FPL.

April 3, 2015

L-2015-022

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Re: Turkey Point Nuclear Plant, Units 3 and 4
Docket Nos. 50-250 and 50-251

Supplement to License Amendment Request 229, Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program

References:

1. Florida Power & Light Company letter L-2014-033, "License Amendment Request No. 229, Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," April 9, 2014 [ML 14105A042]

2. NRC letter "Turkey Point Nuclear Generating Unit Nos. 3 and 4 - Request for Additional Information on License Amendment Request to Revise Technical Specifications to Implement TSTF-425, Revision 3, 'Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specifications Task Force (RITSTF) Initiative 5B' (TAC Nos. MF3931 and MF3932)," August 7, 2014 [ML 14212A713]

3. Florida Power & Light Company letter L-2014-266 "Response to NRC Technical Specifications Branch Request for Additional Information Regarding License Amendment Request No. LAR-229, 'Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program'," August 29, 2014 [ML 14252A228]

In Reference 1 and supplemented by Reference 3, Florida Power & Light Company (FPL) submitted a request for an amendment to the Technical Specifications (TS) for Turkey Point Units 3 and 4. The proposed amendment would modify the TS by relocating specific surveillance frequencies to a licensee-controlled program with implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." The changes are consistent with U.S. Nuclear Regulatory Commission (NRC)-approved TS Task Force Standard TS change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control- RITSTF [Risk-Informed TS Task Force] Initiative 5b," Revision 3.

Subsequent to submittal of the referenced license amendment request (LAR), FPL identified that some of the proposed changes to the TS require modifications. This supplement to the

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NRR

LAR relocates additional frequencies to the Surveillance Frequency Control Program consistent with TSTF-425. The supplement also includes changes as a result of the differences between the content and format of the TS on which TSTF-425 was based (NUREG-1431, Standard Technical Specifications – Westinghouse Plants) and the Turkey Point TS, which are not based on NUREG-1431.

Attachment 1 provides a description and evaluation of proposed modifications to the LAR. Attachment 2 provides revised markups of the affected TS pages showing the proposed changes to the Turkey Point TS. The TS markups in this supplement supersede the corresponding markups previously submitted in Reference 1 and Reference 3. Attachment 3 contains revised markups of the TS Bases, which supersede the markups provided in Reference 1.

The changes provided in this supplement do not alter the conclusion in the LAR that the proposed changes do not involve a significant hazards consideration.

This supplement to the LAR contains no new regulatory commitments and does not modify any existing commitments.

These changes have been reviewed by the Turkey Point Plant Nuclear Safety Committee.

Pursuant to 10 CFR 50.91(b)(1), a copy of this submittal is being forwarded to the designated State of Florida official.

Should you have any questions regarding this submittal, please contact Mr. Mitch Guth, Licensing Manager, at 305-246-6698.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 4/3/15

Sincerely,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachments (3)

cc: NRC Regional Administrator, Region II
NRC Senior Resident Inspector
NRC Project Manager
Ms. Cindy Becker, Florida Department of Health

ATTACHMENT 1

Description and Evaluation of Modifications to Proposed Changes

Introduction

This supplement modifies the proposed changes in Turkey Point License Amendment Request (LAR) No. 229, "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program." Following is a description and evaluation of the proposed changes.

Additional Changes

a. As discussed in LAR No. 229, the Turkey Point Technical Specifications (TS) include plant-specific surveillances that are not contained in NUREG-1431 and, therefore are not included in the NUREG-1431 surveillances provided in TSTF-425. Florida Power & Light Company (FPL) has determined that the relocation of the frequencies for these Turkey Point specific surveillances is consistent with TSTF-425, Revision 3, and with the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996), including the scope exclusions identified in Section 1.0, "Introduction," of the model safety evaluation, because the plant-specific surveillance frequencies involve fixed period frequencies. Changes to the frequencies for these plant-specific surveillances would be controlled under the Surveillance Frequency Control Program (SFCP). Accordingly, FPL proposes to relocate the frequencies for the Turkey Point plant-specific surveillance requirements (SRs) below, which are not included in TSTF-425.

SR	Description
4.1.2.1.a	Verify temperature of rooms containing boration flow path components
4.1.2.2.a	Verify temperature of rooms containing boration flow path components
4.2.2.2.b.2	Verify $F_j(Z)$ within limits
4.3.2.1	Table 4.3-2 notation # (each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay)
4.4.9.2	Verify pressurizer temperatures and spray water temperature differential within limits
4.5.2.a (footnote *)	Verify air supply is shut off and sealed closed
4.9.5	Verify communications between control room and refueling personnel
4.10.1.1	Verify control rod positions
4.10.2.1	Verify thermal power $\leq$ 85%
4.10.2.2	Verify requirements of listed specifications
4.10.5	Verify position indication systems are operable

b. FPL proposes to relocate the SR frequencies below consistent with the frequencies relocated in TSTF-425. The table below provides a cross-reference between the SRs contained in TSTF-425 and the corresponding SRs in the Turkey Point TS.

In some instances, a table notation or footnote in the Turkey Point TS specifies the frequency for a SR, and FPL proposes to relocate these frequencies. This is an administrative deviation from TSTF-425 due to differences in the format and content of NUREG-1431 and the Turkey Point TS, which has no impact on the applicability of the corresponding model safety evaluation.

TSTF-425 SR	Turkey Point SR	Description
3.1.5.1	4.1.3.5.b	Verify shutdown banks within the insertion limit
3.2.4.2	4.2.4.2	Verify QPTR within limit using the movable incore detectors
3.3.1.5	4.3.1.1 (Table 4.3-1, notation 7	Perform reactor trip system actuation logic test
3.4.3.1	4.4.9.1.1	Verify reactor coolant system pressure and temperature within limits
3.4.12.8	4.4.9.3.1.a	Perform channel operational test on required PORV
3.4.12.5	4.4.9.3.2 (footnote **)	Verify reactor coolant vent path is open
3.5.2.5	4.5.2.f.1	Verify each ECCS automatic valve actuates to correct position
3.5.2.6	4.5.2.f.2	Verify each ECCS pump starts automatically
3.9.4.1	4.9.4	Verify each containment penetration in required status
3.9.4.2	4.9.9	Verify containment ventilation isolation on high radiation signal
3.9.7.1	4.9.10	Verify refueling cavity water level

Modifications to Proposed Changes

a. TSTF-425 excludes relocating frequencies that reference other approved programs for the specific interval (such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program). The approved programs for Turkey Point are described in Section 6.0, "Administrative Controls," of the Turkey Point TS. (TS 4.0.5 addresses SRs related to the Inservice Testing Program.) The titles of TS Table 4.4-4, Reactor Coolant Specific Activity Sample and Analysis Program; and Table 4.7-1, Secondary Coolant System Specific Activity Sample and Analysis Program, may be misconstrued as programs. However, Section 6.0 of the TS does not contain programs for sampling and analysis of reactor coolant or secondary coolant specific activity. To avoid a misunderstanding of these SRs, FPL proposes to delete the word "Program" from the titles of TS Table 4.4-4 and Table 4.7-1. SR 4.4.8 and SR 4.7.1.4, which refer

to the tables, are also revised to remove the word "program." These changes are an administrative deviation from TSTF-425 with no impact on the applicability of the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

b. The Turkey Point TS include a number of surveillance requirements that specify a surveillance frequency on a staggered test basis similar to "At least once per 31 days on a STAGGERED TEST BASIS." In these cases, LAR No. 229 relocated only a portion of the surveillance frequency (at least once per 31 days) but retained the phrase "on a STAGGERED TEST BASIS."

   TSTF-425 relocates the phrase "on a STAGGERED TEST BASIS" to licensee control under the SFCP. The purpose of specifying certain surveillances to be performed on a staggered test basis is to increase the reliability of the tested system by identifying common mode failures more quickly. Relocating the frequency requirement to perform surveillances on a staggered test basis along with the periodicity allows flexibility to adjust the frequency based on operational experience and risk assessment results. For example, a frequency may be extended with a new requirement to perform the surveillance on a staggered test basis to reflect a higher risk associated with common mode failures; or conversely, a frequency may be changed to eliminate a requirement to perform the surveillance on a staggered test basis due to a lower risk or operational experience associated with common mode failure. NEI 04-10 contains information to support the correct risk modeling of surveillance frequencies with and without a requirement to perform the surveillance on a staggered test basis.

   Consistent with TSTF-425, FPL proposes to relocate the phrase "on a STAGGERED TEST BASIS" from the Turkey Point surveillance requirements listed below.

Auxiliary Feedwater System	SR 4.7.1.2.1.a
Standby Feedwater System	SR 4.7.1.6.2
Control Room Emergency Ventilation System	SR 4.7.5.d.2
A.C. Sources	SR 4.8.1.1.2.a

c. SR 4.1.3.2.2 for the control rod position indication systems states: "Each of the above required analog rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST performed in accordance with Table 4.1-1." LAR No. 229 revised Table 4.1-1 to specify a frequency of "SFCP" (in accordance with the Surveillance Frequency Control Program) for each of the required surveillance tests.

   FPL proposes to revise SR 4.1.3.2.2 by replacing the phrase *in accordance with Table 4.1-1* with *in accordance with the Surveillance Frequency Control Program*. Additionally, the SR is modified by adding a note that states the channel calibration is not applicable to the demand position indication system as indicated in Table 4.1-1. SR 4.1.3.3.2,

which refers to Table 4.1-1, is also revised to refer to SR 4.1.3.2.2 rather than Table 4.1-1. With these proposed changes, Table 4.1-1 becomes redundant to SR 4.1.3.2.2 and serves no purpose. SR 4.1.3.2.2 now addresses all the necessary information previously contained in Table 4.1-1. The proposed change to eliminate Table 4.1-1 does not remove or relax any requirements but only eliminates duplicate information and simplifies the TS. This change is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the applicability of the corresponding model safety evaluation.

d. SR 4.5.2.b currently states:

> Each ECCS component and flow path shall be demonstrated OPERABLE:
>
> b. At least once per 31 days by:
>
> ...
>
> 3) Verifying that each RHR pump develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1 when tested pursuant to Specification 4.0.5.

LAR No. 229 proposed the following changes to SR 4.5.2:

> Each ECCS component and flow path shall be demonstrated OPERBLE:
>
> b. *In accordance with the Surveillance Frequency Control Program* by:
>
> ...
>
> 3) Verifying that each RHR pump develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1 when tested pursuant to Specification 4.0.5.
>
> c. *In accordance with the Surveillance Frequency Control Program* by:
>
> ...
>
> 1) Verifying that each SI pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5.

In a request for additional information (RAI) (Reference 2 in the cover letter), the NRC staff noted that applying the SFCP to SR 4.5.2.b is not allowed by TSTF-425. TS 4.0.5 requires that Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME OM Code and applicable Addenda, and TSTF-425 does not allow relocating surveillance frequencies that reference other approved programs, such as the Inservice Testing Program, for the specific interval. As a result, FPL's response to the RAI (Reference 3 in the cover letter) revised the proposed change to SR 4.5.2 b.3 to the following:

Each ECCS component and flow path shall be demonstrated OPERABLE:

...3) *At least once per 31 days by verifying* that each RHR pump develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1 when tested pursuant to Specification 4.0.5.

SR 4.5.2 b.3 now requires surveillance testing of the residual heat removal (RHR) pumps every 31 days when tested pursuant to TS 4.0.5. However, TS 4.0.5 requires performing Inservice testing in accordance with the ASME OM Code, and the specified frequency for testing the RHR pumps under TS 4.0.5 is quarterly (once per 92 days). As a result, SR 4.5.2.b.3 contains conflicting frequencies (31 days and 92 days), so compliance with the SR would require performing the 92-day SR required by TS 4.0.5 every 31 days. At the same time, the change to SR 4.5.2.c proposed in LAR No. 229, which was not revised by the RAI response, inappropriately applies the SFCP to surveillance testing of the safety injection (SI) pumps when tested pursuant to TS 4.0.5.

To resolve the issues associated with the proposed changes to SR 4.5.2.b.2 and 4.5.2.b.3, FPL proposes to remove the surveillance frequencies from the SRs. Because the SRs are performed pursuant to TS 4.0.5, and because the ASME OM Code establishes the required surveillance frequency, no need exists to specify the TS 4.0.5 required frequency in the SR. Current SR 4.5.2.c is revised as shown below to address testing of the RHR and SI pumps in accordance with TS 4.0.5:

Each ECCS component and flow path shall be demonstrated OPERABLE:

c. *By verifying that each SI and RHR pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5:*

1) SI Pump

$\geq$1083 psid at a metered flowrate $\geq$ 300 gpm (normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST), or

$\geq$1113 psid at a metered flowrate $\geq$ 280 gpm (Unit 3 SI pumps aligned to Unit 4 RWST).

2) RHR Pump

Develops the indicated differential pressure applicable to operating conditions in accordance with Figure 3.5-1.

The proposed change to SR 4.5.2.c resolves the discrepancies in the SR frequencies for testing the RHR and SI pumps and meets the exclusion criterion in TSTF-425 for frequencies that reference other approved programs for the specific interval. Removing the frequencies establishes consistency with other SRs in the Turkey Point TS that specify testing in accordance with TS 4.0.5. For example, SR 4.6.2.1 requires

demonstrating operability of the containment spray pumps by verifying the each pump develops the indicated differential pressure when tested pursuant to Specification 4.0.5. SR 4.6.2.1 does not contain a specific frequency. Similarly, SR 4.6.4.3 verifies the isolation times of certain containment isolation valves are within limits when tested pursuant to Specification 4.0.5 without specifying a frequency in the SR. The proposed change also aligns the Turkey Point TS more closely with SR 3.5.2.4 in NUREG-1431, where the frequency for verifying ECCS pumps' performance is "In accordance with the Inservice Testing Program."

Removing the frequencies from the SRs that require testing pursuant to TS 4.0.5 is a deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the applicability of the corresponding model safety evaluation. The proposed change is appropriate because it eliminates duplicate and conflicting requirements in the SRs. With this change, the SRs will continue to require testing of the SI and RHR pumps in accordance with the ASME OM Code at the frequency specified in the Code.

e. This supplement updates the TS index to reflect the proposed changes and corrects two misspellings in SR 4.8.1.1.2.f. These changes are administrative in nature and have no impact on the applicability of the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

Revised Proposed Bases Changes

LAR No. 229 discusses that the Turkey Point TS were based on the standard TS at the time they were issued, which did not contain Bases as comprehensive as those in NUREG-1431. Therefore, many of the Bases markups in TSTF-425 are not applicable to the Turkey Point TS. The proposed Bases changes revise only those Bases that currently discuss surveillance frequencies. The LAR also states that the existing Bases information describing the basis for the surveillance frequencies will be relocated to the Turkey Point SFCP. The revised Bases provided in this supplement delete information describing the basis for current surveillance frequencies, which will be relocated to the Turkey Point SFCP.

ATTACHMENT 2

Turkey Point
Mark-ups of Technical Specifications Pages

INSERT 1

In accordance with the Surveillance Frequency Control Program

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

(THIS TABLE NUMBER IS NOT USED)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid storage tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.4a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.4b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

Insert 1

a. ~~At least once per 7 days~~ by verifying that the temperature of the rooms containing flow path components is greater than or equal to 62°F when a flow path from the boric acid tanks is used, and

b. ~~At least once per 31 days~~ by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated OPERABLE:

Insert 1

a. ~~At least once per 7 days~~ by verifying that the temperature of the rooms containing flow path components is greater than or equal to 62°F when a flow path from the boric acid tanks is used;

b. ~~At least once per 31 days~~ by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

c. ~~At least once per 18 months~~ by verifying that the flow path required by Specification 3.1.2.2a. and c. delivers at least 16 gpm to the RCS.

SURVEILLANCE REQUIREMENTS

Insert 1

4.1.3.2.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Analog Rod Position Indication System agree within the Allowed Rod Misalignment of Specification 3.1.3.1 (allowing for one hour thermal soak after rod motion) ~~at least once per 12 hours~~ except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the Demand Position Indication System and the Analog Rod Position Indication System at least once per 4 hours.

4.1.3.2.2 Each of the above required analog rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST performed ~~in accordance with Table 4.1-1~~.

Insert 1

NOTE

CHANNEL CALIBRATION is not applicable to the Demand Position Indication system.

(THIS TABLE NUMBER IS NOT USED)

TABLE 4.1-1

~~ROD POSITION INDICATOR SURVEILLANCE REQUIREMENTS~~

~~Functional Unit~~	~~Check~~	~~Calibration~~	~~Operational Test~~
~~Individual Rod Position~~	~~S~~	~~R~~	~~M~~
~~Demand Position~~	~~S~~	~~N/A~~	~~R~~

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group step counter demand position indicator shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown and control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **

ACTION:

With less than the above required group step counter demand position indicator(s) OPERABLE, open the reactor trip system breakers.

Insert 1

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each of the above required group step counter demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction ~~at least once per 31 days.~~

4.1.3.3.2 OPERABILITY of the group step counter demand position indicator shall be verified in accordance with ~~Table 4.1-1~~.

Surveillance Requirement 4.1.3.2.2.

* With the Reactor Trip System breakers in the closed position.

** See Special Test Exceptions Specification 3.10.5.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2* **

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. ~~At least once per 12 hours~~ thereafter.

Insert 1

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.0

2) The following action shall be taken:

a) Comply with the requirements of Specification 3.2.2 for $F_Q^M(Z)$ exceeding its limit by the percent calculated above.

4.2.2.2 MIDS

Operation is permitted at power above P_T where P_T equals the ratio of $[F_Q]^L$ divided by $[F_Q]^P$ if the following Augmented Surveillance (Movable Incore Detection System, MIDS) requirements are satisfied:

a. The axial power distribution shall be measured by MIDS when required such that the limit of $[F_Q]^L/P$ times K(Z) is not exceeded. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation (Z).

1) If $F_j(Z)$ exceeds $[F_j(Z)]_s$* as defined in the bases by ≤ 4%, immediately reduce thermal power one percent for every percent by which $[F_j(Z)]_s$ is exceeded.

2) If $F_j(Z)$ exceeds $[F_j(Z)]_s$ by > 4% immediately reduce thermal power below P_T. Corrective action to reduce $F_j(Z)$ below the limit will permit return to thermal power not to exceed current P_L** as defined in the bases.

b. $F_j(Z)$ shall be determined to be within limits by using MIDS to monitor the thimbles required per Specification 4.2.2.2.c at the following frequencies.

1. ~~At least once every 24 hours,~~ and

Insert 1

2. Immediately following and ~~as~~ a minimum at 2, 4 and 8 hours following the events listed below and ~~every 24 hours~~ thereafter.

1) Raising the thermal power above P_T, or

2) Movement of control-bank D more than an accumulated total of 15 steps in any one direction.

c. MIDS shall be operable when the thermal power exceeds P_T with:

1) At least two thimbles available for which $\overline{R}_j$ and σ_j as defined in the bases have been determined.

---

* $[F_j(Z)]_s$ is the alarm setpoint for MIDS.

** P_L is reactor thermal power expressed as a fraction of the Rated Thermal Power that is used to calculate $[F_j(Z)]_s$.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of -limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

Insert 1

a. Calculating the ratio ~~at least once per 7 days~~ when the Power Range Upper Detector High Flux Deviation and Power Range Lower Detector High Flux Deviation Alarms are OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when either alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained either from two sets of four symmetric thimble locations or full-core flux map, or by incore thermocouple map is consistent with the indicated QUADRANT POWER TILT RATIO ~~at least once per 12 hours.~~ Insert 1

4.2.4.3 If the QUADRANT POWER TILT RATIO is not within its limit within 24 hours and the POWER DISTRIBUTION LIMITS of 3.2.2 and 3.2.3 are within their limits, a Special Report in accordance with 6.9.2 shall be submitted within 30 days including an evaluation of the cause of the discrepancy.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in UFSAR Section 7.2.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power level indication above 15% of RATED THERMAL POWER (RTP). Adjust excore channel gains consistent with calorimetric power level if the absolute difference is greater than 2%. Below 70% RTP, downward adjustments of NIS excore channel gains to match a lower calorimetric power level are not required. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) This table Notation number is not used.

(6) Incore-Excore Calibration, above 75% of RATED THERMAL POWER (RTP). If the quarterly surveillance requirement coincides with sustained operation between 30% and 75% of RTP, calibration shall be performed at this lower power level. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested ~~at least every 62 days on a STAGGERED TEST BASIS~~.

(8) DELETED

Insert 1

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissive P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of 1/2 decade above the existing count rate.

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the undervoltage and shunt trip attachment of the Reactor Trip Breakers.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Insert 1

TABLE NOTATIONS

~~At least once per 18 months~~ each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is *functioning as required before returning the channel to service.*

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in UFSAR Section 7.2

Insert 1

(1) Each train shall be tested ~~at least every 62 days on a STAGGERED TEST BASIS~~.

(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

(3) The provisions of Specification 4.0.4 are not applicable for entering Mode 3, provided that the applicable surveillances are completed within 96 hours from entering Mode 3.

(4) Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment.

(5) Test of alarm function not required when alarm locked in.

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.25 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133.

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

ACTION:

a. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries per gram once per 4 hours.

b. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 60 microcuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 0.25 microcuries per gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131 for greater than or equal to 48 hours during one continuous time interval, or greater than 60 microcuries per gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

d. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 447.7 microcuries per gram limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133 for greater than or equal to 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by ~~performance of~~ performing the sampling and analysis ~~program of~~ described in Table 4.4-4.

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS ~~PROGRAM~~

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. NOT USED		
2. Tritium Activity Determination	~~1 per 7 days.~~ SFCP	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131	a) ~~1 per 14 days.~~ SFCP b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period.	1, 2, 3, 4
4. Radiochemical Isotopic Determination Including Gaseous Activity	~~Monthly.~~ SFCP	1, 2, 3, 4
5. Isotopic Analysis for DOSE EQUIVALENT XE-133	~~1 per 7 days~~ SFCP	1, 2, 3, 4
6. NOT USED		

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 5°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

Insert 1

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits ~~at least once per 30 minutes~~ during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

Insert 1

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits ~~at least once per 30 minutes~~ during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit ~~at least once per 12 hours~~ during auxiliary spray operation.

Insert 1

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST* on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and ~~at least once per 31 days~~ Insert 1 thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel ~~at least once per 18 months~~ Insert 1; and

c. Verifying the PORV block valve is open ~~at least once per 72 hours~~ Insert 1 when the PORV is being used for overpressure protection.

d. While the PORVs are required to be OPERABLE, the backup nitrogen supply shall be verified OPERABLE ~~at least once per 24 hours~~ Insert 1.*

4.4.9.3.2 The 2.20 square inch vent shall be verified to be open ~~at least once per 12 hours~~ Insert 1** when the vent(s) is being used for overpressure protection.

4.4.9.3.3 Verify the high pressure injection flow path to the RCS is isolated ~~at least once per 24 hours~~ Insert 1 by closed valves with power removed or by locked closed manual valves.

* Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ 275°F.

** Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open ~~at least once per 31 days~~ Insert 1.

3/4.4.10 DELETED

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS component and flow path shall be demonstrated OPERABLE:

Insert 1

a. ~~At least once per 12 hours~~ by verifying by control room indication that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
864A and B	Supply from RWST to ECCS	Open
862A and B	RWST Supply to RHR pumps	Open
863A and B	RHR Recirculation	Closed
866A and B	H.H.S.I. to Hot Legs	Closed
HCV-758*	RHR HX Outlet	Open

To permit temporary operation of these valves for surveillance or maintenance purposes, power may be restored to these valves for a period not to exceed 24 hours.

Insert 1

b. ~~At least once per 31 days~~ by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position~~, and~~ .

~~3) Verifying that each RHR Pump develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1 when tested pursuant to Specification 4.0.5.~~

c. ~~At least once per 92 days by:~~ By verifying that each SI and RHR

~~1)~~ ~~Verifying that each SI~~ pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5~~.~~ :

1) SI pump ≥ 1083 psid at a metered flowrate ≥ 300 gpm (normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST), or

≥ 1113 psid at a metered flowrate ≥ 280 gpm (Unit 3 SI pumps aligned to Unit 4 RWST).

2) RHR pump

Develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1.

*Air Supply to HCV-758 shall be verified shut off and sealed closed ~~once per 31 days~~.

Insert 1

d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. The visual inspection shall be performed:

   1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2) At least once daily of the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

Insert 1

e. ~~At least once per 18 months~~ by:

   1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 525 psig the interlocks cause the valves to automatically close and prevent the valves from being opened, and

   2) Verifying correct interlock action to ensure that the RWST is isolated from the RHR System during RHR System operation and to ensure that the RHR System cannot be pressurized from the Reactor Coolant System unless the above RWST Isolation Valves are closed.

   3) A visual inspection of the containment sump and verifying that the suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

f. ~~At least once per 18 months~~, during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation test signal, and

   2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

      a) Safety Injection pump, and

      b) RHR pump.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

B

ACTION:

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

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify the OPERABILITY of both standby feed-water pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

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

a. ~~At least once per 31 days on a STAGGERED TEST BASIS~~ by:

Insert 1

1) Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or

---

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

PLANT SYSTEMS

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS ~~PROGRAM~~

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	~~At least once per 72 hours.~~ SFCP
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

PLANT SYSTEMS

STANDBY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two Standby Steam Generator Feedwater Pumps shall be OPERABLE* and at least 145,000 gallons of water (indicated volume), shall be in the Demineralized Water Storage Tank.**

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one Standby Steam Generator Feedwater Pump inoperable, restore the inoperable pump to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6d.

b. With both Standby Steam Generator Feedwater Pumps inoperable, restore at least one pump to OPERABLE status within 24 hours, or:

   1. Notify the NRC within the following 4 hours, and provide cause for the inoperability and plans to restore pump(s) to OPERABLE status and,

   2. Submit a SPECIAL REPORT per 3.7.1.6d.

c. With less than 145,000 gallons of water indicated in the Demineralized Water Storage Tank restore the available volume to at least 145,000 gallons indicated within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the inoperability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 The Demineralized Water Storage tank water volume shall be determined to be within limits ~~at least once per 24 hours~~.

Insert 1

4.7.1.6.2 ~~At least monthly~~ verify the standby feedwater pumps are OPERABLE by testing in recirculation ~~on a STAGGERED TEST BASIS~~.

4.7.1.6.3 ~~At least once per 18 months~~, verify operability of the respective standby steam generator feedwater pump by starting each pump and providing feedwater to the steam generators.

*These pumps do not require plant safety related emergency power sources for operability and the flowpath is normally isolated.

**The Demineralized Water Storage Tank is non-safety grade.

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99.95% DOP and 99% halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%***.

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 - 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 2.5% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement***, and

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration***.

Insert 1

d.1 ~~At least once per 12 months~~ by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%***;

d.2 ~~On staggered test basis every 36 months~~, test the supply fans (trains A and B) and measure CRE pressure relative to external areas adjacent to the CRE boundary.***

e. ~~At least once per 18 months~~ by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation,

f. ~~At least once per 18 months~~ by verifying operability of the kitchen and toilet area exhaust dampers, and

g. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.***

***As the mitigation actions of TS 3.7.5 Action a.5 include the use of the compensatory filtration unit, the unit shall meet the surveillance requirements of TS 4.7.5.b, by manual initiation from outside the control room and TS 4.7.5.c, d and g.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE*:

a. Insert 1 ~~At least once per 31 days on a STAGGERED TEST BASIS~~ by:

1) Verifying the fuel volume in the day and skid-mounted fuel tanks (Unit 4-day tank only),

2) Verifying the fuel volume in the fuel storage tank,

3) Verifying the lubricating oil inventory in storage,

4) Verifying the diesel starts and accelerates to reach a generator voltage and frequency of 3950-4350 volts and 60 ± 0.6 Hz. Insert 1 ~~Once per 184 days~~, these conditions shall be reached within 15 seconds after the start signal from ~~fro m~~ normal conditions. For all other starts, warmup procedures, such as idling and gradual acceleration as recommended by the manufacturer may be used. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded** to 2300 - 2500 kW (Unit 3), 2650-2850 kW (Unit 4)***, operates at this loaded condition for at least 60 minutes and for Unit 3 until automatic transfer of fuel from the day tank to the skid mounted tank is demonstrated, and the cooling system is demonstrated OPERABLE.

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency buses ~~busses.~~

---

* All diesel generator starts for the purpose of these surveillances may be proceeded by a prelube period as recommended by the manufacturer.

** May include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

***Momentary transients outside these load bands do not invalidate this test.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

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

a. The equipment door closed and held in place by a minimum of four bolts.

b. A minimum of one door in each airlock is closed, or, both doors of the containment personnel airlock may be open if:
   1) at least one personnel airlock door is capable of being closed.
   2) The plant is in MODE 6 with at least 23 feet of water above the reactor vessel flange, and
   3) a designated individual is available outside the personnel airlock to close the door.

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, or manual valve, or

   2) Be capable of being closed by an OPERABLE automatic containment ventilation isolation valve.

APPLICABILITY: During movement of recently irradiated fuel within the containment.

ACTION:
With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of recently irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment ventilation isolation valve within 100 hours prior to the start of and ~~at least once per 7 days~~ Insert 1 during movement of recently irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment ventilation isolation valves per the applicable portions of Specification 4.6.4.2.

*Exception may be taken under Administrative Controls for opening of certain valves and airlocks necessary to perform surveillance or testing requirements.

REFUELING OPERATIONS

3/4.9.5 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.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and ~~at least once per 12 hours~~ during CORE ALTERATIONS.

Insert 1

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

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

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the containment ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

Insert 1

4.9.9 The Containment Ventilation 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 Ventilation Isolation occurs on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

3/4.9.10 REFUELING CAVITY WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.10 Refueling cavity water level shall be maintained $\geq$ 23 feet above the top of the reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTION:

With the refueling cavity water level not within limit, suspend movement of irradiated fuel assemblies within containment immediately.

SURVEILLANCE REQUIREMENTS

4.9.10 Verify refueling cavity water level is $\geq$ 23 feet above the top of the reactor vessel flange within 2 hours prior to the start of and ~~at least once per 24 hours~~ thereafter during movement of irradiated fuel assemblies within containment.

Insert 1

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm). boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

Insert 1

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined ~~at least once per 2 hours~~.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

Insert 1

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER ~~at least once per hour~~ during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed ~~at least once per 12 hours~~ during PHYSICS TESTS:

Insert 1

a. Specifications 4.2.2.1 and 4.2.2.5, and

b. Specification 4.2.3.3.

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

Insert 1

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and ~~at least once per 24 hours~~ thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Analog Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

ATTACHMENT 3

Turkey Point

Mark-up of Technical Specifications Bases Pages

INSERT BASES 1

In accordance with the Surveillance Frequency Control Program

ATTACHMENT 2
Technical Specification Bases
(Page 40 of 183)

3/4.1.3 (Continued)

TS 3.1.3.2 Action a.2.c) requires the use of the Movable Incore Detector System to verify rod position prior to increasing thermal power above 50 percent Rated Thermal Power (RTP) and within 8 hours of reaching 100 percent RTP. These provisions are intended to establish and confirm the position of the rod with the inoperable RPI to ensure that power distribution requirements are **NOT** violated.

The ACTION Statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with Tavg greater than or equal to 500°F and with all Reactor Coolant Pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor Trip at operating conditions.

Insert Bases 1

Control rod positions and OPERABILITY of the Rod Position Indicators are required to be verified ~~on a nominal basis of once per 12 hours~~ with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

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3/4.2.5 DNB Parameters

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial UFSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR above the applicable design limits throughout each analyzed transient. The limits for Tavg and pressurizer pressure have been moved to the COLR. The measured RCS flow value of 270,000 gpm corresponds to a Thermal Design Flow of 260,700 gpm with an allowance of 3.5% to accommodate calorimetric measurement uncertainty.

~~The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to ensure that the DNB-related flow assumption is met and to ensure correlation of the flow indication channels with measured flow. The indicated percent flow surveillance on a 12 hour basis will provide sufficient verification that flow degradation has **NOT** occurred. An indicated percent flow which is greater than the thermal design flow plus instrument channel inaccuracies and parallax errors is acceptable for the 12 hour surveillance on RCS flow. To minimize measurement uncertainties it is assumed that the RCS flow channel outputs are averaged.~~

The surveillance frequencies are controlled under the Surveillance Frequency Control Program.

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3/4.3 Instrumentation

3/4.3.1
&
3/4.3.2 Reactor Trip System and Engineered Safety Features Actuation System Instrumentation

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) The associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) The specified coincidence logic is maintained, (3) Sufficient redundancy is maintained to permit a channel to be out of-service for testing or maintenance (due to plant specific design, pulling fuses and using jumpers may be used to place channels in trip), and (4) Sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. ~~The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Surveillances for the analog RPS/ESFAS Protection and Control rack instrumentation have been extended to quarterly in accordance with WCAP-10271, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System, and supplements to that report as generically approved by the NRC and documented in their SERs (Letters to the Westinghouse Owner's Group from the NRC dated February 21, 1985, February 22, 1989, and April 30, 1990).~~

Under some pressure and temperature conditions, certain surveillances for Safety Injection cannot be performed because of the system design. Allowance to change modes is provided under these conditions as long as the surveillances are completed within specified time requirements.

The surveillance frequencies are controlled under the Surveillance Frequency Control Program.

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3/4.4.2 (Continued)

In MODE 5 only one Pressurizer Code Safety is required for overpressure protection. In lieu of an actual OPERABLE Code Safety Valve, an unisolated and unsealed vent pathway (i.e., a direct, unimpaired opening, a vent pathway with valves locked open and/or power removed and locked on an open valve) of equivalent size can be taken credit for as synonymous with an OPERABLE Code Safety.

Demonstration of the safety valves lift settings will occur only during shutdown and will be performed in accordance with the provisions of the ASME OM Code. The Pressurizer Code Safety Valves lift settings allows a +2%, -3% setpoint tolerance for OPERABILITY; however, the valves are reset to within ±1% during the surveillance to allow for drift.

3/4.4.3 Pressurizer

Surveillance requirement 4.4.3.1

S

~~The 12 hour periodic surveillance is sufficient to~~ ensure that the maximum water volume parameter is restored to within its limit following expected transient operation. The maximum water volume (1133 cubic feet) ensures that a steam bubble is formed and thus the RCS is **NOT** a hydraulically solid system. The requirement that both backup pressurizer heater groups be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

The surveillance frequencies are controlled under the Surveillance Frequency Control Program.

3/4.4.4 Relief Valves

The opening of the power-operated relief valves (PORVs) fulfills **NO** safety-related function and **NO** credit is taken for their operation in the safety analysis for MODE 1, 2 or 3. Equipment necessary to establish PORV operability in Modes 1 and 2 is limited to Vital DC power and the Instrument Air system. Equipment necessary to establish block valve operability is limited to an AC power source. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a PORV fail in the open position.

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3/4.4.6.2 (Continued)

Action d.

With one or more RCS Pressure Isolation Valves with leakage greater than 5 gpm, the leakage must be reduced to below 5 gpm within 1 hour or the reactor must be brought to at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

The allowable outage times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillance Requirements

SR 4.4.6.2.1

Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is **NOT** PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System water inventory balance.

a. & b.

These SRs demonstrate that the RCS operational leakage is within the LCO limits by monitoring the containment atmosphere gaseous or particulate radioactivity monitor and the containment sump level ~~at least once per 12 hours~~.

Insert Bases 1

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3/4.4.6.2 (Continued)

c.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two notes. Note *** states that this SR is **NOT** required to be performed until 12 hours after establishment of steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operations is required to perform a proper inventory balance since calculations during maneuvering are **NOT** useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, Pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity, containment normal sump inventory and discharge, and reactor head flange leak-off. It should be noted that leakage past seals and gaskets is **NOT** PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, Reactor Coolant System Leakage Detection Systems.

Note ** states that this SR is **NOT** applicable to primary-to-secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

~~The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.~~

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

d.

This SR demonstrates that the RCS Operational Leakage is within the LCO limits by monitoring the Reactor Head Flange Leak-off System ~~at least once per 24 hours~~.

Insert Bases 1

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3/4.4.6.2 (Continued)

e.

This SR verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one SG. Satisfying the primary-to-secondary leakage limit ensure that the operational leakage performance criterion in the Steam Generator Program is met. If this SR is **NOT** met, compliance with LCO 3.4.5, Steam Generator (SG) Tube Integrity, should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 5. The operational leakage rate limit applies to leakage through any one SG. If it is **NOT** practical to assign the leakage to an individual SG, all the primary to secondary leakage should be conservatively assumed to be from one SG.

The SR is modified by Note ***, which states that the Surveillance is **NOT** required to be performed until 12 hours after establishment of steady state operation. For RCS primary to-secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

is controlled under the Surveillance Frequency Control Program.

The surveillance frequency ~~of 72 hours is a reasonable interval to trend primary to secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents~~. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

SR 4.4.6.2.2

It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping, which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

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3/4.4.8 (Continued)

The RCS Specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

The iodine specific activity in the reactor coolant is limited to 0.25 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 447.7 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that the offsite and Control Room doses will meet the appropriate SRP acceptance criteria.

The SLB, SGTR, Locked Rotor, and RCCA Ejection Accident analyses show that the calculated doses are within limits. Violation of the LCO may result in reactor coolant activity levels that could, in the event of any one of these accidents, lead to doses that exceed the acceptance criteria.

The ACTIONS permit limited operation when DOSE EQUIVALENT I-131 is greater than 0.25 μCi/gm and less than 60 μCi/gm. The Actions require sampling within 4 hours and every 4 hours following to establish a trend.

Insert Bases 1

One surveillance requires performing a gamma isotropic analysis as a measure of noble gas specific activity of the reactor coolant ~~at least once per 7 days~~. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This surveillance provides an indication of any increase in the noble gas specific activity.

Insert Bases 1

A second surveillance is performed to ensure that iodine specific activity remains within the LCO limit ~~once per 14 days~~ during normal operation and following fast power changes when iodine spiking is more apt to occur. The frequency between 2 and 6 hours after a power change of greater than 15% RATED THERMAL POWER within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation.

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3/4.5.2 & 3/4.5.3 (Continued)

When PC-600/-601 are calibrated, a test signal is supplied to each circuit to check operation of the relays and annunciators operated by subject controllers. This test signal will prevent MOVs 862A, 862B, 863A, 863B from opening. Therefore, it is appropriate to tag out the MOV breakers, and enter Technical Specification Action Statement 3.5.2.a. and 3.6.2.1 when calibrating PC-600/-601.

With the RCS temperature below 350°F, operation with less than full redundant equipment is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

TS 3.5.2, Action g. provides an allowed outage/action completion time (AOT) of up to 7 days to restore an inoperable RHR Pump to OPERABLE status, provided the affected ECCS Subsystem is inoperable only because its associated RHR Pump is inoperable. This 7 day AOT is based on the results of a deterministic and probabilistic safety assessment, and is referred to as a Risk-Informed AOT Extension. Planned entry into this AOT requires that a Risk Assessment be performed in accordance with the Configuration Risk Management Program (CRMP), which is described in the administrative procedure that implements the Maintenance Rule pursuant to 10CFR50.65.

TS Surveillance 4.5.2.a requires that each ECCS component and flow path be demonstrated operable ~~at least once per 12 hours~~ by verifying by Control Room indication that the valves listed in Section 4.5.2.a are in the indicated positions with power to the valve operators removed. Verifying Control Room indication applies to the valve position and **NOT** to the valve operator power removal. The breaker position may be verified by either the off condition of the breaker position indication light in the Control Room, or the verification of the locked open breaker position in the field. Verifying that power is removed to the applicable valve operators can be accomplished by direct field indication of the breaker (locked in the open position), or by observation of the breaker position status lamp in the Control Room (lamp is off when breaker is open). Surveillance Requirements for throttle valve position stops prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration.

The surveillance frequencies are controlled under the Surveillance Frequency Control Program.

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3/4.5.2 & 3/4.5.3 (Continued)

ECCS "accessible discharge piping" is defined as discharge piping outside of containment in accordance with NRC Generic Letter 2008-01, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems interpretation. High point vents (current or added) outside of containment on the HHSI and RHR Systems discharge piping are considered accessible. These valves must be included in the ~~monthly~~ venting procedure to comply with Technical Specification Surveillance Requirement 4.5.2.b.1. This clarification was added as a corrective action to CR# 2009-18558.

In the RHR test, differential head is specified in feet. This criteria will allow for compensation of test data with water density due to varying temperature.

ECCS pump testing for the SI and RHR Pumps accounts for possible underfrequency conditions, i.e., the results of pump testing performed at 60 Hz is then adjusted to reflect possible degraded grid conditions (60±0.6) to the lower limit (59.4 Hz).

CAUTION

Interim Compensatory Measure

TS 3.5.2 Action 'a' has been determined to be non-conservative with respect to the safety analysis as it allows up to 72 hours for restoration of the inoperable flow path despite inoperability of both ECCS trains during this period. Therefore, until appropriate changes to TS 3/4.5.2 via LAR 212 are approved and implemented, TS 3.0.3 shall be entered vice TS 3.5.2 Action 'a' in the event that the suction flow path from the RWST to the ECCS is inoperable. Reference AR 1811016.

Technical Specifications Surveillance Requirement 4.5.2.e.3 requires that each ECCS component and flow path be demonstrated OPERABLE ~~every 18 months~~ Insert Bases 1 by visual inspection which verifies sump components (trash racks, screens, etc.) show **NO** evidence of structural distress or abnormal corrosion. The strainer modules are rigid enough to provide both functions as trash racks and screens without losing their structural integrity and particle efficiency. Therefore, strainer modules are functionally equivalent to trash racks and screens. Accordingly, the categorical description, sump components, is broad enough to require inspection of the strainer modules.

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3/4.6.2.2 (Continued)

The allowable out-of-service time requirements for the Containment Cooling System have been maintained consistent with that assigned other inoperable ESF equipment and do **NOT** reflect the additional redundancy in cooling capability provided by the Containment Spray System.

The surveillance requirement for ECC flow is verified by correlating the test configuration value with the design basis assumptions for system configuration and flow. ~~An 18-month surveillance interval is acceptable based on the use of water from the CCW system, which results in a low risk of heat exchanger tube fouling.~~ The surveillance frequency is controlled under the Surveillance Frequency Control Program.

3/4.6.2.3 Recirculation pH Control System

The Recirculation pH Control System is a passive safeguard consisting of 10 stainless steel wire mesh baskets (2 large and 8 small) containing sodium tetra borate decahydrate (NaTB) located in the containment basement (14' elevation). The initial containment spray will be boric acid solution from the Refueling Water Storage Tank. The recirculation pH control system adds NaTB to the Containment Sump when the level of boric acid solution from the Containment Spray and the coolant lost from the Reactor Coolant System rises above the bottom of the buffering agent baskets. As the sump level rises, the NaTB will begin to dissolve. The addition of NaTB from the buffering agent baskets ensures the containment sump pH will be greater than 7.0. The resultant alkaline pH of the spray enhances the ability of the recirculated spray to scavenge fission products from the containment atmosphere. The alkaline pH in the recirculation sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on stainless steel piping systems exposed to the solution.

The OPERABILITY of the recirculation pH control system ensures that there is sufficient NaTB available in the containment to ensure a sump pH greater than 7.0 during the recirculation phase of a postulated LOCA. The baskets will **NOT** interact with surrounding equipment during a seismic event.

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3/4.6.2.3 (Continued)

To satisfy the surveillance requirement, the two large baskets and eight small baskets must contain a combined mass greater than 7500 lbm of NaTB. As shown in the above table, this will ensure the sump pH exceeds 7.0 at the onset of spray recirculation and for the duration of the analyzed 30-day period. The large baskets have a length and width of 54 inches, and a height of 33.25 inches and are elevated 3.5 inches above the containment floor. The smaller baskets have a length and width of 36 inches and a height of 30 inches and are elevated 4.5 inches above the containment floor. Varying basket dimensions or elevation (e.g. basket leg height) impacts the surface to volume ratio and changes the time the NaTB is in contact with containment sump water. For instance, shorter legs would allow the NaTB to contact containment sump water sooner, therefore increasing the pH at the onset of recirculation. Longer legs, however, would reduce the pH at the onset of recirculation. The level of NaTB in the baskets required to provide an equilibrium sump solution pH greater than 7.0 is 14.75 inches from the top of the basket; 18.50 inches for the large baskets and 15.25 inches for the small baskets from the bottom of the basket. ~~The 18 month frequency for Surveillance Requirement 4.6.2.3 is sufficient to ensure that the stainless steel buffering agent baskets are intact and contain the required quantity of NaTB.~~

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

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3/4.7.1.2 Auxiliary Feedwater System

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total Loss-Of-Offsite Power. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two D.C. motor operated valves and one A.C. motor operated valve on each unit isolate the three main steam lines from these headers. Both the D.C. and A.C. motor operated valves are powered from safety-related sources. Auxiliary feedwater can be supplied through redundant lines to the safety-related portions of the main feedwater lines to each of the steam generators. Air operated fail closed flow control valves are provided to modulate the flow to each steam generator. Each Steam Driven Auxiliary Feedwater Pump has sufficient capacity for single and two unit operation to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

ACTION statement 2 describes the actions to be taken when both Auxiliary Feedwater Trains are inoperable. The requirement to verify the availability of both Standby Feedwater Pumps is to be accomplished by verifying that both pumps have successfully passed their ~~monthly~~ surveillance tests within the last surveillance interval. The requirement to complete this action before beginning a unit shutdown is to ensure that an alternate feedwater train is available before putting the affected unit through a transient. If **NO** alternate feedwater trains are available, the affected unit is to stay at the same condition until an auxiliary feedwater train is returned to service, and then invoke ACTION statement 1 for the other train. If both Standby Feedwater Pumps are made available before one Auxiliary Feedwater Train is returned to an OPERABLE status, then the affected units shall be placed in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

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3/4.7.1.2 (Continued)

ACTION statement 3 describes the actions to be taken when a single Auxiliary Feedwater Pump is inoperable. The requirement to verify that two independent Auxiliary Feedwater Trains are OPERABLE is to be accomplished by verifying that the requirements for Table 3.7-3 have been successfully met for each train within the last surveillance interval. The provisions of Specification 3.0.4 are **NOT** applicable to the third auxiliary feedwater pump provided it has **NOT** been inoperable for longer than 30 days. This means that a units can change OPERATIONAL MODES during a unit's heatup with a single Auxiliary Feedwater Pump inoperable as long as the requirements of ACTION Statement 3 are satisfied.

The specified flow rate acceptance criteria conservatively bounds the limiting AFW flow rate modeled in the single unit Loss of Normal Feedwater analysis. Dual unit events such as a two unit Loss of Offsite Power require a higher pump flow rate, but it is **NOT** practical to test both units simultaneously. The ~~monthly~~ flow surveillance test specified in 4.7.1.2.1.1 is considered to be a general performance test for the AFW system and does **NOT** represent the limiting flow requirement for AFW. Check valves in the AFW system that require full stroke testing under limiting flow conditions are tested under Technical Specification 4.0.5.

The ~~monthly~~ testing of the Auxiliary Feedwater Pumps will verify their OPERABILITY. Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification of correct operation will be made both from instrumentation within the Control Room and direct visual observation of the pumps.

The frequencies of surveillance requirements 4.7.1.2.1a and 4.7.1.2.1b are controlled under the Surveillance Frequency Control Program.

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in accordance with the Surveillance Frequency Control Program.

3/4.7.1.6 (Continued)

The Standby Steam Generator Feedwater Pumps are **NOT** designed to NRC requirements applicable to Auxiliary Feedwater Systems and **NOT** required to satisfy Design Basis Events requirements. These pumps may be out of service for up to 24 hours before initiating formal notification because of the extremely low probability of a demand for their operation.

The guidelines for NRC notification in case of both pumps being out of service for longer than 24 hours are provided in applicable plant procedures, as a voluntary 4 hour notification.

Adequate demineralized water for the Standby Steam Generator Feedwater system will be verified ~~once per 24 hours~~. The Demineralized Water Storage Tank provides a source of water to several systems and therefore, requires daily verification.

The Standby Steam Generator Feedwater Pumps will be verified OPERABLE ~~monthly on a STAGGERED TEST BASIS~~ by starting and operating them in the recirculation mode. Also, ~~during each unit's refueling outage,~~ each Standby Steam Generator Feedwater Pump will be started and aligned to provide flow to the nuclear unit's steam generators.

This surveillance regimen will thus demonstrate operability of the entire flow path, backup non-safety grade power supply and pump associated with a unit at least each refueling outage. The pump, motor driver, and normal power supply availability would typically be demonstrated by operation of the pumps in the recirculation mode monthly on a staggered test basis.

periodically

The diesel engine driver for the B Standby Steam Generator Feedwater Pump will be verified operable ~~once every 31 days on a staggered test basis performed on the B Standby Steam Generator Feedwater Pump~~. In addition, an inspection will be performed on the diesel ~~at least once every 18 months~~ in accordance with procedures prepared in conjunction with its manufacture's recommendations for the diesel's class of service. This inspection will ensure that the diesel driver is maintained in good operating condition consistent with FPL's overall objectives for system reliability.

The surveillance frequencies are controlled under the Surveillance Frequency Control Program.

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3/4.7.1.7 (Continued)

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

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

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

SR 4.7.1.7.a verifies that each FCV, FIV, and bypass line valve will actuate to its isolation position on an actuation or simulated actuation signal. ~~The 18 month Frequency is based on a refueling cycle interval and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.~~

SR 4.7.1.7.b verifies that the closure time of each FCV, FIV, and bypass line valve, when tested in accordance with the Inservice Testing Program, is within the limits assumed in the accident and containment analyses. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should **NOT** be tested at power, since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code Section XI (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2.

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3/4.7.4 Ultimate Heat Sink

The limit on Ultimate Heat Sink (UHS) temperature in conjunction with the surveillance requirements of Technical Specification 3/4.7.2 will ensure that sufficient cooling capacity is available either: (1) To provide normal cooldown of the facility, or (2) To mitigate the effects of accident conditions within acceptable limits.

FPL has the option of monitoring UHS temperature by monitoring the temperature in the ICW System piping going to the inlet of the CCW Heat Exchangers. Monitoring UHS temperature after the ICW Pumps, but prior to CCW Heat Exchangers is considered to be equivalent to temperature monitoring before the ICW Pumps. The supply water leaving the ICW Pumps will be mixed, and therefore, it will be representative of the bulk UHS temperature to the CCW Heat Exchanger inlet. The effects of pump heating on the supply water are negligible due to low ICW head and high water volume. Accordingly, monitoring UHS temperature after the ICW Pumps, but prior to the CCW Heat Exchangers provides an equivalent location for monitoring UHS temperature.

With the implementation of the CCW Heat Exchanger Performance Monitoring Program, the limiting UHS temperature can be treated as a variable with an absolute upper limit of 104°F without compromising any margin of safety. Demonstration of actual heat exchanger performance capability supports system operation with postulated canal temperatures greater than 104°F. Therefore, an upper Technical Specification limit of 104°F is conservative.

The frequency of verifying

~~Verifying~~ UHS water temperature ~~at least once per 24 hours is adequate to ensure the limit of 104ºF is **NOT** exceeded~~ when the water temperature is less than 100ºF ~~as there is ample (greater than or equal to 4ºF) margin to the limit.~~ Due to daily variations in temperature, when UHS water temperature exceeds 100ºF the water temperature shall be verified at least once per hour to ensure that Cooling Canal System temperature variations are appropriately captured, thus ensuring the Technical Specification limit is **NOT** exceeded.

is controlled under the Surveillance Frequency Control Program.

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3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

A thermographic examination of high-risk potential ignition sources in the Cable Spreading Room and the Control Room,

Restriction of planned hot work in the Cable Spreading Room and Control Room during the extended AOT, and

Establishment of a continuous fire watch in the Cable Spreading Room when in the extended AOT.

In addition to the predetermined restrictions, assessments performed in accordance with the provisions of the Maintenance Rule (a)(4) will ensure that any other risk significant configurations are identified before removing an EDG from service for pre-planned maintenance.

A configuration risk management program has been established at Turkey Point 3 and 4 via the implementation of the Maintenance Rule and the On line Risk Monitor to ensure the risk impact of out of service equipment is appropriately evaluated prior to performing any maintenance activity.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, Selection of Diesel Generator Set Capacity for Standby Power Supplies, March 10, 1971; 1.108, Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants, Revision 1, August 1977; and 1.137, Fuel-oil Systems for Standby Diesel Generators, Revision 1, October 1979.

The EDG Surveillance testing requires that each EDG be started from normal conditions ~~only once per 184 days~~ with **NO** additional warmup procedures.

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

Normal conditions in this instance are defined as the pre-start temperature and lube oil conditions each EDG normally experiences with the continuous use of prelube systems and immersion heaters.

3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does **NOT** represent a failure to meet the Limiting Condition for Operation of TS 3.8.1.1, since the new fuel oil has **NOT** been added to the Diesel Fuel Oil Storage Tanks.

Within 30 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM□-D1552-79 or ASTM-D2622-82. The 30 day period is acceptable because the fuel oil properties of interest, even if they are **NOT** within limits, would **NOT** have an immediate effect on EDG operation. The Diesel Fuel Oil Surveillance in accordance with the Diesel Fuel Oil Testing Program will ensure the availability of high quality diesel fuel oil for the EDGs.

<u>Lubricity Specification for Ultra Low Sulfur Diesel Fuel Oil</u>

To ensure that Ultra Low Sulfur Diesel fuel (15 pm sulfur, S15) is acceptable for use in the Emergency Diesel Generators, a test is added in the Diesel Fuel Oil Testing Program that validates, satisfactory lubricity (Reference: Engineering Evaluation PTN-ENG-SEMS-06-0035).

The test for lubricity is based on ASTM D975-06, testing per ASTM D6079, using the High Frequency Reciprocating Rig (HFRR) test at 60 degrees C and the acceptance criterion requires a wear scar **NO** larger than 520 microns.

At least once every 31 days, a sample of fuel oil is obtained from the storage tanks in accordance with ASTM-D2276-78. The particulate contamination is verified to be less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does **NOT** mean the fuel oil will **NOT** burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

ATTACHMENT 2
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3/4.9 Refueling Operations

3/4.9.1 Boron Concentration

The limitations on reactivity conditions during REFUELING ensure that: (1) The reactor will remain subcritical during CORE ALTERATIONS, and (2) A uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. With the required valves CLOSED during refueling operations, the possibility of uncontrolled boron dilution of the filled portion of the RCS is precluded. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water. The boration rate requirement of 16 gpm of 3.0 wt% (5245 ppm) boron or equivalent ensures the capability to restore the SHUTDOWN MARGIN with one OPERABLE charging pump.

3/4.9.2 Instrumentation

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. There are four source range neutron flux channels, two primary, and two backup. All four channels have visual and alarm indication in the Control Room and interface with the Containment Evacuation Alarm System. The primary source range neutron flux channels can also generate reactor trip signals and provide audible indication of the count rate in the Control Room and containment. At least one primary source range neutron flux channel to provide the required audible indication, in addition to its other functions, and one of the three remaining source range channels shall be OPERABLE to satisfy the LCO.

T.S. surveillance requirement 4.9.2.b and c states:

Each required Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

Insert Bases 1

c. An ANALOG CHANNEL OPERATIONAL TEST ~~at least once per 7 days.~~

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3/4.9.9 Containment Ventilation Isolation System

The OPERABILITY of this system ensures that the containment ventilation penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

T.S. surveillance requirement 4.9.9 states:

Insert Bases 1

4.9.9 The Containment Ventilation 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 Ventilation Isolation occurs on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

A normal refueling consists of 2 CORE ALTERATION sequences: unloading the core, and reloading the core, typically with a suspension of CORE ALTERATIONS in between. The core unload sequence begins with control rod unlatching, followed by removal of upper internals, followed by unloading fuel assemblies to the SFP. The core reload sequence consists of reloading fuel assemblies from the SFP, followed by upper internals installation, followed by latching control rods. Therefore, if the Containment Ventilation Isolation System is demonstrated OPERABLE ~~at least once per 7 days~~ following the specified testing within 100 hours prior to the start of control rod unlatching, then Containment Ventilation Isolation System operability need **NOT** be demonstrated within 100 hours prior to the start of core reload. Otherwise, the specified testing is required to be performed within 100 hours prior to the start of core reload.

Insert Bases 1