



March 9, 2015

ULNRC-06174

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

10 CFR 50.90

Ladies and Gentlemen:

**DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
LICENSING DOCUMENT CHANGE NOTICE (LDCN) 14-0014
APPLICATION TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT
TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION
FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE
CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (CLIIP)**

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Ameren Missouri (Union Electric Company) herewith transmits an application for amendment to Facility Operating License Number NPF-30 for the Callaway Plant. The proposed amendment would modify Technical Specification (TS) requirements regarding steam generator tube inspections and reporting as described in TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" with some minor administrative differences that are described in the enclosure to this letter.

The enclosure to this letter uses the Model Application for Adoption of TSTF-510 and provides a description and assessment of the proposed change. Attachments 1 through 3 to the Enclosure provide the TS page markups, TS Bases page markups, and retyped TS pages, respectively, in support of this amendment request. Attachment 2 is provided for information only. Final TS Bases changes will be processed under the program for updates per TS 5.5.14, "Technical Specifications (TS) Bases Control Program," at the time the requested amendment is implemented.

Ameren Missouri respectfully requests NRC approval of the proposed license amendment prior to March 1, 2016, in order to allow sufficient time for implementation of the revised TS requirements for application in the Callaway refueling outage (Refuel 21) scheduled to begin April 1, 2016. If the requested amendment is made effective upon issuance, to be implemented within 90 days (as typically specified), sufficient time will be allowed for implementation regardless of whether the amendment is issued on or well before March 1, 2016. Ameren Missouri further requests that the license amendment be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

The Callaway Onsite Review Committee has reviewed and approved the proposed changes and has approved the submittal of this amendment application. In addition, in accordance with 10 CFR 50.91 "Notice for public comment; State consultation," Section (b)(1), a copy of this amendment application is being provided to the designated Missouri State official.

This amendment request does not contain new commitments. If there are any questions regarding this submittal, please contact Mr. Tom Elwood at 314-225-1905.

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

Very truly yours,

Executed on: 3/9/2015



Scott Maglio
Manager, Regulatory Affairs

DRB/

Enclosure: Description and Assessment of the Proposed Change

Attachments to the Enclosure:

- 1 – Technical Specification Page Markups
- 2 – Technical Specification Bases Page Markups
- 3 – Retyped Technical Specification Pages

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DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

1.0 DESCRIPTION

The proposed change revises Limiting Condition for Operation (LCO) 3.4.17, "Steam Generator (SG) Tube Integrity," as well as Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report." The proposed changes are needed to address implementation issues associated with the inspection periods and to address other administrative changes and clarifications.

The proposed amendment is consistent with TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," as applicable to Steam Generators (SGs) with Alloy 690 thermally treated tubing.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Ameren Missouri has reviewed TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," as well as the model safety evaluation dated October 27, 2011 (76FR 66763) provided as part of the Federal Register Notice for Availability. As described in the subsequent paragraphs, Ameren Missouri has concluded that the justifications presented in TSTF-510 and the model safety evaluation prepared by the NRC staff are applicable to Ameren Missouri and justify the requested amendment for incorporation of the applicable changes to the Callaway Plant Technical Specifications (TSs).

2.2 Optional Changes and Variations

The proposed changes include correction of an administrative inconsistency in TSTF-510, Paragraph d.2 of the Steam Generator Tube Inspection Program. In Section 2.0, "Proposed Change," TSTF-510 states that references to "tube repair criteria" in Paragraph d are revised to "tube plugging [or repair] criteria." However, in the following sentence in Paragraph d.2, this change was inadvertently omitted, "If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated." Ameren Missouri does not have approved tube repair criteria. Therefore, the noted sentence is revised to state "tube plugging" criteria.

The above correction is consistent with use of the phrase "tube plugging criteria" (as opposed to "tube plugging [or repair] criteria") throughout the proposed TS changes for the Callaway Plant, since as noted, Ameren Missouri does not have approved tube repair criteria.

It should also be noted that due to the existing differences in the numbering of Callaway Plant's TSs and the numbering of the Standard TSs (NUREG-1431), the Callaway Plant TSs to

be revised in accordance with TSTF-510 are numbered differently than those shown in TSTF-510 itself. Specifically, the Callaway Plant TS Limiting Condition for Operation (LCO) titled "Steam Generator (SG) Tube Integrity" is numbered "3.4.17" instead of "3.4.20" (as the TS is numbered in NUREG-1431). Likewise, the Callaway Plant TS titled "Steam Generator Tube Inspection Report" is numbered "5.6.10" instead of "5.6.7" (as the TS is numbered in NUREG-1431). These differences are administrative and do not affect the applicability of TSTF-510 to the Callaway Plant TSs.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

Ameren Missouri requests adoption of an approved change to the Standard TSs into the plant specific TSs, in order to revise Technical TS 5.5.9, "Steam Generator (SG) Program"; TS 5.6.10, "Steam Generator Tube Inspection Report"; and Limiting Condition for Operation (LCO) 3.4.17, "Steam Generator Tube Integrity"; to address inspection periods and other administrative changes and clarifications.

As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

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

Response: No.

The proposed change revises the Steam Generator (SG) Program to modify the frequency of verification of SG tube integrity and SG tube sample selection. A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of the plant's licensing basis. The proposed SG tube inspection frequency and sample selection criteria will continue to ensure that the SG tubes are inspected such that the probability of a SGTR is not increased. The consequences of a SGTR are bounded by the conservative assumptions in the design basis accident analysis. The proposed change will not cause the consequences of a SGTR to exceed those assumptions. Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes to the Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The proposed change does not affect the design of the SGs or their method of operation. In addition, the proposed change does not impact any other plant system or component.

Therefore, it is concluded that this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. These safety functions are maintained by ensuring integrity of the SG tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change will continue to require monitoring of the physical condition of the SG tubes such that there will not be a reduction in the margin of safety compared to the current requirements. Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Ameren Missouri concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

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ATTACHMENT 1
TECHNICAL SPECIFICATION PAGE MARKUPS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube ~~repair~~ ^{plugging} criteria shall be plugged in accordance with Steam Generator Program.

[Callaway is not licensed for any alternate repair criteria.]

APPLICABILITY: MODES 1 2, 3, and 4.

ACTIONS

NOTE

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair ^{plugging} criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or inspection.	7 days
	<u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

plugging

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

- a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

<u>ASME OM Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities, and
- d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
 1. Structural integrity performance criterion: All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown) and all anticipated transients included in the design specification and design basis accidents. This includes retaining a safety factor of 3.0 (3DP) against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total for all four steam generators.
 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

c. Provisions for SG tube ~~repair~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube ~~repair~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An ~~assessment~~ of degradation *assessment* shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG ~~replacement~~ *installation*.

*INSERT
5.5.9.d.2*

2. ~~Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~

3. If crack indications *affected and potentially affected* are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is ~~less~~). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

Amendment 187 |
results in more frequent inspections

INSERT 5.5.9.d.2

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.
 - a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
 - b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
 - c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
 - d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG;
- b. ~~Active~~ degradation mechanisms found;
- c. Nondestructive examination techniques utilized for each degradation mechanism;
- d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;
- e. Number of tubes plugged during the inspection outage for each ~~active~~ degradation mechanism;
- f. ~~Total number and percentage of tubes plugged to date; and~~
- g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator; and

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ATTACHMENT 2
TECHNICAL SPECIFICATION BASES PAGE MARKUPS

BASES (Continued)

APPLICABLE
SAFETY
ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a primary to secondary LEAKAGE rate of 1 gpm to the unaffected steam generators, exceeding the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via a postulated stuck-open atmospheric steam dump (ASD) valve or via a partially stuck-open main steam safety valve (see Ref. 2).

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 3), 10 CFR 100 (Ref. 4) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the ~~repair~~ criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ criteria is removed from service by plugging. If a tube was determined to satisfy the ~~repair~~ criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

plugging

(continued)

BASES

LCO
(continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm total for all four steam generators. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the

plugging

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG ~~repair~~ ^{plugging} criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

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

(continued)

BASES (Continued)SURVEILLANCE
REQUIREMENTSSR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the ~~tube repair~~ ^{plugging} criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 7). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. *INSERT SR 3.4.17.1 Bases*

SR 3.4.17.2

During a SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ ^{plugging} criteria is removed from service by plugging. The tube ~~repair~~ ^{plugging} criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size

(continued)

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INSERT SR 3.4.17.1 Bases

If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.9 until subsequent inspections support extending the inspection interval.

BASES

**SURVEILLANCE
REQUIREMENTS****SR 3.4.17.2 (continued)**

measurement and for future flaw growth. In addition, the tube ~~repair~~ *plugging* criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of *plugging* prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the ~~repair~~ criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
 2. FSAR Section 15.6.3.
 3. 10 CFR 50 Appendix A, GDC 19.
 4. 10 CFR 100.
 5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
 6. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
 7. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines."
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ATTACHMENT 3
RETYPE TECHNICAL SPECIFICATION PAGES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with Steam Generator Program.

APPLICABILITY: MODES 1 2, 3, and 4.

ACTIONS

----- NOTE -----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.</p>	<p>A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or inspection.</p>	<p>7 days</p>
	<p><u>AND</u></p> <p>A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.</p>	

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

- a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

<u>ASME OM Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of **SR 3.0.2** are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;
- c. The provisions of **SR 3.0.3** are applicable to inservice testing activities; and
- d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
 1. Structural integrity performance criterion: All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 (3DP) against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total for all four steam generators.
 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

- c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. Degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
 - 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

 - 2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

- (a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
- (b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
- (c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
- (d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

- 3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

- e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Rev. 2, and uses the test procedure guidance in Regulatory Guide 1.52, Revision 2, Positions C.5.a, C.5.c and C.5.d.

- a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

- c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Filtration	2.0%	70%
Control Room Pressurization	2.0%	70%
Emergency Exhaust System	2.0%	70%

- d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration	5.4" WG	2000 cfm, ± 200 cfm
Control Room Pressurization	5.4" WG	500 cfm, +500, - 50 cfm
Emergency Exhaust System	5.4" WG	9000 cfm, ± 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI 510-1975 and corrected to design nameplate voltage settings.

ESF Ventilation System	Wattage
Control Room Pressurization	15 ± 2 KW
Emergency Exhaust System	37 ± 3 KW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure, Revision 0". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures, Revision 2".

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);
- b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and
- c. A surveillance program to ensure that the quantity of radioactivity contained in the outdoor liquid radwaste tanks listed below that are not

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste System is less than the quantities determined in accordance with the Standard Review Plan, Section 15.7.3:

- a. Reactor Makeup Water Storage Tank,
- b. Refueling Water Storage Tank,
- c. Condensate Storage Tank, and
- d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of **SR 3.0.2** and **SR 3.0.3** are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 1. an API gravity or an absolute specific gravity within limits,
 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 3. a water and sediment content within limits for ASTM 2D fuel oil.
- b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition of new fuel oil to storage tanks; and
- c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 31 days based on applicable ASTM D-2276 standards.
- d. The provisions of **SR 3.0.2** and **SR 3.0.3** are applicable to the Diesel Fuel Oil Testing Program test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
 1. a change in the TS incorporated in the license; or
 2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

- a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
 - 1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

(continued)

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.
 3. The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during the Refuel 14 outage (fall of 2005).
 4. The first Type A test performed after the October 26, 1999 Type A test shall be performed no later than October 25, 2014.
- b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 48.1 psig.
 - c. The maximum allowable containment leakage rate, L_a , at P_a , shall be 0.20% of the containment air weight per day.
 - d. Leakage rate acceptance criteria are:
 1. Containment leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60 L_a$ for the Type B and C tests and $\leq 0.75 L_a$ for Type A tests;
 2. Air lock testing acceptance criteria are:
 - a) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$;
 - b) For each door, leakage rate is $\leq 0.005 L_a$ when pressurized to ≥ 10 psig.
 - e. The provisions of Technical Specification **SR 3.0.2** do not apply to the test frequencies in the Containment Leakage Rate Testing Program.
 - f. The provisions of Technical Specification **SR 3.0.3** are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals (continued)

5.5.17 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

- a. The definition of the CRE, CRE boundary, control building envelope (CBE), and the CBE Boundary.
- b. Requirements for maintaining the CRE and CBE boundaries in their design condition, including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air inleakage past the CRE and CBE boundaries in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following exception is taken to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

1. The Tracer Gas Test based on the Brookhaven National Laboratory Atmospheric Tracer Depletion (ATD) Method is used to determine the unfiltered air inleakage past the CRE and CBE boundaries. The ATD Method is described in AmerenUE letters dated December 15, 2004 (ULNRC-05104), June 6, 2006 (ULNRC-05298), July 16, 2007 (ULNRC-05427), and October 30, 2007 (ULNRC-05448).
- d. Measurement, at designated locations, of the CRE pressure relative to the outside atmosphere during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

(continued)

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program (continued)

- e. The quantitative limits on unfiltered air leakage into CRE and CBE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air leakage measured by the testing described in paragraph c. The unfiltered air leakage limit for radiological challenges is the leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air leakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.
- f. The provisions of **SR 3.0.2** are applicable to the Frequencies for assessing CRE habitability, determining CRE and CBE unfiltered leakage, and measuring CRE pressure and assessing CRE and CBE as required by paragraphs c and d, respectively.

5.5.18 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
 - b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
 - c. The provisions of Surveillance Requirements **3.0.2** and **3.0.3** are applicable to the Frequencies established in the Surveillance Frequency Control Program.
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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period.

The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Not used.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
1. Moderator Temperature Coefficient limits in **Specification 3.1.3**,
 2. Shutdown Bank Insertion Limit for **Specification 3.1.5**,
 3. Control Bank Insertion Limits for **Specification 3.1.6**,
 4. Axial Flux Difference Limits for **Specification 3.2.3**,
 5. Heat Flux Hot Channel Factor, $F_Q(Z)$, F_Q^{RTP} , $K(Z)$, $W(Z)$ and F_Q Penalty Factors for **Specification 3.2.1**,
 6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}^N$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for **Specification 3.2.2**,
 7. Shutdown Margin Limits for **Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8**,
 8. Reactor Core Safety Limits Figure for **Specification 2.1.1**,
 9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for **Specification 3.3.1**, and
 10. Reactor Coolant System Pressure and Temperature DNB Limits for **Specification 3.4.1**.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY."
 2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION."
 3. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE."

(continued)

5.6 Reporting Requirements

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."
 5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."
 6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."
 7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."
 8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."
 9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER DT AND THERMAL OVERTEMPERATURE DT TRIP FUNCTIONS."
 10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."
 11. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."
 12. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
 - 1. **Specification 3.4.3**, "RCS Pressure and Temperature (P/T) Limits," and
 - 2. **Specification 3.4.12**, "Cold Overpressure Mitigation System (COMS)."
- b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves".
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

5.6.8 PAM Report

When a report is required by **Condition B or F of LCO 3.3.3**, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not used.

(continued)

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the **Specification 5.5.9**, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG;
 - b. Degradation mechanisms found;
 - c. Nondestructive examination techniques utilized for each degradation mechanism;
 - d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;
 - e. Number of tubes plugged during the inspection outage for each active degradation mechanism;
 - f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator; and
 - g. The results of condition monitoring, including the results of tube pulls and in-situ testing.
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5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR Part 20:

- 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation:
- a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment;
 - b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
 - c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
 - d. Each individual or group entering such an area shall possess:
 1. A radiation monitoring device that continuously displays radiation dose rates in the area; or
 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose rate information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or
 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

- (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
 - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.
- e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:
 - 1. All such door and gate keys shall be maintained under the administrative control of the Shift Manager/Operating Supervisor or Radiation Protection Department Supervision, or his or her designee.
 - 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.
- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

(Continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
 - 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
 - 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
 - 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and
 - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
 - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or
 - 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably

(Continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

Achievable” principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

- e. Except for individual qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.
 - f. Such individual areas that are within a larger area, such as PWR containment, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.
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