or equal to 3% or reduce the hydrogen concentration to less than 4% within 96 hours.~~ LA.7

~~b. With the concentration of oxygen in the waste gas holdup system or tank greater than 4% by volume and greater than 4% hydrogen by volume without delay suspend all additions of waste gases to the system or tank and reduce the concentration of oxygen to less than or equal to 3% or the concentration of hydrogen to less than or equal to 4% within 96 hours in the system or tank.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.10.a 4.11.2.1 The concentration of oxygen in the waste gas holdup system shall be determined to within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitors required OPERABLE by Table 3.3-12 of Specification 3.3.3.9. LA.7

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Radioactivity Monitoring Program Surveillance Frequencies. — A.11

ITS 5.5

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.11 RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

Add proposed ITS 5.5.10 generic program statement — A.11

5.5.10, 5.5.10.b 3.11.2.2 The quantity of radioactivity contained in each gas storage tank shall be limited to 43,800 curies noble gas (considered as Xe-133). — LA.7

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

LA.7

SURVEILLANCE REQUIREMENTS

5.5.10.b 4.11.2.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days whenever radioactive materials are added to the tank and at least once per 24 hours during primary coolant system degassing operations. — LA.7

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. — A.11

ITS 5.5

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Rev. 2, February 1978.

b. Deleted.

c. Deleted.

d. PROCESS CONTROL PROGRAM implementation.

e. OFFSITE DOSE CALCULATION MANUAL implementation.

f. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Rev. 1, June 1974, and Regulatory Guide 4.1, Rev. 1, April 1975.

See ITS 5.4

5.5.4 g. Component Cyclic or Transient Limits program, which provides controls to track the UFSAR, Section 4.1, cyclic and transient occurrences to ensure that components are maintained within the limits. A.12

h. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes, shall be reviewed prior to implementation as set forth in Qualification Assurance Program Description, Appendix C, Section 6.5.

6.8.3 Deleted.

See ITS 5.4

ITS 5.5

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n. This documentation shall contain: — See CTS 6.0

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.c 6.14.1 Changes to the ODCM:

5.5.1.c.1 a. Shall be documented and records of reviews performed shall be retained ~~as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n~~. This documentation shall contain: LA.8

5.5.1.c.1.a) 1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.c.1.b) 2. A determination that the change will maintain the level of radioactive effluent control pursuant to 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.c.2 b. Shall become effective after ~~review and acceptance by the PORC and~~ the approval of the plant manager. LA.8 LA.9

5.5.1.c.3 c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

Add proposed ITS 5.5.13 and ITS 5.5.15 — M.2

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 6.8.4.a specifies the requirements for the Radioactive Effluent Controls Program, however there is no statement as to whether or not the provisions of CTS 4.0.2 and CTS 4.0.3 are applicable. ITS 5.5.3 states that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance Frequencies. This changes the CTS by adding the allowances of ITS SR 3.0.2 and SR 3.0.3 to the Radioactive Effluent Controls Program.

This statement is needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.2 and SR 3.0.3 are not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. In addition, prior to Amendments 189 (Unit 1) and 175 (Unit 2), dated February 10, 1995, these requirements were located in the LCO sections of the Technical Specifications. Amendments 189 (Unit 1) and 175 (Unit 2) relocated the Radiological Effluents Technical Specification from the Technical Specifications to other plant controlled documents, and added CTS 6.8.4.a to the CTS. Since this change is a clarification required to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in nature. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 4.0.5.b does not include all of the required Surveillance Frequencies for performing inservice testing activities. ITS 5.5.6.a adds a new required Frequency of "Biennially or every 2 years." This changes the CTS by adding a new Frequency to the required Frequencies for performing inservice testing activities.

This change is acceptable because the change does not include any new requirements, but only provides clarification of required Frequencies for performing inservice testing activities. Therefore, this change is considered administrative. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 4.0.5.d states that the performance of the above testing activities shall be in addition to other specified Surveillance Requirements. ITS 5.5.6 does not include a similar statement. This changes the CTS by deleting the statement.

CTS 4.0.5.d restates that all applicable requirements must be met. Repeating this overall requirement as a specific detail is redundant and unnecessary. Therefore, this detail can be omitted without any technical change in the requirements and is considered administrative in nature. This change is designated as administrative because it does not result in technical changes to the CTS.

A.5 CTS 4.0.5 specifies the requirements for the Inservice Testing Program, however there is no statement whether the provisions of CTS 4.0.3 are applicable. ITS 5.5.6.c states that the provisions of SR 3.0.3 are applicable to the inservice testing activities. This changes the CTS by adding the allowances of ITS SR 3.0.3 to the Technical Specification Inservice Testing Program requirements.

This statement is needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.3 is not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. Since this change is a clarification required to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in nature. This change is designated as administrative because it does not result in a technical change to the CTS.

A.6 CTS 4.4.5.1, 4.4.5.2, 4.4.5.3, and 4.4.5.4, including Table 4.4-1 and 4.4-2, specify the requirements for the steam generator tube surveillance testing activities. In the ITS, these requirements are included as ITS 5.5.7, "Steam Generator (SG) Program," and a generic statement describing the program has been included. In addition, a statement has been added which states that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Steam Generator Program test Frequencies. This changes the CTS by adding a generic description of the program and specifically stating that the allowances of ITS SR 3.0.2 and SR 3.0.3 are applicable to the Steam Generator Program.

The ITS SR 3.0.2 and SR 3.0.3 statement is needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.2 and SR 3.0.3 are not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. Since this change is a clarification required to maintain provisions that are allowed in the CTS (since CTS 4.0.2 and CTS 4.0.3 apply to the Surveillances of CTS 3/4.4.5), it is considered acceptable. In addition, the generic statement describing the program is also acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A.7 CTS 4.6.1.2 requires the performance of containment leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.1.63, dated September 1995. CTS 4.6.1.2 is also modified by two exceptions. CTS 4.6.1.2.b states that the requirements of Specification 4.0.2 are not applicable. CTS 4.6.1.3.a contains a requirement to perform air lock testing in accordance with 10 CFR 50 Appendix J Option B and Regulatory Guide 1.163, dated September 1995. ITS 5.5.14.a requires a program to establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the listed exceptions. ITS 5.5.14.e states that the provision of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program. This changes the CTS by including the requirements of CTS 4.6.1.2 and 4.6.1.3 in a program, adding the statement that the provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program, and deleting the statement that the provisions of Specification 4.0.2 are not applicable.

This change is acceptable because no changes have been made to the existing requirements. The CTS and proposed ITS 5.5.14 continue to require the same testing to be performed. The statement associated with CTS 4.0.2 is not needed since the Frequency extensions of ITS SR 3.0.2 are not applied to Frequencies identified in the Administrative Controls Section of the ITS, unless specifically identified. The statement associated with ITS SR 3.0.3 is needed to maintain allowances for Surveillance Frequency extensions contained in the ITS since ITS SR 3.0.2 and SR 3.0.3 are not applied to Frequencies identified in the Administrative Controls Chapter of the ITS, unless specifically identified. Since these changes are clarifications required to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative in nature. This change is designated as administrative because it does not result in technical changes to the CTS.

A.8 CTS 4.7.5.1.c, 4.7.6.1.b, and 4.9.12.b require the performance of ventilation filter testing "following painting, fire, or chemical release in any ventilation zone communicating with the system." ITS 5.5.9 requires the performance of the same ventilation filter testing "following painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation that could adversely affect the filter bank or charcoal adsorber capability." This changes the CTS by requiring the filter testing to be performed only if the associated system was in operation and the painting, fire, or chemical release is considered significant enough to adversely affect the filter bank or charcoal adsorber capability.

The purpose of ITS 5.5.9 is to ensure that ventilation filter testing is only performed when there is a potential adverse impact on the affected filter. Current CNP practice is that not all painting, fire, or chemical release results in the need to perform certain ventilation filter tests. Only painting, fire, or chemical release that could affect the functional capability of the ventilation filter trains (i.e., that are significant) would require performance of the tests. The words "that could adversely affect the filter bank or charcoal adsorber capability" were added for clarity and consistency with current practice to avoid a misinterpretation that any painting, fire, or chemical release (such as using a small can of paint to do touch-up work) would result in the need to perform the tests. Similarly, the wording "while it is in operation" was added to clarify that this is the time when the painting, fire, or chemical release could be communicating with the system. This clarification is administrative, and is consistent with other ITS submittals. In addition, the NRC in a letter to Entergy Operations, Inc., dated September 11, 1997, supported the clarification that not all painting, fires, or chemical releases required the filter trains to be tested. Furthermore, this clarification is also consistent with Regulatory Guide 1.52, Revision 3. This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

A.9 The Surveillances (CTS 4.7.5.1.c, 4.7.5.1.d, 4.7.5.1.e.1, 4.7.5.1.f, and 4.7.5.1.g) associated with the ventilation filter testing for the Control Room Emergency Ventilation (CREV) System, the Surveillances (CTS 4.7.6.1.b, 4.7.6.1.c, 4.7.6.1.d.1, 4.7.6.1.e, and 4.7.6.1.f) associated with the ventilation filter testing for the Engineered Safety Features (ESF) Ventilation System, and the Surveillances (CTS 4.9.12.b, 4.9.12.c, 4.9.12.d.1, 4.9.12.e, and 4.9.12.f) associated with the filter testing for the Fuel Handling Area Exhaust Ventilation (FHAEV) System have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.9). As such, a general program statement has been added as ITS 5.5.9. Also, a statement of the applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extension do apply. This changes the CTS by moving the ventilation filter testing Surveillances associated with the CREV, ESF Ventilation, and FHAEV Systems to a program in ITS 5.5 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

The addition of the program statement is acceptable because it is describing the intent of the CTS Surveillances. The addition of the ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A.10 The Surveillances associated with diesel fuel oil testing (CTS 4.8.1.1.2.c and d) have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.11). As such, a general program statement has been added as ITS 5.5.11. Also, a statement of the applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extension do apply. This changes the CTS by moving the diesel fuel oil testing Surveillances to a program in ITS 5.5 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

The addition of the program statement is acceptable because it is describing the intent of the CTS Surveillances. The addition of the ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A.11 The liquid holdup tank requirements in CTS 3/4.11.1, the explosive gas mixture requirements in CTS 3/4.11.2.1, and the gas storage tank requirements in CTS 3/4.11.2.2 have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.10). As such, a general program statement has been added. Also, a statement of applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for Surveillance Frequency extensions do apply. This changes the CTS by moving the liquid holdup tank, explosive gas mixture, and gas storage tank requirements to a program in ITS 5.5.10 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

The addition of the program statement is acceptable because it is describing the intent of the CTS Specifications. The addition of the ITS SR 3.0.2 and SR 3.0.3

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statement is a clarification needed to maintain provisions that are currently allowed in the LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A.12 CTS 6.8.1.g requires written procedures to be established, implemented and maintained covering the activities of the component cyclic or transient limits program, which provides controls to track the UFSAR Section 4.1, cyclic and transient occurrences to ensure that components are maintained within the limits. ITS 5.5.4 requires a program to track the UFSAR, Section 4.1 cyclic and transient occurrences to ensure that components are maintained within the design limits. This changes the CTS by placing the requirements of the Component Cyclic or Transient Limits Program currently located in the procedure section of the CTS Administration Controls Chapter into the Program section of the ITS Administrative Controls Chapter.

One purpose of CTS 6.8.1.g is to ensure that there is a program to track the UFSAR, Section 4.1 cyclic and transient occurrences to ensure that components are maintained within the design limits. Since this change is a clarification that CTS 6.8.1.g also requires a program to be established, it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A.13 CTS 4.4.10.1 requires the inspection of each reactor coolant pump flywheel. ITS 5.5.5 requires a program to provide for the inspection of each reactor coolant pump flywheel. In addition, a statement has been added which states the provisions of ITS SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency. This changes the CTS by including the requirements of CTS 4.4.10.1 in a program in the Administrative Controls Chapter of the Technical Specifications instead of as a Surveillance and specifically stating that the allowances of ITS SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency. Other changes to 3/4.4.10.1 is discussed in the Discussion of Changes for CTS 3/4.4.10.1.

This change is acceptable because no changes have been made to the existing requirements. The CTS and proposed ITS 5.5.5 continue to require the same reactor coolant pump flywheel inspections to be performed. The ITS SR 3.0.2 and SR 3.0.3 statement is needed to maintain allowances for Surveillance Frequency extensions contained in the CTS because ITS SR 3.0.2 and SR 3.0.3 are not normally applied to Frequencies identified in the Administrative Controls Chapter of the ITS. Since this change is a clarification required to maintain provisions that are allowed in the CTS (since CTS 4.0.2 and CTS 4.0.3 apply to the Surveillances of CTS 3/4.4.10), it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A.14 (Unit 1 only) CTS 4.4.5.4.a does not contain a definition for Preservice Inspection. ITS 5.5.7.d.1.i) includes the definition. This changes the Unit 1 CTS by adding a definition for Preservice Inspection.

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

CTS 4.4.5.2.b, 4.4.5.3.a, and Table 4.4-1 (ITS 5.5.7.a.2, ITS 5.5.7.e.1, and ITS Table 5.5.7-1) refer to a preservice inspection. This proposed change is acceptable because the definition is consistent with the definition for preservice inspection in CTS 4.4.5.4.a.9 for Unit 2, and because ITS 5.5.7.a.2, 5.5.7.e.1, and Table 5.5.7-1 continue to refer to the preservice inspections. This change is designated as administrative because it does not result in technical changes to the CTS.

A.15 CTS 4.0.5 requires pump and valve testing per the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. ITS 5.5.6 requires pump and valve testing per the requirements of the ASME Operation and Maintenance Standards and Guides (OM Codes). This changes the CTS by referring to the ASME OM Codes instead of ASME Boiler and Pressure Code, Section XI.

In the 1987 Addenda to the 1986 edition of ASME Boiler and Pressure Vessel Code, Section XI, the requirements for Inservice Testing were removed and relocated to the ASME/ANSI OM Codes. This change was endorsed in 10CFR50.55a. 10CFR50.55a(f) now addresses the requirements for inservice testing using the ASME/ANSI OM Codes and 10CFR50.55a(g) addresses the requirements for inservice inspection using ASME Boiler and Pressure Vessel Code, Section XI. The ITS has been revised to incorporate the current Code requirements. In addition, the terms weekly, monthly, and semiannually are not used in the applicable ASME/ANSI OM Codes. Therefore, these Frequencies have been deleted. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 License Conditions 2.H (Unit 1) and 2.G (Unit 2) provide the requirements for a System Integrity program. The program is not explicit as to which systems outside containment must be monitored. ITS 5.5.2 includes the requirements for the Leakage Monitoring Program and provides a list of systems that should be monitored because they could contain highly radioactive fluids during a serious transient or accident.

The purpose of the Leakage Monitoring Program is to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems added to the Specification include the Safety Injection System, Chemical and Volume Control System, Residual Heat Removal System, Containment Spray System, post accident sampling, and the boron injection tank injection flowpath of the Centrifugal Charging System. The change is acceptable because these systems are currently monitored to satisfy the current License Conditions and is a complete list of those systems that could contain highly radioactive fluids during a serious transient or accident. This change is designated as more restrictive because it adds an explicit list of systems to the Technical Specifications.

M.2 The CTS does not include program requirements for a Safety Function Determination Program or Battery Monitoring and Maintenance Program. The

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ITS includes programs for these activities. This changes the CTS be adding the following programs:

ITS 5.5.13, "Safety Function Determination Program (SFDP)"; and
ITS 5.5.15, "Battery Monitoring and Maintenance Program."

The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the Technical Specifications. The Battery Monitoring and Maintenance Program is included to provide for battery restoration and maintenance. The specific wording associated with these two programs may be found in ITS 5.5.13 and ITS 5.5.15. The changes are acceptable because they support implementation of the requirements of the ITS and the UFSAR. This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 6 - Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 6.8.4.b, "Radiological Environmental Monitoring Program," describes a program to monitor the radiation and radionuclides in the environs of the plant. ITS Chapter 5.0 does not require such a program. This changes the CTS by moving the requirements for the Radiological Environmental Monitoring Program to the Offsite Dose Calculation Manual (ODCM).

The purpose of CTS 6.8.4.b is to provide representative measurements of radioactivity in the highest potential exposure pathways, and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The removal of the requirement for this program 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. ITS 5.6.2 still requires an annual report of the results of the "Radiological Environmental Monitoring Program." Also, this change is acceptable because these types of procedural details will be adequately controlled in the ODCM. Changes to the ODCM are controlled by the ODCM change control process in ITS 5.5.1, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of requirement change because the requirements for a program are being removed from the Technical Specifications.

LA.2 *(Type 6 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPD, or IIP)* Operating License Conditions 2.I (Unit 1) and 2.H (Unit 2) specify that the Iodine Monitoring Program shall be implemented and provides a description of what the program shall include. ITS 5.5 does not include this program. This changes the CTS by moving the details of the Iodine Monitoring Program to the Technical Requirements Manual (TRM).

The removal of this requirement 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. This program is required by the CNP Units 1 and 2 commitment to NUREG-0578, Item 2.1.8.c, as stated in a letter from R.S. Hunter (AEP) to Harold R. Denton (NRC) dated December 10, 1980. The program is designed to minimize radiation exposure to plant personnel in vital areas of the plant after an accident, and has no impact on nuclear safety or the health and safety of the public. The training aspect of the program is accomplished as part of the continual training program for personnel in the cognizant organizations, as well as during training for those individuals responsible for implementing the radiological emergency planning procedures. Provisions for monitoring and performing maintenance of the sampling and analysis equipment are addressed in chemistry and radiation protection procedures. This change is acceptable because the program requirements 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 requirement change because requirements are being removed from the Technical Specifications.

LA.3 *(Type 6 - Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 4.0.5 provides requirements for the Inservice Inspection Program. The ITS does not include Inservice Inspection Program requirements. In addition, since the Inservice Testing Program is the only requirement remaining, the reference to ASME Code Class 1, 2, and 3 "components" has been changed to "pumps and valves" for clarity. Pumps and valves are the only components related to the Inservice Testing Program (as described in CTS 4.0.5.a). This changes the CTS by moving these requirements from the Technical Specifications to the Inservice Inspection Program (IIP).

The removal of these requirements 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 Technical Specifications still retain requirements for the affected components to be OPERABLE. Also, this change is acceptable because these requirements will be adequately controlled by the IIP, which is required by 10 CFR 50.55a. Compliance with 10 CFR 50.55a is required by the CNP Units 1 and 2 Operating Licenses. This change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications.

LA.4 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.0.5.a specifies that the Inservice Testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a. ITS 5.5.6 states that the Inservice Testing Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. This changes the CTS by

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moving these procedural details from the Technical Specifications to the Inservice Testing Program.

The removal of these details for meeting Technical Specification 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 requirements for the control for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. Also, this change is acceptable because these types of details will be adequately controlled in the plant controlled Inservice Testing Program. Changes to the Inservice Testing Program will be controlled by the provisions of 10 CFR 50.55a. This change is designated as a less restrictive removal of detail change because the details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.5 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.7.5.1.c.3, 4.7.5.1.d.1, 4.7.5.1.d.2, 4.7.6.1.b.4, 4.7.6.1.c.1, 4.7.6.1.c.2, 4.9.12.b.4, 4.9.12.c.1, and 4.9.12.c.2 require that within 31 days after removal of a carbon sample the laboratory analysis results are shown to be within limit. ITS 5.5.9.c requires the same analysis to be performed however the detail of "within 31 days after removal of a carbon sample" is not included. This changes the CTS by moving these procedural details from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to perform the testing at the appropriate Frequencies. Also, this change is acceptable because these types of procedural details 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 procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.6 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.8.1.1.2.c, 4.8.1.1.2.c.1), 4.8.1.1.2.c.1)a), 4.8.1.1.2.c.1)b), 4.8.1.1.2.c.2), 4.8.1.1.2.c.3), 4.8.1.1.2.c.4), and 4.8.1.1.2.d specify test and sampling requirements for new diesel fuel oil and diesel fuel oil in the storage tanks in accordance with certain ASTM standards (i.e., D4057-81, D975-81, D1298-80, D4176-82, D2622-82, and D2276-83) and provide limits for kinematic viscosity, flash point, API gravity, absolute specific gravity, and specific gravity. ITS 5.5.11 does not include either the explicit reference to the ASTM standards or the specific limits, but continues to require the verification that the new and stored diesel fuel oil is tested in accordance with the applicable standards and that the parameters are within limits. This changes the CTS by moving the procedural details on the testing requirements and the specific limits to the Bases of ITS 3.8.3.

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The removal of these details for performing Surveillance Requirements from the CTS 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 requirement to determine that new and stored diesel fuel oil are within the applicable limits. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

LA.7 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3/4.11.1 includes the details for implementing the requirements for the liquid holdup tank. CTS 3/4.11.2.1 includes the details for implementing the requirements for the explosive gas mixture. CTS 3/4.11.2.2 includes the details for implementing the requirements for the gas storage tank. The details for implementing these requirements, including the specific limits, are not included in the ITS. The ITS only includes a requirement to maintain a program for these requirements. This changes the CTS by moving these procedural details for implementing the requirements, including the specific limits, from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details for the specific limits, Applicability, Actions, and 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. ITS 5.5.10 still retains the requirement to include a program, which provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor temporary liquid storage tanks. Also, this change is acceptable because these types of procedural details 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 procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.8 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.14.1.a requires changes to the ODCM to be documented and records of reviews performed to be retained as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n. CTS 6.14.1.b requires changes to the ODCM to be effective after review and acceptance by the PORC and the approval of the plant manager. ITS 5.5.1.c.1 requires changes to the ODCM to be documented and records of reviews performed to be retained. ITS 5.5.1.c.2 requires changes to the ODCM to become effective after the approval of the plant manager. This changes the CTS by moving the record retention requirement reference and the PORC review and approval requirement to the Quality Assurance Program Description (QAPD).

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The removal of these details, which are related to meeting Technical Specification 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. ITS 5.5.1 still retains the requirement for changes to the ODCM. Also, this change is acceptable because these types of procedural details will be adequately controlled in the QAPD. Any changes to the QAPD are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.9 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.14.1.b uses the title "Plant Manager." ITS 5.5.1.c.2 uses the generic title "plant manager." This changes the CTS by moving the specific CNP organizational title to the UFSAR and replacing it with a generic title.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific CNP organizational title out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the plant manager are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 10 – 18 to 24 Month Surveillance Frequency Change, Non-Channel Calibration Type)* License Conditions 2.H (Unit 1) and 2.G (Unit 2) specify that the integrated leak test requirements for each system outside containment that would or could contain highly radioactive fluids during a serious transient or accident must be performed at a frequency not to exceed refueling cycle intervals. ITS 5.5.2 specifies that the same test must be performed at least once per 24 months and an allowance has been added which states that the provisions of ITS SR 3.0.2 are applicable. This changes the CTS by extending the Frequency of the Surveillance from 18 months (i.e., the current CNP normal refueling cycle interval) to 24 months (i.e., a maximum of 30 months accounting for the allowable grace period specified in ITS SR 3.0.2).

The purpose of License Conditions 2.H (Unit 1) and 2.G (Unit 2) is to ensure the leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident is reduced to as low as

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practicable levels. This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical surveillance data and maintenance data sufficient to determine failure modes have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. Extending the Surveillance test interval for the System Integrity integrated leak test verification SR is acceptable because most portions of the subject systems included in this program are visually walked down, while the plant is operating, during plant testing, and/or operator/system engineer walkdowns. In addition, housekeeping/safety walkdowns also serve to detect any gross leakage. If leakage is observed from these systems, corrective actions will be taken to repair the leakage. Finally, the plant radiological surveys will also identify any potential sources of leakage. These visual walkdowns and surveys provide monitoring of the systems at a greater frequency than once per refueling cycle, and support the conclusion that the impact, if any, on safety is minimal as a result of the proposed changes. Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.2 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.6.1.2.a specifies that the overall integrated leakage rate shall be limited to $\leq L_a$. CTS 3.6.1.2.b specifies that combined leakage rate shall be limited to $\leq 0.60\ L_a$ for all penetrations and valves subject to Types B and C tests. However, the CTS 3.6.1.2 Action does not allow the unit to increase Reactor Coolant System temperature above 200°F if either the measured overall integrated leakage rate exceeds $0.75\ L_a$ or if the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeds $0.60\ L_a$. ITS 5.5.14 specifies that the containment leakage rate acceptance criterion is $1.0\ L_a$ and that during the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests. This changes the CTS by only requiring the $0.60\ L_a$ and $0.75\ L_a$ limits to be met during the first unit startup following testing in accordance with the Containment Leakage Rate Testing Program.

The purpose of ITS 5.5.14 is to ensure the appropriate limits are specified for the Containment Leakage Rate Testing Program. This change is acceptable because the acceptance limits continue to ensure the containment leakage is within the value assumed in the accident analysis. Currently, the overall integrated leakage rate of $\leq L_a$ and the combined leakage rate of $\leq 0.6\ L_a$ applies in MODES 1, 2, 3, and 4. The CTS 3.6.1.2 Action will not allow the unit to enter MODE 4 from MODE 5 unless the integrated leakage rate is $\leq 0.75\ L_a$ and the combined leakage rate for all penetrations and valves subject to Types B and C

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tests is ≤ 0.60 L_a. In the ITS, the containment leakage rate acceptance criterion is ≤ 1.0 L_a and is applicable in MODES 1, 2, 3, and 4. The other limits (i.e., ≤ 0.60 L_a and ≤ 0.75 L_a) are only applicable during the first unit startup following testing in accordance with this program. This will allow subsequent unit startups (after the first unit startup following testing in accordance with the program) to proceed as long as the containment leakage rate acceptance criterion of ≤ 1.0 L_a is met. This is acceptable because the leakage limit of L_a is assumed in the accident analysis. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L.3 *(Category 10 – 18 to 24 Month Surveillance Frequency Change, Non-Channel Calibration Type)* CTS 4.7.5.1.c, 4.7.5.1.e.1, 4.7.6.1.b, 4.7.6.1.d.1, 4.9.12.b, and 4.9.12.d.1 require the performance of ventilation filter testing once per 18 months. ITS 5.5.9 requires these same Surveillances to be performed once per 24 months. This changes the CTS by extending the Frequency of the Surveillance from 18 months (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and ITS SR 3.0.2) to 24 months (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and ITS SR 3.0.2).

The purpose of CTS 4.7.5.1.c, 4.7.5.1.e.1, 4.7.6.1.b, 4.7.6.1.d.1, 4.9.12.b, and 4.9.12.d.1 is to ensure that the Control Room Emergency Ventilation (CREV) System, the Engineered Safety Features (ESF) Ventilation System, and the Fuel Handling Area Exhaust Ventilation (FHAEV) System charcoal adsorbers and HEPA filters can perform their safety function. This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical surveillance data and maintenance data sufficient to determine failure modes have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. Extending the Surveillance test interval for the HEPA filter dioctyl phthalate (DOP) tests, the charcoal adsorber halogenated hydrocarbon refrigerant tests, the laboratory analysis test, and the flow test is acceptable since other tests may be required to be performed during the operating cycle. Tests described in ITS 5.5.9.a (the HEPA filter test) and 5.5.9.b (the charcoal adsorber halogenated hydrocarbon), shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability. Tests described in ITS 5.5.9.c (laboratory test of the charcoal sample) shall be performed once per 24 months; after 720 hours of adsorber operation, after any structural maintenance on the HEPA filter or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability. The additional Surveillance

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Frequencies are adequate to ensure the filters remain OPERABLE during the cycle. Tests described in ITS 5.5.9.d (combined pressure drop across the combined HEPA filter and charcoal adsorbers) shall be performed once per 24 months. The CREV, ESF Ventilation, and FHAEV Systems are required to be tested every 46 days on a STAGGERED TEST BASIS (CREV and ESF Ventilation Systems) or 92 days (FHAEV System) for ≥ 15 minutes. This testing ensures that a significant portion of the associated ventilation system is operating properly and will detect significant failures. Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.4 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.7.5.1.d.2 requires the performance of a halogenated hydrocarbon refrigerant gas test on the CREV System charcoal adsorber and a DOP test on the CREV System HEPA filter banks after the reinstallation of the adsorber tray used for obtaining a carbon sample. CTS 4.7.6.1.b.4 and 4.7.6.1.c.2 require the performance of a halogenated hydrocarbon refrigerant gas test on the ESF Ventilation System charcoal adsorber after the reinstallation of the adsorber tray used for obtaining a carbon sample. CTS 4.9.12.b.4 and 4.9.12.c.2 require the performance of a halogenated hydrocarbon refrigerant gas test on the FHAEV System charcoal adsorber after the reinstallation of the adsorber tray used for obtaining a carbon sample. ITS 5.5.9 does not contain these explicit post maintenance testing requirements. This changes the CTS by deleting these explicit post maintenance requirements.

The purpose of CTS 4.7.5.1.d.2, 4.7.6.1.b.4, 4.7.6.1.c.2, 4.9.12.b.4, and 4.9.12.c.2 is to verify the OPERABILITY of the ventilation filter trains after the reinstallation of the adsorber tray used for taking a carbon sample. This change is acceptable because the deleted Surveillance Requirements are not necessary to verify the equipment used to meet the LCO can perform its required function. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. Any time the OPERABILITY of a system or component has been affected by repair, maintenance, modification, or replacement of a component, post maintenance testing is required to demonstrate the OPERABILITY of the system or component. This is described in the Bases for ITS SR 3.0.1 and required under ITS SR 3.0.1. The OPERABILITY requirements for the affected ventilation filter trains are described in the Bases for ITS 3.7.10, 3.7.12, and 3.7.13. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control) provide adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50, Appendix B is required under the Units 1 and 2 Operating Licenses. As a result, post maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. In addition, ITS 5.5.9 requires the performance of ITS 5.5.9.a (a halogenated hydrocarbon refrigerant gas test on the charcoal adsorber) and ITS 5.5.9.b (a DOP test on the HEPA filter banks) after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing. Therefore, if after the reinstallation of the adsorber tray used for obtaining a carbon sample it is determined that ITS 5.5.9.a or 5.5.9.b are not met, the applicable ITS SRs must be declared not met and the appropriate Required Actions must be entered. Therefore, although the explicit Surveillance Frequency has been deleted, both ITS SR 3.0.1 and ITS 5.5.9 will require the performance of these tests if it is determined that the Surveillances may not be satisfied after reinstallation of the adsorber trays. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.5 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.8.1.1.2.c.4 requires the evaluation that certain diesel fuel oil properties are within the appropriate limits within 31 days of obtaining the sample. ITS 5.5.11.b requires this same evaluation to be performed within 31 days following addition of the new fuel oil to the storage tanks. This changes the CTS by changing the time by which the evaluation for these properties must be completed.

The purpose of ITS 5.5.11.b is to ensure that the properties of the new diesel fuel oil added to the storage tanks are acceptable. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. CTS 4.8.1.1.2.c.4 requires the evaluation that certain diesel fuel oil properties are within the appropriate limits within 31 days of obtaining the sample, while the ITS time limit begins after the fuel oil is added to the storage tanks. The new fuel oil can affect the stored fuel oil only when it is added to the storage tanks. Failure to meet the limit for these other fuel oil properties would not have an immediate effect on diesel generator operation because the oil added is normally only a small portion of the entire fuel oil storage volume. The 31 day period is also acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on diesel generator operation. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.6 *(Category 1 – Relaxation of LCO Requirements)* Operating License Conditions 2.C.(7) (Unit 1) and 2.C.(3)(v) (Unit 2) specify that the Secondary Water Chemistry Monitoring Program shall be described in the station chemistry manual and provides a description of what the manual should contain. ITS 5.5.8 does not specify that the program must be described in the station chemistry manual. It only states what shall be included in the Secondary Water Chemistry Program. This changes the CTS by deleting the details of where the description of the Secondary Water Chemistry Program shall reside from the Technical Specifications.

The purpose of the Secondary Water Chemistry Program is to ensure proper controls are placed on monitoring secondary water chemistry in order to inhibit steam generator tube degradation. The change is acceptable because the Technical Specifications still retain the requirement to have a Secondary Water Chemistry Program and the Technical Specifications continue to describe the contents of the program. Thus, the Technical Specifications continue to control the general content of the program and any changes will still require NRC approval. In addition, removal of this detail for meeting Technical Specification requirements (i.e., the actual location of the program) 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. This change is designated as less restrictive because less stringent Technical Specifications requirements are being applied in the ITS than were applied in the CTS.

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

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5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

6.8.4 The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

Definition 1.30

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and (3)

Definition 1.30

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.6.2] and Specification [5.6.3]. (11) (1)

6.14.1 c. Licensee initiated changes to the ODCM: (2)

6.14.1.a 1 Shall be documented and records of reviews performed shall be retained. This documentation shall contain: (2)

6.14.1.a.1 1a Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and (i) (3) (2)

6.14.1.a.2 b A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations; (i) (3) (2)

6.14.1.b 2 Shall become effective after the approval of the plant manager; and (i) (3) (2) (2)

6.14.1.c 3 Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

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5.5 Programs and Manuals

Unit 1 License Condition 2.H, Unit 2 License Condition 2.G

5.5.2 ~~Primary Coolant Sources Outside Containment~~ Leakage Monitoring Program (6)

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include ~~Recirculation Spray,~~ Safety Injection, Chemical and Volume Control, ~~gas stripper, and Hydrogen Recombiner.~~ INSERT 1 (1) The program shall include the following: System (3) (3)

a. Preventive maintenance and periodic visual inspection requirements and

b. Integrated leak test requirements for each system at least once per ~~18~~ 24 (1) months.

The provisions of SR 3.0.2 are applicable.

~~5.5.3 [Post Accident Sampling~~ (4)

~~- REVIEWER'S NOTE -~~
~~This program may be eliminated based on the implementation of WCAP-14986, Rev. 1, "Post Accident Sampling System Requirements: A Technical Basis," and the associated NRC Safety Evaluation dated June 14, 2000.~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~

6.8.4.a 5.5.~~4~~ 3 (3) Radioactive Effluent Controls Program (4)

6.8.4.a

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1 **INSERT 1**

Residual Heat Removal System, Containment Spray System, post accident sampling, and the boron injection tank injection flowpath of the Centrifugal Charging System

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5.5 Programs and Manuals

5.5.9 3 Radioactive Effluent Controls Program (continued) 11

6.8.4.a.1) a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM, ; 3

6.8.4.a.2) b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402, ; 3

6.8.4.a.3) c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM, ; 3

6.8.4.a.4) d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I, ; 3

and projected

6.8.4.a.5) e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days, 6 ~~Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,~~ ; 3

6.8.4.a.6) f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I, ; 3

6.8.4.a.7) g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following: 6

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and ; 3

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ, ;

6.8.4.a.8) h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, ; 3

CTS

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued) 4 11

3

6.8.4.a.9) i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and ; 3

6.8.4.a.10) j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

DOC A.2 The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency. ies 11 6

6.8.1.g 5.5.5 4 Component Cyclic or Transient Limit 5 4.1 4 1 6

This program provides controls to track the FSAR, Section 4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 [Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.] 7

4.4.10.1 5.5.7 5 Reactor Coolant Pump Flywheel Inspection Program 4

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. 8

penetrant testing, or combination of the two tests

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT) and/or (PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI. 8 5 8

magnetic particle testing

shall

once every

DOC A.13

INSERT 1A

21

TSTF-421 Not shown

21 **INSERT 1A**

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency.

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5.5.7 Reactor Coolant Pump Flywheel Inspection Program (continued)

9

- REVIEWER'S NOTES -

1. The inspection interval and scope for RCP flywheels stated above can be applied to plants that satisfy the staff requirements in the safety evaluation of Topical Report, WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination."

2. Licensees shall confirm that the flywheels are made of SA 533 B material. Further, licensees having Group-15 flywheels (as determined in WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination") need to demonstrate that material properties of their A516 material is equivalent to SA 533 B material, and its reference temperature, RT, is less than 30 °F.

TSTF-421 Not shown

3. For flywheels not made of SA 533 B or A516 material, licensees need to either demonstrate that the flywheel material properties are bounded by those of SA 533 B material, or provide the minimum specified ultimate tensile stress, the fracture toughness, and the reference temperature, RT_{NDT}, for that material. For the latter, the licensees should employ these material properties, and use the methodology in the topical report, as extended in the two responses to the staff's RAI, to provide an assessment to justify a change in inspection schedule for their plants.

4. Licensees with Group-10 flywheels need to confirm that their flywheels have an adequate shrink fit to preclude loss of shrink fit of the flywheel at the maximum overspeed, or to provide an evaluation demonstrating that no detrimental effects would occur if the shrink fit was lost as maximum overspeed.

4.0.5 5.5.8 6 Inservice Testing Program 4

4.0.5 This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 ~~components~~ pumps and valves. The program shall include the following: 10

4.0.5.b a. Testing frequencies specified in ~~Section XI of~~ the ASME ~~Boiler and Pressure Vessel Code~~ Operation and Maintenance Standards and Guides (OM Codes) and applicable Addenda are as follows: 11 10 10 11

10

ASME ~~Boiler and Pressure Vessel Code~~ OM Codes and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days

10

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5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued) (6) (4) (11)

4.0.5.b

ASME ~~Boiler and Pressure Vessel Code~~ OM Codes and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities	
~~Monthly~~	~~At least once per 31 days~~	(10)
Quarterly or every 3 months	At least once per 92 days	
~~Semiannually or every 6 months~~	~~At least once per 184 days~~	
~~Every 9 months~~	~~At least once per 276 days~~	(10)
Yearly or annually	At least once per 366 days	
Biennially or every 2 years	At least once per 731 days	

(10)

4.0.5.c
b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities. (11)

DOC A.5
c. The provisions of SR 3.0.3 are applicable to inservice testing activities, and

4.0.5.e
d. Nothing in the ASME ~~Boiler and Pressure Vessel Code~~ OM Codes shall be construed to supersede the requirements of any TS. (10)

5.5.7 (7) Steam Generator (SG) ~~Tube Surveillance~~ Program (4) (6) (12)

4.4.5.1, 4.4.5.2, 4.4.5.3, 4.4.5.4, Tables 4.4-1 and 4.4-2

INSERT 2

~~- REVIEWER'S NOTE -~~
~~The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

DOC A.6
The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG ~~Tube Surveillance~~ Program test frequencies. (12)

INSERT 2A (11) (6)

Unit 1 License Condition 2.C.(7), Unit 2 License Condition 2.C.(3)(v)

5.5.8 (8) Secondary Water Chemistry Program (4)

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: (13)

a. Identification of a sampling schedule for the critical variables and control points for these variables; (3)

b. Identification of the procedures used to measure the values of the critical variables; (3)

12

INSERT 2

This program provides requirements for steam generator tube sample selection and inspection. Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.7-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.7-2. The inservice inspection of steam generator tubes shall be performed at the Frequencies specified in Specification 5.5.7.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.7.d.

a. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators. The tubes selected for these inspections shall be selected on a random basis except:

 1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

 2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

  a) All nonplugged tubes that previously had detectable wall penetrations greater than or equal to 20%;

  b) Tubes in those areas where experience has indicated potential problems; and

  c) A tube inspection pursuant to Specification 5.5.7.d.1.h) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection;

 3. The tubes selected as the second and third samples (if required by Table 5.5.7-2) during each inservice inspection may be subjected to a partial tube inspection provided:

  a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found; and

  b) The inspections include those portions of the tubes where imperfections were previously found.

b. The results of each sample inspection shall be classified into one of the following three categories:

12

INSERT 2 (continued)

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	Greater than or equal to 5% and less than or equal to 10% of the total tubes inspected are degraded tubes or one or more tubes, but not more than 1% of the total tubes inspected, are defective.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than or equal to 10%) further wall penetrations to be included in the above percentage calculations.

c. The above required inservice inspections of steam generator tubes shall be performed at the following Frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality or replacement of steam generators. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under All Volatile Treatment conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection Frequency may be extended to a maximum of once per 40 months.

2. If the results inservice inspection of a steam generator conducted in accordance with Table 5.5.7-2 at 40 month intervals fall in Category C-3, the inspection Frequency shall be increased to once per 20 months. The increase in inspection Frequency shall apply until a subsequent inspection satisfies the criteria of Specification 5.5.7.c.1, at which time the Frequency may be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.7-2 during the shutdown subsequent to any of the following conditions:

   a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of LCO 3.4.13;

12

INSERT 2 (continued)

b) A seismic occurrence greater than the Operating Basis Earthquake;

c) A loss of coolant accident requiring actuation of the engineered safety features; or

d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

a) Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

b) Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

c) Degraded Tube means an imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation:

d) Percent Degradation means the percentage of the tube wall thickness affected or removed by degradation;

e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service. Any tube which, upon inspection, exhibits tube wall degradation of 40% or more of the nominal tube wall thickness shall be plugged prior to returning the steam generator to service;

g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a main steam line or feedwater line break, as specified in Specification 5.5.7.c.3 above;

h) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support to the cold leg; and

i) Preservice Inspection means an inspection of the full length of each tube in the steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial entry into MODE 1 using the equipment and techniques expected to be used during subsequent inservice inspections.

12

INSERT 2 (continued)

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.7-2.

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INSERT 2A

Table 5.5.7-1 (page 1 of 1)
Minimum Number of Steam Generators to be Inspected During Inservice Inspection

Preservice Inspection	Yes
Number of Steam Generators per Unit	4
First Inservice Inspection	2
Second and Subsequent Inservice Inspections	1[(a)]

(a) The third and fourth steam generators not inspected during the first inservice inspection shall be inspected during the second and third inspections, respectively. The fourth and subsequent inspections may be limited to one steam generator on a rotating schedule encompassing 3 N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

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INSERT 2A (continued)

Table 5.5.7-2 (page 1 of 1)
Steam Generator (SG) Tube Inspection

<table>
<tr><td colspan="3">First Sample Inspection</td><td colspan="2">Second Sample Inspection</td><td colspan="2">Third Sample Inspection</td></tr>
<tr><td>Sample Size</td><td>Result</td><td>Action Required</td><td>Result</td><td>Action Required</td><td>Result</td><td>Action Required</td></tr>
<tr><td rowspan="9">A minimum of S tubes per SG</td><td>C-1</td><td>None</td><td>NA</td><td>NA</td><td>NA</td><td>NA</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this SG</td><td>C-1</td><td>None</td><td>NA</td><td>NA</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this SG</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Plug defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>NA</td><td>NA</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this SG, plug defective tubes, inspect 2S tubes in each other SG, and notify NRC pursuant to Specification 5.6.7</td><td>All other SGs are C-1</td><td>None</td><td>NA</td><td>NA</td></tr>
<tr><td>Some SGs are C-2, but no additional SGs are C-3</td><td>Perform action for C-2 result for second sample</td><td>NA</td><td>NA</td></tr>
<tr><td>Additional SG is C-3</td><td>Inspect all tubes in each SG, plug or repair defective tubes, and notify NRC pursuant to Specification 5.6.7</td><td>NA</td><td>NA</td></tr>
</table>

Where: S = 3 (N/n)%;
N is the number of SGs in the unit; and
n is the number of SGs inspected during an inspection.

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5.5 Programs and Manuals

Unit 1 License Condition 2.C(7), Unit 2 License Condition 2C.(3)(v)

5.5.17 Secondary Water Chemistry Program (continued)

c. Identification of process sampling points ~~which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage~~;

d. Procedures for the recording and management of data ~~,~~ ;

e. Procedures defining corrective actions for all off control point chemistry conditions ~~,~~ ; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

4.7.5.1.c, d, e, f, g, 4.7.6.1.b, c, d, e, f, 4.9.12.b, c, d, e, f, DOC A.9

5.5.18 Ventilation Filter Testing Program (VFTP)

The VFTP ~~A program~~ shall ~~be~~ establish~~ed to implement~~ the ~~following~~ required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1]~~. INSERT 3

INSERT 3A

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a ~~penetration and system bypass~~ < [0.05]% when tested in accordance with the standard and ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989]~~ at the system flowrate specified below ~~[± 10%]~~.

4.7.5.1.c.2, 4.7.5.1.c.4, 4.7.5.1.f, 4.7.6.1.b.3, 4.7.6.1.b.5, 4.7.6.1.e, 4.9.12.b.3, 4.9.12.b.5, 4.9.12.e

ESF Ventilation System	Flowrate
[ ]	[ ]

INSERT 4

INSERT 4A

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a ~~penetration and system bypass~~ < [0.05]% when tested in accordance with the standard and ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989]~~ at the system flowrate specified below ~~[± 10%]~~.

4.7.5.1.c.1, 4.7.6.1.c.4, 4.7.5.1.g, 4.7.6.1.b.2, 4.7.6.1.b.5, 4.7.6.1.f, 4.9.12.b.2, 4.9.12.b.5, 4.9.12.f

ESF Ventilation System	Flowrate
[ ]	[ ]

INSERT 5

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained ~~as described in [Regulatory Guide 1.52, Revision 2]~~, shows the methyl iodide penetration less than or equal to the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86 °F) and the relative humidity (RH) specified below ~~:~~.

4.7.5.1.c.3, 4.7.5.1.d, 4.7.6.1.b.4, 4.7.6.1.c.1, 4.9.12.b.4, 4.9.12.c

INSERT 5A

15

INSERT 3

Tests described in Specifications 5.5.9.a and 5.5.9.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability.

Tests described in Specification 5.5.9.c shall be performed once per 24 months; after 720 hours of adsorber operation; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the charcoal adsorber capability.

Tests described in Specification 5.5.9.d shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

15

INSERT 3A

removal efficiency of $\geq$ 99% of the dioctyl phthalate (DOP)

15

INSERT 4

ESF Ventilation System	ANSI Standard	Flowrate (cfm)
CREV System	N510-1975	$\geq$ 5,400 and $\leq$ 6,600
ESF Ventilation System	N510-1980	$\geq$ 22,500 and $\leq$ 27,500
FHAEV System	N510-1980	$\geq$ 27,000 and $\leq$ 33,000

15

INSERT 4A

removal efficiency of $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas

15

INSERT 5

ESF Ventilation System	ANSI Standard	Flowrate (cfm)
CREV System	N510-1975	≥ 5,400 and ≤ 6,600
ESF Ventilation System	N510-1980	≥ 22,500 and ≤ 27,500
FHAEV System	N510-1980	≥ 27,000 and ≤ 33,000

1

INSERT 5A

from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers

CTS

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4

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (continued)

11 15

4.7.5.1.c.3, 4.7.5.1.d, 4.7.6.1.b.1, 4.7.6.1.c.1, 4.9.12.b.4, 4.9.12.c

ESF Ventilation System	Penetration	RH	Face Velocity (fps)
[ ]	[See Reviewer's Note]	[See Reviewer's Note]	[See Reviewer's Note]

INSERT 6

9

REVIEWER'S NOTE

The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.

ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30 °C (86 °F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.

Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]

When ASTM D3803-1989 is used with 30 °C (86 °F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:

Safety factor ≥ 2 for systems with or without humidity control.

Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst-case design-basis conditions.

If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.

*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.

4.7.5.1.e.1, 4.7.6.1.d.1, 4.9.12.d.1

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, ~~the prefilters,~~ and the charcoal adsorbers is less than the value specified below when tested ~~in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989~~ at the system flowrate specified below ~~(± 10%)~~.

15 1

1

3

15

INSERT 6

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	1	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

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(4)

5.5 Programs and Manuals (11)

4.7.5.1.e.1, 4.7.6.1.d.1, 4.9.12.d.1

5.5.9 Ventilation Filter Testing Program (continued)

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

INSERT 7 (15)

[e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989]. (1)

ESF Ventilation System	Wattage]
[ ]	[ ]

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.~~ (15)

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program (4)

DOC A.11, 3.11.1, 3.11.2.1, 3.11.2.2

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in unprotected outdoor temporary liquid storage tanks. ~~The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].~~ (1) (18)

The program shall include:

3.11.2.1, 4.11.2.1

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); (1) (11) (3)

3.11.2.2, 4.11.2.2

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank ~~and fed into the offgas treatment system~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and (11) (1) (1) (3)

3.11.1, 4.11.1

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor temporary liquid ~~radwaste~~ storage tanks that are not surrounded by liners, dikes, (11) (19)

15 **INSERT 7**

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	6	≥ 5,400 and ≤ 6,600
ESF Ventilation System	6	≥ 22,500 and ≤ 27,500
FHAEV System	6	≥ 27,000 and ≤ 33,000

4

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued) 11

10

3.11.1.4
4.11.1

or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. 11

DOC A.11

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies. 11

5.5.13 Diesel Fuel Oil Testing Program 11 4

DOC A.10

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following: 16

4.8.1.1.2.c, 4.8.1.1.2.c.1)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has: within limits

or a specific gravity

4.8.1.1.2.c.2)

1. An API gravity ~~or an absolute specific gravity~~ within limits, ~~i~~ 3 16

4.8.1.1.2.c.1).a), 4.8.1.1.2.c.1).b).

2. A flash point and kinematic or Saybolt viscosity within limits ~~for ASTM 2D fuel oil~~, ~~i~~ and INSERT B 3 16

4.8.1.1.2.c.3)

3. A clear and bright appearance with proper color, ~~i~~ 3

Specification 5.5.11 5

4.8.1.1.2.c.4)

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in ~~a~~ a. above, are within limits ~~for ASTM 2D fuel oil~~, and ~~i~~ 3

4.8.1.1.2.d

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A~~-2 or A-3~~. 16

DOC A.10

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies. 11

5.5

16

INSERT 8

, if the gravity was not determined by comparison with the supplier's certification, a

CTS

5.5 Programs and Manuals

6.8.5 5.5.12 **Technical Specifications (TS) Bases Control Program** 4

This program provides a means for processing changes to the Bases of these Technical Specifications.

6.8.5.a a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

6.8.5.b b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or 3

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

6.8.5.c c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR. 6

6.8.5.d d. Proposed changes that meet the criteria of Specification 5.5.12.c above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). 11

DOC M.2 5.5.13 **Safety Function Determination Program (SFDP)** 4

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following: 2

1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected; 2 3

2. Provisions for ensuring the ~~plant~~ unit is maintained in a safe condition if a loss of function condition exists; 2 6

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and 2 3

CTS

5.5 Programs and Manuals

DOC M.2 5.5.13 Safety Function Determination Program (continued) 4 11

4 d. Other appropriate limitations and remedial or compensatory actions. 2

b. A loss of safety function exists when, assuming no concurrent single failure, ~~and assuming~~ no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and: 2 5

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, ~~or~~ 2 3 11

2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or 2

3. A required system redundant to the support system(s) for the supported systems ~~(a) and (b)~~ above is also inoperable. 2 5

INSERT 9

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system. 2

DOC A.7 5.5.14 Containment Leakage Rate Testing Program 4

~~[OPTION A]~~ 17

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~

~~b. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.~~

~~c. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests.~~

~~2. Air lock testing acceptance criteria are:~~

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INSERT 9

described in Specifications 5.5.13.b.1 and 5.5.13.b.2

CTS

5.5 Programs and Manuals

4 11

5.5.6 Containment Leakage Rate Testing Program (continued)

14

a) Overall air lock leakage rate is ≤[0.05 L_a] when tested at ≥ P_a.

b) For each door, leakage rate is ≤[0.01 L_a] when pressurized to [≥10 psig].

17

d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

[OPTION B]

4.6.1.2, 4.6.1.3.a

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as *modified by approved exemptions. This program shall be in accordance* with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions: 1 17

1. . . .] INSERT 10 1

LCO 3.6.1.2.a, LCO 3.6.13.b

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 45 psig. The containment design pressure is [50 psig]. 17 17 12

LCO 3.6.1.2.a

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25 % of containment air weight per day. 1

d. Leakage rate acceptance criteria are:

LCO 3.6.1.2.a, LCO 3.6.1.2.b, 3.6.1.2 Action

1. *Containment leakage rate acceptance criterion is* 1.0 L_a. *During the* first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests. 17

2. Air lock testing acceptance criteria ~~are~~ is: 17 1

LCO 3.6.1.3.b

~~a)~~ Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥P_a.

b) For each door, leakage rate is ≤[0.01 L_a] when pressurized to [≥10 psig]. 17

CTS

17

INSERT 10

Unit 1 4.6.1.2 Note 2, Unit 2 4.6.1.2 Note 1

1. The Type A testing Frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "at least once per 10 years based on acceptable performance history" is modified to be "at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in October 1992 (Unit 1) and May 1992 (Unit 2).

Unit 1 4.6.1.2 Note 1

2. (Unit 1 only) A one-time exception to the requirement to perform post-modification Type A testing is allowed for the steam generators and associated piping, as components of the containment barrier. For this case, ASME Section XI leak testing will be used to verify the leak tightness of the repaired or modified portions of the containment barrier. Entry into MODES 3 and 4 following the extended outage that commenced in 1997 may be made to perform this testing.

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5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

DOC A.7 14

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

~~f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~ 17

~~[OPTION A/B Combined]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C][Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 [,as modified by the following exemptions:~~

~~1. . . .]~~

~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, [45 psig]. The containment design pressure is [50 psig].~~

~~c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.~~

~~d. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and [$< 0.75\ L_a$ for Option A Type A tests][$\leq 0.75\ L_a$ for Option B Type A tests].~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is $\leq$ [$0.05\ L_a$] when tested at $\geq P_a$.~~

~~b) For each door, leakage rate is $\leq$ [$0.01\ L_a$] when pressurized to [$\geq$ 10 psig].~~

17

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

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DOC M.2

5.5.17 15 Battery Monitoring and Maintenance Program 4

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer including the following: 11 1 1

a. Actions to restore battery cells with float voltage < 2.13 V, and 1 1 3

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

1. The brackets are removed and the proper plant specific information/value is provided.

2. This Specification has been renumbered to be consistent with the ITS format and for clarity.

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

4. The bracketed ISTS 5.5.3, Post Accident Sampling, is not included in the CNP Units 1 and 2 ITS. The requirements for Post Accident Sampling have been deleted from the CTS in License Amendments 261 (Unit 1) and 244 (Unit 2) dated January 16, 2002. Subsequent programs have been renumbered, as necessary.

5. Editorial changes made for enhanced clarity or to be consistent with the Writer's Guide.

6. 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.

7. ISTS 5.5.6 provides requirements for the Pre-Stressed Concrete Containment Tendon Surveillance Program. There is no requirement for this program in the CTS. Not including this ISTS program in the CNP Units 1 and 2 ITS is consistent with the CNP Units 1 and 2 licensing bases.

8. ISTS 5.5.7 (ITS 5.5.5) provides requirements for the Reactor Coolant Pump Flywheel Inspection Program. The allowance to perform the inspection per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 has been deleted. This change is consistent with the CNP Units 1 and 2 licensing bases. The Surveillance Frequency has also been modified to be consistent with the CNP Units 1 and 2 licensing bases.

9. The Reviewer's Note has been deleted since it is not intended to be included in the ITS.

10. The Inservice Testing (IST) Program (ISTS 5.5.8) has been modified to state that the IST Program provides control for ASME Code Class 1, 2, and 3 "pumps and valves" in place of the current "components." 10 CFR 50.55a(f) provides the regulatory requirements for an IST Program. It specifies that ASME Code Class 1, 2, and 3 pumps and valves are the only components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program. It specifies that ASME Code Class 1, 2, and 3 components are covered by the ISI Program, and that pumps and valves are covered by the IST Program in 10 CFR 50.55a(f). The ISTS does not include ISI Program requirements as these requirements have been relocated to a plant specific document. Therefore, the components to which the IST Program applies (i.e., pumps and valves) have been added for clarity. In addition, the statement "The program shall include the following:" has been deleted because not all of the statements that follow are really part of the program requirements. Also, in the 1987 Addenda to the 1986 edition of ASME Boiler and Pressure Vessel Code, Section XI, the requirements for Inservice

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

Testing were removed and relocated to the ASME/ANSI OM Code. This change was endorsed in 10 CFR 50.55a. 10 CFR 50.55a(f) now addresses the requirements for inservice testing using the ASME/ANSI OM Code and 10 CFR 50.55a(g) addresses the requirements for inservice inspection using ASME Boiler and Pressure Vessel Code, Section XI. The ITS has been revised to incorporate the current ASME/ANSI OM Code requirements. In addition, the terms weekly, monthly, semiannually, and every 9 months are not used in the ASME/ANSI OM Code and have been deleted.

11. Typographical/grammatical error corrected.

12. ISTS 5.5.9 (ITS 5.5.7) provides the requirements for the Steam Generator (SG) Program. Consistent with the associated Reviewer's Note, the CNP Units 1 and 2 current licensing basis, reflected in CTS 4.4.5.1, 4.4.5.2, 4.4.5.3, and 4.4.5.4, for SG tube inspections are included in this program. The corresponding ISTS Reviewer's Note is deleted. The Reviewer's Note provides information for the NRC to identify acceptable methods to meet the requirements. The Reviewer's Note is not meant to be retained in the final version of the plant-specific submittal.

13. ISTS 5.5.10 (ITS 5.5.8) provides the requirements for the Secondary Water Chemistry Program. The program in the ISTS includes requirements to provide controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion. ITS 5.5.8 provides controls for monitoring secondary water chemistry only to inhibit SG tube degradation. This modification is consistent with the current requirements in License Condition 2.C.(7) (Unit 1) and 2.C.(3)(v) (Unit 2).

14. ISTS 5.5.10.c includes a requirement that the Secondary Water Chemistry Program identify process sampling points, "which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage." ITS 5.5.8.c only includes the requirement that the Secondary Chemistry Program identify process sampling points and does not provide any explicit monitoring points. This change is consistent with current Operating Licensing Conditions 2.C.(7).3 (Unit 1) and 2.C.(3)(v)3 (Unit 2).

15. ISTS 5.5.11 (ITS 5.5.9) provides requirements for the Ventilation Filter Testing Program. ITS 5.5.9 is revised to reflect the CNP Units 1 and 2 licensing bases. The 18 month Frequencies in the CTS have been changed to 24 months in the ITS.

16. The following changes have been made to ISTS 5.5.13 (ITS 5.5.11):

    a. Specific gravity has been added as an option to API gravity or absolute specific gravity consistent with the current licensing basis;

    b. Saybolt viscosity has been added as an option to kinematic viscosity and the viscosity check is only required if the gravity was not determined by comparison with the supplier's certification, consistent with current licensing basis;

    c. The type of fuel oil, Type 2D, has been deleted consistent with current licensing basis; and

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

d. The words "ASTM D-2276 Method A-2 or A-3" in ISTS 5.5.13.c (ITS 5.5.11.c) have been changed to "ASTM D-2276 Method A" in ITS 5.5.11.c to be consistent with current licensing basis.

17. ISTS 5.5.16 (ITS 5.5.14) provides requirements for the Containment Leakage Rate Testing Program. The requirements of the ISTS are revised to reflect the Containment Leakage Rate Testing Program requirements of CTS 3/4.6.1.2 and 3/4.6.1.3. The containment design pressure limit specified in ISTS 5.5.16.b was not included because it currently does not exist in the CTS, and because this limit does not provide any useful input to the Containment Leakage Rate Testing Program. The air lock door leakage test of ISTS 5.5.16.d.2.b) is not included because it is not required by the CTS. In addition, the statement in ISTS 5.5.16.f that "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J" has been deleted. This phrase is not consistent with the allowances in ISTS 5.5.16.a (ITS 5.5.14.a), which states that the "program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance- Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:" These exceptions stated in ITS 5.5.14.a are modifications to the testing Frequencies required by 10 CFR 50, Appendix J. In addition, there is no need to state any specific exception to any of the other requirements of the Specifications that discuss testing Frequencies, because the convention of application of requirements in the sections of ISTS 5.5 is that no other Specification requirements apply unless otherwise stated. For example, ISTS SR 3.0.2 does not apply to any of the ISTS 5.5 sections, unless specifically noted. Therefore, there is no need to include a statement that ITS SR 3.0.2 does not apply to the Frequencies of ITS 5.5.14.

18. The program details of the Explosive Gas and Storage Tank Radioactivity Monitoring Program are described in ISTS 5.5.12 (ITS 5.5.10) parts a, b, and c. Therefore, the sentence in the introductory paragraph that specifies a method to determine the explosive gas and storage tank radioactivity is not necessary.

19. Changes are made to ISTS 5.5.12.c (ITS 5.5.10.c) to be consistent with the first paragraph in ISTS 5.5.12 (ITS 5.5.10).

20. ISTS 5.5.11.d demonstrates that the pressure drop across the combined HEPA filters, prefilters, and charcoal adsorbers is less than the specified pressure drop when tested at the specified system flow rate. The referenced methods for performing the test, Regulatory Guide 1.52, Revision 2 and ASME N510-1989, do not provide the methods for performing this test. As a result, these test method references have been deleted in ITS 5.5.9.d.

21. The requirement of ISTS 5.5.7 (ITS 5.5.5) is currently located in an individual Specification in the CTS (CTS 4.4.10.1). Thus, CTS 4.0.2 (ITS SR 3.0.2) and CTS 4.0.3 (ITS SR 3.0.3) apply to the CTS Surveillance Frequency. To maintain consistency with the current licensing basis requirements, an allowance that ITS SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program Surveillance Frequency has been included in ITS 5.5.5. In addition, approved TSTF-421, which extends the Frequency to 20 years has not been adopted.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 6

ITS 5.6, Reporting Requirements

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

A.1

ITS 5.6

ITS

6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

6.8.5 Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the Technical Specification shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior Nuclear Regulatory Commission approval provided the changes do not require either of the following:

1. A changes in the Technical Specification incorporated in the license or

2. A change to the Updated Final Safety Analysis Report or Bases that requires Nuclear Regulatory Commission approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the Updated Final Safety Analysis Report.

d. Proposed changes that meet the criteria of Specification 6.8.5.b above shall be reviewed and approved by the Nuclear Regulatory Commission prior to implementation. Changes to the Bases implemented without prior Nuclear Regulatory Commission approval shall be provided to the Nuclear Regulatory Commission on a frequency consistent with 10 CFR 50.71(e).

See ITS 5.5

5.6 6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

5.6 6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted ~~to the Regional Administrator unless otherwise noted~~. In accordance with 10 CFR 50.4 A.2

~~STARTUP REPORT~~

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.~~

~~6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

L.1

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.9 STARTUP REPORT (Continued)

6.9.1.3 ~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~ L.1

ANNUAL REPORTS[1]

by April 30 (for Occupational Radiation Exposure Report) L.2

5.6.1, 5.6.7 6.9.1.4 Annual reports covering the activities of the unit ~~as~~ described below for the previous calendar year shall be submitted ~~prior to March 1~~ of each year. ~~The initial report shall be submitted prior to March 1 of the year following initial criticality.~~ A.3

6.9.1.5 Reports required on an annual basis shall include:

5.6.1 a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving annual exposures greater than 100 mrem according to work and job functions[2], e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing and refueling. Also included is a tabulation of the total person rem exposures for station, utility, and other personnel associated with each work and job function. The dose assignment to various duty functions may be estimates based on pocket dosimeter, electronic dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose received shall be assigned to specific major work functions.

5.6.7 b. The complete results of steam generator tube in-service inspections performed during the report period (reference Specification 4.4.5.5.b).

~~c. Documentation of all challenges to the pressurizer power operated relief valves (PORVs) or safety valves.~~ L.3

~~d. Information regarding any instances when the I-131 specific activity limit was exceeded.~~ L.4

5.6.1 Note [1] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

5.6.1 [2] This tabulation supplements the requirements of 20.2206 of 10 CFR Part 20.

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3]

by May 15 — L.2

5.6.2 6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~before May 1~~ of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

INSERT 1 — M.1

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT[3] — A.1

5.6.3 6.9.1.7 The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted within 90 days after January 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5.6.2 Note, 5.6.3 Note

[3] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station~~; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material for each unit.~~ — A.9

ITS

M.1 **INSERT 1**

5.6.2 The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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ITS 5.6

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

MONTHLY REACTOR OPERATING REPORT

5.6.4 6.9.1.8 Routine reports of operating statistics and shutdown experience, ~~including documentation of all challenges to the PORVs or safety valves,~~ shall be submitted on a monthly basis ~~to the U.S. Nuclear Regulatory Commission (Attn: Document Control Desk), Washington, D.C. 20555, with a copy to the Regional Office~~ no later than the 15th of each month following the calendar month covered by the report. L.3 A.4

5.6.5 CORE OPERATING LIMITS REPORT

5.6.5.a 6.9.1.9.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following: A.5

Reactor Core Safety Limits; SHUTDOWN MARGIN;

a. Moderator Temperature Coefficient Limits for Specification 3/4.1.1.4,

b. ~~Rod Drop Time Limits for Specification 3/4.1.3.3,~~

c. Shutdown Rod Insertion Limits for Specification 3/4.1.3.4,

d. Control Rod Insertion Limits for Specification 3/4.1.3.5, A.6

e. Axial Flux Difference for Specification 3/4.2.1,

f. Heat Flux Hot Channel Factor for Specification 3/4.2.2,

g. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3, and

h. Allowable Power Level for Specification 3/4.2.6.

RTS Instrumentation Overpressure ΔT and Overpower ΔT Allowable Value parameter values; RCS Pressure, Temperature, and Flow DNB Limits; and Boron Concentration.

5.6.5.b 6.9.1.9.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," ~~July 1985~~ (Westinghouse Proprietary). LA.1

b. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," ~~September 1974~~ (Westinghouse Proprietary).

c. WCAP-10216-P-A, ~~Revision 1A,~~ "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification," ~~February 1994~~ (Westinghouse Proprietary).

d. WCAP-10266-P-A ~~Rev. 2,~~ "The 1981 Version of Westinghouse Evaluation Mode Using BASH Code," ~~March 1987~~ (Westinghouse Proprietary).

e. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," ~~July 1991~~ (Westinghouse Proprietary).

WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Westinghouse Proprietary). A.6

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ITS 5.6

ITS

6.0 ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

5.6.5.c 6.9.1.9.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

5.6.5.d 6.9.1.9.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC ~~document control desk with copies to the Regional Administrator and Resident Inspector~~. A.4

~~SPECIAL REPORTS~~

~~6.9.2 Special reports shall be submitted to the attention of the document control desk - U.S. Nuclear Regulatory Commission (Washington, D.C. 20555), with copies to the Region III Administrator and the Resident Inspector at the Cook Nuclear Plant within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:~~

~~a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.~~

~~b. Seismic Monitoring Instrumentation Actuated, Specification 4.3.3.3.2.~~

~~c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.~~

~~d. High Specific Activity in RCS Coolant, Specification 3.4.8.~~

~~e. RCS Pressure Transient Mitigated By RHR Safety Valve or RCS Vent(s), Specification 3.4.9.3.~~

~~f. Moderator Temperature Coefficient, Specification 3.1.1.4.~~

~~g. Sealed Source Leakage in Excess of Limits, Specification 4.7.7.1.3.~~

~~h. ECCS Actuation, Specifications 3.5.2 and 3.5.3.~~

~~i. Violation of Safety Limit, Specification 6.7.1.~~

A.7

6.10 DELETED

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TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1. (See ITS 3.4.15)

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day. (See CTS 3/4.3.3.1)

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirements, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test. (See ITS 3.3.6)

ACTION 22A- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements: (See ITS 3.3.3)

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

5.6.6 2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. Technical Specification Section 3.0.3 is Not Applicable. (See ITS 3.3.3)

ACTION 22B- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements. (See CTS 3/4.3.3.1)

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.

4. Technical Specification Section 3.0.3 is Not Applicable.

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SURVEILLANCE REQUIREMENTS (continued)

5.6.7 4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include: A.8

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. A.10

ITS 5.6

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ITS

6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

6.8.5 Technical Specification Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the Technical Specification shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior Nuclear Regulatory Commission approval provided the changes do not require either of the following:

1. A change in the Technical Specification incorporated in the license or

2. A change to the Updated Final Safety Analysis Report of Bases that requires Nuclear Regulatory Commission approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the Updated Final Safety Analysis Report.

d. Proposed changes that meet the criteria of Specification 6.8.5.b above shall be reviewed and approved by the Nuclear Regulatory Commission prior to implementation. Changes to the Bases implemented without prior Nuclear Regulatory Commission approval shall be provided to the Nuclear Regulatory Commission on a frequency consistent with 10 CFR 50.71(e).

See ITS 5.5

5.6 6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

5.6 6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted ~~to the Regional Administrator unless otherwise noted~~ In accordance with 10 CFR 50.4 A.2

~~STARTUP REPORT~~

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.~~

~~6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

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6.0 ADMINISTRATIVE CONTROLS

STARTUP REPORT (Continued)

6.9.1.3 ~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~ L.1

ANNUAL REPORTS[1]

by April 30 (for Occupational Radiation Exposure Report) L.2

5.6.1, 5.6.7 6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted ~~prior to March 1~~ of each year. ~~The initial report shall be submitted prior to March 1 of the year following initial criticality.~~ A.3

6.9.1.5 Reports required on an annual basis shall include:

5.6.1 a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving annual exposures greater than 100 mrem according to work and job functions[2], e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing and refueling. Also included is a tabulation of the total person rem exposures for station, utility, and other personnel associated with each work and job function. The dose assignment to various duty functions may be estimates based on pocket dosimeter, electronic dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose received shall be assigned to specific major work functions.

5.6.7 b. The complete results of steam generator tube in-service inspections performed during the report period (reference Specification 4.4.5.5.b).

c. ~~Documentation of all challenges to the pressurizer power operated relief valves (PORVs) or safety valves.~~ L.3

d. ~~Information regarding any instances when the I-131 specific activity limit was exceeded.~~ L.4

5.6.1 Note [1] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

5.6.1 [2] This tabulation supplements the requirements of 20.2206 of 10 CFR Part 20.

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6.0 ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3]

by May 15 — L.2

5.6.2 6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~before May 1~~ of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

INSERT 1 — M.1

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT[3] — A.1

5.6.3 6.9.1.7 The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted within 90 days after January 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5.6.2 Note, 5.6.3 Note — [3] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; ~~however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material for each unit.~~ — A.9

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ITS

M.1 **INSERT 1**

5.6.2 The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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6.0 ADMINISTRATIVE CONTROLS

MONTHLY REACTOR OPERATING REPORT

5.6.4 6.9.1.8 Routine reports of operating statistics ~~and shutdown experience~~, ~~including documentation of all challenges to the PORVs or safety valves,~~ shall be submitted on a monthly basis ~~to the U.S. Nuclear Regulatory Commission (Attn: Document Control Desk), Washington, D.C. 20555, with a copy to the Regional Office no later than the 15th of each month following the calendar month covered by the report.~~ L.3 A.4

5.6.5 CORE OPERATING LIMITS REPORT

5.6.5.a 6.9.1.9.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

Reactor Core Safety Limits; SHUTDOWN MARGIN;

a. Moderator Temperature Coefficient Limits for Specification 3/4.1.1.4,

~~b. Rod Drop Time Limits for Specification 3/4.1.3.4,~~ A.5

c. Shutdown Rod Insertion Limits for Specification 3/4.1.3.5,

d. Control Rod Insertion Limits for Specification 3/4.1.3.6,

e. Axial Flux Difference for Specification 3/4.2.1, A.6

f. Heat Flux Hot Channel Factor for Specification 3/4.2.2,

g. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3, and

h. Allowable Power Level for Specification 3/4.2.6.

RTS Instrumentation Overpressure ΔT and Overpower ΔT Allowable Value parameter values; RCS Pressure, Temperature, and Flow DNB Limits; and Boron Concentration.

5.6.5.b 6.9.1.9.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," ~~July 1985~~ (Westinghouse Proprietary). LA.1

b. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," ~~September 1974~~ (Westinghouse Proprietary).

c. WCAP-10216-P-A, ~~Revision 1A,~~ "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification," ~~February 1994~~ (Westinghouse Proprietary).

d. WCAP-10266-P-A ~~Rev. 2,~~ "The 1981 Version of Westinghouse Evaluation Mode Using BASH Code," ~~March 1987~~ (Westinghouse Proprietary).

e. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," ~~July 1991~~ (Westinghouse Proprietary).

WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Westinghouse Proprietary). A.6

ITS 5.6

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ITS

6.0 ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

5.6.5.c 6.9.1.9.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

5.6.5.d 6.9.1.9.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC ~~document control desk with copies to the Regional Administrator and Resident Inspector~~. A.4

~~SPECIAL REPORTS~~

~~6.9.2 Special reports shall be submitted to the attention of the document control desk - U.S. Nuclear Regulatory Commission (Washington, D.C. 20555), with copies to the Region III Administrator and the Resident Inspector at the Cook Nuclear Plant within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:~~

~~a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.~~

~~b. Seismic Monitoring Instrumentation Actuated, Specification 4.3.3.3.2.~~

~~c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.~~ A.7

~~d. High Specific Activity in RCS Coolant, Specification 3.4.8.~~

~~e. RCS Pressure Transient Mitigated by RHR Safety Valve or RCS Vent(s), Specification 3.4.9.3.~~

~~f. Moderator Temperature Coefficient, Specification 3.1.1.4.~~

~~g. Sealed Source Leakage in Excess of Limits, Specification 4.7.7.1.3.~~

~~h. ECCS Actuation, Specifications 3.5.2 and 3.5.3.~~

~~i. Violation of Safety Limit, Specification 6.7.1.~~

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.3 INSTRUMENTATION

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1. See ITS 3.4.15

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day. See CTS 3/4.3.3.1

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test. See ITS 3.3.6

ACTION 22A- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements: See ITS 3.3.3

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

5.6.6 2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. Technical Specification Section 3.0.3 Not Applicable. See ITS 3.3.3

ACTION 22B- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status. See CTS 3/4.3.3.1

3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.

4. Technical Specification Section 3.0.3 Not Applicable.

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REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

See ITS 5.5

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

5.6.7 4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include: A.8

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. A.10

D.C. COOK - UNIT 2 3/4 4-11

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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.

A.2 CTS 6.9.1 requires, in addition to the requirements of 10 CFR, reports be submitted to the Regional Administrator. ITS 5.6 requires that the reports be submitted in accordance with 10 CFR 50.4. This changes the CTS by removing the explicit requirement to send reports to the Regional Administrator.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.4. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 6.9.1.4 regarding annual reports requires the initial report to be submitted prior to March 1 of the year following initial criticality. The ITS does not include such a statement. This changes the CTS by deleting a requirement for report submissions that have already occurred and will not be repeated.

This change is acceptable because the one time reporting requirement has already been met and no longer needs to be specified. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 6.9.1.8 requires the Monthly Reactor Operating Report be submitted to the U.S. Nuclear Regulatory Commission with a copy to the Regional Office. CTS 6.9.1.9.4 requires the CORE OPERATING LIMITS REPORT (COLR) to be provided to the NRC document control desk with copies to the Regional Administrator and Resident Inspector. ITS 5.6.4 requires the Monthly Operating Report to be submitted and ITS 5.6.5.d requires the COLR to be provided to the NRC. This changes the CTS by removing the specifics regarding distribution of the reports to the NRC.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the requirements deleted from the Technical Specifications are already required by 10 CFR 50.4. This change is designated as administrative because it does not result in technical changes to the CTS.

A.5 CTS 6.9.1.9.1 requires, in part, that core operating limits be established and documented in the COLR for the rod drop time limits in CTS 3/4.1.3.3. ITS 5.6.5.a does not include a reference to rod drop time limits. This changes

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DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

the CTS eliminating the reference to rod drop time limits being core operating limits that are included in the COLR.

Rod drop time limits are included in the CTS and the ITS, not the COLR. The information that CTS 3/4.1.3.3 is referring to in the COLR is the definition of what constitutes the full withdrawn position for the purposes of performing the rod drop time Surveillance. This information is not a core operating limit and is therefore not included in the list of individual Specifications that address core operating limits in ITS 5.6.5. This change is acceptable because the information that was moved to the COLR and is referenced in CTS 3/4.1.3.3 (i.e., what constitutes the full withdrawn position) remains in the COLR. This change is designated as administrative because it does not result in technical changes to the CTS.

A.6 CTS 6.9.1.9.1 contains a list of the core operating limits established and documented in the COLR and CTS 6.9.1.9.2 contains a list of the locations for the analytical methods used to determine the core operating limits. ITS 5.6.5.a includes additional core operating limits established and documented in the COLR. These are Reactor Core Safety Limits; SHUTDOWN MARGIN; Reactor Trip System Instrumentation Functions 6 and 7 (Overtemperature ΔT and Overpower ΔT; respectively) Allowable Value parameter values; RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits; and Boron Concentration. These limits had previously been addressed in other parts of the CTS, but are being moved to the COLR in the ITS, and because of this are listed in ITS 5.6.5.a. ITS 5.6.5.b.6 includes the document describing the analytical methods for the Overtemperature ΔT and Overpower ΔT Allowable Value parameter values. This changes the CTS by adding core operating limits established and documented in the COLR (and applicable methodology) because they are being moved there as part of changes to other parts of the CTS. Technical aspects of the changes are addressed in the Discussion of Changes for the respective individual ITS Specifications.

This change is acceptable because it administratively documents changes made to other parts of the CTS and the COLR. This change is designated as administrative because it does not result in technical changes to the CTS.

A.7 CTS 6.9.2 requires special reports be submitted to the NRC and lists the CTS Specifications that require special reports to be submitted. The ITS does not require these special reports to be prepared and submitted. This changes the CTS by deleting the references to the CTS Specifications requiring special reports. Justification for disposition of each of the special report requirements is addressed by the Discussion of Changes for the respective ITS or CTS Specification.

The purpose of CTS 6.9.2 is to identify the Specifications that require special reports to be submitted. This change is acceptable because the special reports are no longer required by the respective Specifications. Justification for disposition of each of the special report requirements is addressed by the Discussion of Changes for the respective ITS or CTS Specification. This change is designated as administrative because it does not result in technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

A.8 CTS 4.4.5.5.b requires the complete results of the steam generator tube inservice inspection to be included in the Annual Operating Report. ITS 5.6.7 requires these same results to be submitted on an annual basis (i.e., prior to March 1 for the inspection that was completed in the previous calendar year). This changes the CTS by eliminating the requirement to include the steam generator tube inservice inspection results in the Annual Operating Report.

The purpose of CTS 4.4.5.5.b is to ensure the results of the steam generator tube inservice inspection are provided to the NRC. It is not necessary to specify the report that will include the results. This change is acceptable because the steam generator tube inservice inspection results will still be required to be provided to the NRC at the same Frequency as in the CTS. This change is designated as administrative because it does not result in technical changes to the CTS.

A.9 CTS 6.9.1.6 and 6.9.1.7 Footnote 3 states that, for these reports, the submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material for each unit. ITS 5.6.2 and 5.6.3 does not include the portion of the statement concerning units with separate radwaste systems. This changes the CTS by deleting the reference to units with separate radwaste systems.

This change is acceptable because CNP Units 1 and 2 share a radwaste system; they do not have separate radwaste systems. This change is designated as administrative because it does not result in technical changes to the CTS.

A.10 CTS 4.4.5.5.c requires a prompt notification to the NRC pursuant to CTS 6.9.1 prior to resumption of plant operation and a followup written report if the results of the steam generator tube inspection fall into the Category C-3. ITS 5.6.7.c requires Category C-3 results to be reported to the NRC in accordance with 10 CFR 50.72 and a Licensee Event Report to be submitted in accordance with 10 CFR 50.73. This changes the CTS by explicitly referencing the applicable Regulations that require the report.

The purpose of CTS 4.4.5.5.c is to ensure NRC prompt notification and followup written reporting if an inspection result falls into Category C-3. 10 CFR 50.72 governs prompt phone notifications and 10 CFR 50.73 governs written reports. These changes are acceptable because they are consistent with the current manner in which the CTS 4.4.5.5.c notification and reporting are performed. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 The second paragraph of ITS 5.6.2 includes details required to be included in the Annual Radiological Environmental Operating Report. CTS 6.9.1.6 does not contain this level of detail. This changes the CTS by requiring additional detail to be included in the Annual Radiological Environmental Operating Report.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

The purpose of the second paragraph of ITS 5.6.2 is to specify details to be included in the Annual Radiological Environmental Operating Report. This change is acceptable because the content requirements are consistent with the objectives outlined in the Offsite Dose Calculation Manual. This change is designated more restrictive because it adds new reporting requirements to the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.9.1.9.2 specifies the revision numbers and dates of the referenced methodologies used for the development of the COLR. ITS 5.6.5.b does not contain this level of detail. This changes the CTS by moving the specific methodology references for revisions and dates to the COLR.

The removal of these details, which are related to meeting Technical Specifications requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the references for the COLR and only NRC-approved methodologies may be used. The methodologies used to develop the parameters in the COLR have obtained prior approval by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "CORE OPERATING LIMITS REPORT". ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analyses are met and that only NRC-approved methodologies are used. This change is designated as a less restrictive removal of detail change because information relating to the methodology used to develop cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.9.1.1, CTS 6.9.1.2, and CTS 6.9.1.3 contain requirements for submitting a report of plant startup and power escalation testing following receipt of an operating license; amendments to the license involving planned increase in power level; installation of fuel that has a different design or has been manufactured by a different fuel supplier; and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. The ITS does not contain such reporting requirements. This changes the CTS by deleting the requirements of CTS 6.9.1.1, CTS 6.9.1.2, and CTS 6.9.1.3.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

The purpose of CTS 6.9.1.1, CTS 6.9.1.2 and CTS 6.9.1.3, is to provide a summary of plant startup and power escalation testing following the four specified conditions as verification that the unit operated as expected. This change is acceptable because the regulations provide adequate reporting requirements. If there were any unit conditions outside the expected parameters during unit startup, they would be reported to the NRC if they met the reporting requirements in the regulations. Otherwise, the reports would document that the unit operated as expected and already approved by the NRC, as required by regulations. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L.2 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.9.1.4 requires annual reports described in CTS 6.9.1.5, which include the Occupational Radiation Exposure Report, to be submitted prior to March 1 of each year. CTS 6.9.1.6 requires the Annual Radiological Environmental Operating Report to be submitted before May 1 of each year. ITS 5.6.1 requires the Occupational Radiation Exposure Report to be submitted by April 30 of each year. ITS 5.6.2 requires the Annual Radiological Environmental Operating Report to be submitted by May 15 of each year. This changes the CTS by allowing an additional time to submit these reports each year.

The purpose of the due date for submitting the Occupational Radiation Exposure Report and Annual Radiological Environmental Operating Report is to ensure that the reports are provided in a reasonable period of time to the NRC for review. This change is acceptable because the reports are still required to be submitted in a reasonable time frame. Given that the reports are still required to be provided to the NRC on or before April 30 or May 15, respectively, and cover the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner for the interval between March 1 and April 30, and May 1 and May 15, respectively. Additionally, there is no requirement for the NRC to approve the reports. This change is designated as less restrictive because it allows more time to prepare and submit the reports to the NRC.

L.3 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.9.1.5.c and 6.9.1.8 require annual and monthly reporting of all challenges to the Reactor Coolant System pressurizer operated relief valves (PORVs) or safety valves. ITS 5.6 does not include these reporting requirements. This changes the CTS by deleting the requirement to include documentation of all challenges to the Reactor Coolant System PORVs or safety valves in the annual and monthly reports.

The purpose of the annual and monthly reporting requirements is to ensure the NRC receives appropriate routine reports of operating statistics and shutdown experience. This change is acceptable because the regulations provide adequate details of reporting requirements, and the reporting of these challenges does not affect continued plant operation. The change deletes the requirement to include documentation of all challenges to the Reactor Coolant System PORVs or safety valves in the annual and monthly reports. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

L.4 *(Category 8 – Deletion of Reporting Requirements)* CTS 6.9.1.5.d requires annual reporting of information regarding any instances when the I-131 specific activity limit for the primary coolant is exceeded. ITS 5.6 does not contain any requirements for such a report. This changes the CTS by not including the requirements for the annual reporting of instances when the Technical Specification I-131 specific activity limit for the primary coolant is exceeded.

The purpose of CTS 6.9.1.5.d is to specify the requirements for submitting information regarding any instances when the Technical Specification I-131 specific activity limit for the primary coolant is exceeded in an annual report. This change is acceptable because the regulations provide adequate details of reporting requirements, and the reporting of exceeding the I-131 limit does not affect continued plant operation. Operations or conditions prohibited by the plant's Technical Specifications are required to be reported in accordance with 10 CFR 50.73. Subsequent reports would be provided if necessary, without requiring a specific annual report. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

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

CTS

5.0 ADMINISTRATIVE CONTROLS

6.9 5.6 Reporting Requirements

6.9.1 The following reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1.4, 6.9.1.5 including Footnotes 1 and 2

5.6.1 Occupational Radiation Exposure Report

-REVIEWER'S NOTE- (1)
[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.] (2)

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year. ~~[The initial report shall be submitted by April 30 of the year following the initial criticality.]~~ (1) (2)

6.9.1.6 including Footnote 3

5.6.2 Annual Radiological Environmental Operating Report

-REVIEWER'S NOTE- (1)
[A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.] (2)

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all

CTS

5.6 Reporting Requirements

6.9.1.6 including footnote 3

5.6.2 Annual Radiological Environmental Operating Report (continued) ①

environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~(~~in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979~~)~~. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. ②

6.9.1.7 including footnote 3

5.6.3 Radioactive Effluent Release Report

- REVIEWER'S NOTE - ①

~~[~~A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station~~; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit~~.~~]~~ ②

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted ~~prior to May 1 of each year in accordance with 10 CFR 50.36a~~. INSERT 1 ③ The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

6.9.1.8

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

6.9.1.9.1

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~[The individual specifications that address core operating limits must be referenced here.]~~

INSERT 2 ②

CTS

3 **INSERT 1**

6.9.1.7 within 90 days of January 1 of each year

2 **INSERT 2**

6.9.1.9.1
1. SL 2.1.1, "Reactor Core Safety Limits";
2. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
3. LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
4. LCO 3.1.5, "Shutdown Bank Insertion Limits";
5. LCO 3.1.6, "Control Bank Insertion Limits";
6. LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
7. LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
8. LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
9. LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," Functions 6 and 7 (Overtemperature ΔT and Overpower ΔT, respectively) Allowable Value parameter values;
10. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
11. LCO 3.9.1, "Boron Concentration."

CTS

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued) ①

6.9.1.9.2 b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~[Identify the Topical Report(s) by number and title or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).]~~ INSERT 3 ②

6.9.1.9.3 c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.9.4 d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~

~~b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:~~

~~[Identify the NRC staff approval document by date.]~~

~~c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.~~ ④

5.6

CTS

2 **INSERT 3**

6.9.1.9.2

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Westinghouse Proprietary);
2. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," (Westinghouse Proprietary);
3. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification," (Westinghouse Proprietary);
4. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Mode Using BASH Code," (Westinghouse Proprietary);
5. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," (Westinghouse Proprietary); and
6. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Westinghouse Proprietary).

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CTS

5.6 Reporting Requirements

5.6.6 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

- REVIEWER'S NOTE -

The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

6. LTOP arming temperature limit development methodology.

7. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

8. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase $RT_{NDT} + 2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.

4

CTS

5.6 Reporting Requirements

Table 3.3-6 Action 22A.2

5.6.~~7~~ 6 Post Accident Monitoring Report (4) (2)

When a report is required by Condition B or ~~G~~ H of LCO 3.3.~~3~~ 6 "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

~~5.6.8 [Tendon Surveillance Report~~

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~ (2)

6.9.1.4, 6.9.1.5.6, 4.4.5.5

5.6.~~9~~ 7 ~~[~~Steam Generator Tube Inspection Report~~]~~ (4) (2)

INSEERT 4

~~- REVIEWER'S NOTES -~~

~~1. Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~

~~2. These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~ (5)

CTS

5 **INSERT 4**

6.9.1.5.b, 4.4.5.5

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the NRC.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the NRC prior to March 1 for the inspection that was completed in the previous calendar year. This report shall include:

1. Number and extent of tubes inspected;
2. Location and percent of wall-thickness penetration for each indication of an imperfection; and
3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the NRC in accordance with 10 CFR 50.72. A Licensee Event Report shall be submitted in accordance with 10 CFR 50.73 and shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

JUSTIFICATION FOR DEVIATIONS
ITS 5.6, REPORTING REQUIREMENTS

1. Grammatical/typographical error corrected.

2. The brackets are removed and the proper plant specific information/value is provided.

3. ISTS 5.6.3 requires submittal of the Radioactive Effluent Release Report prior to May 1 of each year in accordance with 10 CFR 50.36a. The phrase "in accordance with 10 CFR 50.36a" is duplicative of the requirements in 10 CFR 50.36a, and is therefore not required to be in the Technical Specifications. 10 CFR 50.36a states that the report must be submitted within one year of the previous report. The existing CNP CTS submittal date for this report is not May 1 of each year. Since Technical Specifications cannot supersede the requirements of 10 CFR 50, implementation of this change would require NRC approval of an exemption request in accordance with 10 CFR 50.12. This is considered to be outside the scope of the ITS conversion. Therefore, the submittal date for this report is revised in ITS 5.6.3 to reflect the CNP CTS (i.e., within 90 days of January 1 of each year).

4. ISTS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," is not adopted in the ITS. CTS Figures 3.4-2 and 3.4-3, which provide Reactor Coolant System heatup and cooldown limitations, respectively, were adopted in ITS 3.4.3, "RCS Pressure and Temperature (P/T) Limits." Subsequent Specifications are renumbered accordingly. In addition, since the PTLR is not included in the ITS, approved TSTF-419, which modifies ISTS 5.6.6, is not incorporated.

5. The ISTS Reviewer's Notes have been deleted since they were not intended to be included in the ITS. The requirements for the Steam Generator Tube Inspection Report have been included consistent with these ISTS Reviewer's Notes and the CNP CTS requirements.

6. Changes made to reflect those changes made to ITS 3.3.3, "Post Accident Monitoring (PAM) Instrumentation."

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 7

ITS 5.7, High Radiation Area

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

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure. See CTS 6.0

6.12 HIGH RADIATION AREA

5.7 6.12.1 Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 10 CFR 20.1601(a) and (b), each high radiation area in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mrem but less than or equal to
5.7.1 1000 mrem in 1 hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made aware of it.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Plant Radiation Protection Manager in the Radiation Work Permit. LA.1

5.7.2 6.12.2 The requirements of 6.12.1 shall also apply to each high radiation area in which the radiation level at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates is greater than 1000 mrem in 1 hour. When possible, locked doors shall be provided to prevent unauthorized entry into such areas, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the Plant Radiation Protection Manager. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas. In the event that it is not possible or practicable to provide locked doors due to area size or configuration, the area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. LA.1

5.7.1 * Health Physics (Radiation Protection) personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure. See CTS 6.0

~~6.12 HIGH RADIATION AREA~~

5.7 6.12.1 Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 10 CFR 20.1601(a) and (b), each high radiation area in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mrem but less than or equal to 1000 mrem in 1 hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

5.7.1

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made aware of it.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~Plant~~ Radiation Protection Manager in the Radiation Work Permit. LA.1

5.7.2 6.12.2 The requirements of 6.12.1 shall also apply to each high radiation area in which the radiation level at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates is greater than 1000 mrem in 1 hour. When possible, locked doors shall be provided to prevent unauthorized entry into such areas, and the keys shall be maintained under the administrative control of the Shift Manager on duty ~~and~~/or the ~~Plant~~ Radiation Protection Manager. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas. In the event that it is not possible or practicable to provide locked doors due to area size or configuration, the area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. LA.1

5.7.1 * Health Physics (Radiation Protection) personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP 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. 2, "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

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.12.1.c uses the title "Plant Radiation Protection Manager" and CTS 6.12.2 uses the titles "Shift Manager" and "Plant Radiation Protection Manager." ITS 5.7.1.c uses the generic title "radiation protection manager" and ITS 5.7.2 uses the generic titles "shift manager" and "radiation protection manager." This changes the CTS by moving the specific CNP organizational titles to the UFSAR and replacing them with generic titles.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific CNP organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the radiation protection manager and the shift manager are still retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

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DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

LESS RESTRICTIVE CHANGES

None

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

CTS

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

1

6.12.1 As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that

INSERT 1

1

CTS

1 INSERT 1

6.12.1 including Footnote * 5.7.1 Each high radiation area in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mrem but less than or equal to 1000 mrem in 1 hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiological Work Permit (RWP). Radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by at least one of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area;

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of it; or

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by a radiation protection manager in the RWP.

6.12.2 5.7.2 In addition to the requirements of Specification 5.7.1 above, for each high radiation area in which the radiation level at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates is greater than 1000 mrem in 1 hour, locked doors shall be provided, when possible, to prevent unauthorized entry into such areas and the keys shall be maintained under administrative control of the shift manager on duty or a radiation protection manager. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas. In the event that it is not possible or practicable to provide locked doors due to area size or configuration, the area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

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5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source of from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

1

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designees, and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displaces radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and

1

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

1

JUSTIFICATION FOR DEVIATIONS
ITS 5.7, HIGH RADIATION AREA

1. ISTS 5.7 provides requirements for High Radiation Areas. The brackets are removed and the proper plant specific information/value is provided. ITS 5.5.7 is revised to reflect the CNP current licensing basis and high radiation area controls The change is consistent with the requirements in CTS 6.12.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.7, HIGH RADIATION AREA

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 8

Relocated/Deleted Current Technical Specifications (CTS)

CTS 6.0, Administrative Controls

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

6.0 ADMINISTRATIVE CONTROLS

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Director, who must be qualified as specified in Section 6.2.2.g.

See ITS 5.2 and ITS 5.3

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

LA.1

6.5 DELETED

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6.0 ADMINISTRATIVE CONTROLS

~~6.6 REPORTABLE EVENT ACTION~~

~~6.6.1 The following actions shall be taken for REPORTABLE EVENTS:~~

~~a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73.~~

A.1

~~b. Each REPORTABLE EVENT shall be reviewed by the PORC, and the results of this review shall be submitted to the NSRB and the Site Vice President.~~

LA.2

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Chairman of the NSRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems or structures; and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the NSRB and the Senior Vice President – Nuclear Operations within 14 days of the violation.

d. Operation of the unit shall not be resumed until authorized by the Commission.

See ITS Chapter 2.0

6.0 ADMINISTRATIVE CONTROLS

~~6.11 RADIATION PROTECTION PROGRAM~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

LA.3

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 10 CFR 20.1601(a) and (b), each high radiation area in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mrem but less than or equal to 1000 mrem in 1 hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

See ITS 5.7

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made aware of it.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Plant Radiation Protection Manager in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1 shall also apply to each high radiation area in which the radiation level at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates is greater than 1000 mrem in 1 hour. When possible, locked doors shall be provided to prevent unauthorized entry into such areas, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the Plant Radiation Protection Manager. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas. In the event that it is not possible or practicable to provide locked doors due to area size or configuration, the area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

* Health Physics (Radiation Protection) personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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6.0 ADMINISTRATIVE CONTROLS

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

LA.4

b. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

See ITS 5.5

2. A determination that the change will maintain the level of radioactive effluent control pursuant to 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

DEFINITIONS

PROCESS CONTROL PROGRAM (PCP)

1.28 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

LA.4

1.29 Deleted.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

See ITS 5.5

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

See ITS Chapter 1.0

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

6.0 ADMINISTRATIVE CONTROLS

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Director, who must be qualified as specified in Section 6.2.2.g.

See ITS 5.2 and ITS 5.3

~~**6.4 TRAINING**~~

~~6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.~~

LA.1

6.5 DELETED

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6.0 ADMINISTRATIVE CONTROLS

~~6.6 REPORTABLE EVENT ACTION~~

~~6.6.1 The following actions shall be taken for REPORTABLE EVENTS:~~

~~a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73.~~

~~b. Each REPORTABLE EVENT shall be reviewed by the PORC, and the results of this review shall be submitted to the NSRB and the Site Vice President.~~

A.1

LA.2

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Chairman of the NSRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the PORC. The report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems or structures; and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the NSRB and the Senior Vice President – Nuclear Operations within 14 days of the violation.

d. Operation of the unit shall not be resumed until authorized by the Commission.

See ITS Chapter 2.0

6.0 ADMINISTRATIVE CONTROLS

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

LA.3

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 10 CFR 20.1601(a) and (b), each high radiation area in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mrem but less than or equal to 1000 mrem in 1 hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

See ITS 5.7

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made aware of it.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Plant Radiation Protection Manager in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1 shall also apply to each high radiation area in which the radiation level at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates is greater than 1000 mrem in 1 hour. When possible, locked doors shall be provided to prevent unauthorized entry into such areas, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the Plant Radiation Protection Manager. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas. In the event that it is not possible or practicable to provide locked doors due to area size or configuration, the area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

* Health Physics (Radiation Protection) personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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6.0 ADMINISTRATIVE CONTROLS

~~6.13 PROCESS CONTROL PROGRAM (PCP)~~

~~6.13.1 Changes to the PCP:~~

~~a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n. This documentation shall contain:~~

~~1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and~~

~~2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~

~~b. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.~~

LA.4

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program Description, Appendix C, Section 6.10.2.n. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the level of radioactive effluent control pursuant to 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

See ITS 5.5

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DEFINITIONS

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

See ITS Chapter 1.0

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.26 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

SOURCE CHECK

1.27 A SOURCE CHECK shall be the qualitative assessment of Channel response when the Channel sensor is exposed to a radioactive source.

~~PROCESS CONTROL PROGRAM (PCP)~~

~~1.28 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

LA.4

DISCUSSION OF CHANGES
CTS 6.0, ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CHANGES

A.1 CTS 6.6.1, Reportable Event Action, including CTS 6.6.1.a, specifies, in the case of a Reportable Event, that the Commission be notified and a report be submitted pursuant to the requirements of 10 CFR 50.73. The requirements of CTS 6.6.1 and 6.6.1.a are not included in the ITS. This changes the CTS by removing the requirements for Reportable Event Action.

This change is acceptable because the requirements of CTS 6.6.1 and 6.6.1.a are contained in 10 CFR 50.72 and 10 CFR 50.73. Therefore, there is no need to repeat these requirements in the Technical Specifications. Since the CNP Units 1 and 2 Operating Licenses require compliance with 10 CFR 50, the 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

LA.1 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 6.4 states that a retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55. ITS Chapter 5.0 does not require such a program. This changes the CTS by moving the requirements for the retraining and replacement training program to the UFSAR.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. These training provisions are adequately addressed by other proposed ITS Chapter 5.0 provisions and by regulations. ITS 5.3, "Unit Staff Qualifications," provides requirements to ensure adequate, competent staff in accordance with ANSI N 18.1-1971 and Regulatory Guide 1.8, 1975. ITS 5.2 details organization requirements. ITS 5.2.2.a, 5.2.2.b, and 10 CFR 50.54 state minimum shift crew requirements. Training and requalification of NRC licensed positions is contained in 10 CFR 50.55. Placement of training requirements in the UFSAR will ensure that training programs are properly maintained in accordance with CNP Unit 1 and 2 commitments and applicable regulations. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly

DISCUSSION OF CHANGES
CTS 6.0, ADMINISTRATIVE CONTROLS

evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications.

LA.2 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 6.6.1.b states that each reportable event shall be reviewed by the PORC, and the results of this review shall be submitted to the NSRB and the Site Vice President. The ITS does not include this requirement. This changes the CTS by moving these details of Reportable Event Action to the Quality Assurance Program Description (QAPD).

The removal of these 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. Given that these reviews and submittal of results are required following the event without a specified completion time, the proposed relocated requirements are not necessary to assure operation of the facility in a safe manner. As such, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Also, this change is acceptable because these types of procedural details will be adequately controlled in the QAPD. Any changes to the QAPD are made under 10 CFR 50.54(a), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because requirements are being removed from the Technical Specifications.

LA.3 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 6.11 provides requirements for the Radiation Protection Program. The ITS does not include these requirements. This changes the CTS by moving the requirements for the Radiation Protection Program to the UFSAR.

The removal of these 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 Radiation Protection Program requires procedures to be prepared for personnel radiation protection consistent with 10 CFR 20. These procedures are for nuclear plant personnel and have no impact on nuclear safety or the health and safety of the public. Requirements to have procedures to implement 10 CFR 20 are contained in 10 CFR 20.1101(b). Periodic review of these procedures is addressed in 10 CFR 20.1101(c). Since the CNP Units 1 and 2 Operating Licenses require compliance with 10 CFR 20, there is no need to repeat the requirements in the ITS. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

DISCUSSION OF CHANGES
CTS 6.0, ADMINISTRATIVE CONTROLS

LA.4 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS Definition 1.28 contains the definition for the Process Control Program (PCP). CTS 6.13.1 describe the process for control of changes to the PCP. The ITS does not include these requirements. This changes the CTS by moving the requirements of the PCP to the UFSAR.

The removal of these 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 PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Compliance with these regulations is required by the CNP Units 1 and 2 Operating Licenses, and procedures are the method to ensure compliance with the program. Regulations provide an adequate level of control for the affected requirements and inclusion of this requirement in the Technical Specifications is not necessary. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of requirements because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 6.0, ADMINISTRATIVE CONTROLS

There are no specific NSHC discussions for this Specification.

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