VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10 CFR 50.90

July 30, 2012

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 12-486
NL&OS/ETS
Docket Nos. 50-338/339
License Nos. NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS TO ADOPT TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

Pursuant to 10 CFR 50.90, Dominion requests amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed amendment would modify 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."

Attachment 1 provides a description and assessment of the proposed changes including: the requested plant-specific licensing basis that demonstrates compliance with the 10 CFR 50, Appendix A General Design Criteria referenced in the Traveler as well as the plant-specific administrative variations from the TS changes described in TSTF-510, Revision 2. Attachment 2 provides the current TS pages marked up to show the proposed changes. Attachment 3 provides the proposed TS pages. Attachment 4 provides the marked-up TS Bases pages for information only.

The proposed amendment does not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92. The Facility Safety Review Committee has reviewed and concurred with the determinations herein.

Approval of the proposed amendment is requested by July 2013. Once approved, the amendment shall be implemented within 60 days.

A001
NRR

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Virginia.

If you have any questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

J. Alan Price
Vice President – Nuclear Engineering

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 30TH day of July, 2012.

My Commission Expires: May 31, 2014.

Vicki L. Hull
Notary Public

Commitments made in this letter: None

Attachments:
1. Description and Assessment
2. Marked-up Technical Specifications Pages
3. Proposed Technical Specification Pages
4. Marked-up Technical Specification Bases Pages

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Attachment 1

Discussion of Change

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 And 2

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

The proposed change revises Specification 5.5.8, "Steam Generator (SG) Program" and 5.6.7, "Steam Generator Tube Inspection Report." The proposed changes are needed to address implementation issues associated with the inspection periods, and address other administrative changes and clarifications. For consistency, additional administrative changes are being made to Specification 3.4.20 "Steam Generator (SG) Tube Integrity."

The proposed amendment is consistent with TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Dominion has reviewed TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," (ADAMS Accession No. ML110610350) and the model safety evaluation dated October 19, 2011 (ADAMS Accession No. ML112101513) as identified in the Federal Register Notice of Availability, dated October 27, 2011 (76 FR 66763). As described in the subsequent paragraphs, Dominion has concluded that the justifications presented in TSTF-510 and the model safety evaluation prepared by the NRC staff are applicable to North Anna Units 1 and 2 and justify this amendment for the incorporation of the changes to the North Anna Technical Specifications (TS).

2.2 Optional Changes and Variations

Dominion is not proposing any technical variations or deviations from the TS changes described in the TSTF-510, Revision 2, or the applicable parts of the NRC staff's model safety evaluation dated October 19, 2011. However, Dominion is proposing the following administrative variations from the TS changes described in the TSTF-510, Revision 2, or the applicable parts of the NRC staff's model safety evaluation dated October 19, 2011.

The North Anna TS numbering system is different than the Improved Technical Specifications (ITS) on which TSTF-510 was based. Specifically, the "Steam Generator (SG) Program" in the North Anna TS is numbered 5.5.8 rather than 5.5.9 and the Steam Generator (SG) Tube Integrity TS is numbered 3.4.20 rather than 3.4.17. These differences are administrative and do not affect the applicability of TSTF-510 to the North Anna TS.

The proposed change also corrects 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.2 is 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" (emphasis added).

The underlined phrase should state "tube plugging [or repair] criteria," consistent with the other changes made in TSTF-510. Dominion is changing the phrase to "tube plugging criteria." This change is administrative and should not result in this application being removed from the Consolidated Line Item Improvement Process.

This administrative error was indentified in a February NRC-TSTF meeting and documented in a letter from the TSTF to the NRC dated March 28, 2012 (TSTF letter No. 12-09).

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

Dominion requests adoption of an approved change to the standard technical specifications (STS) into the plant specific technical specifications (TS), to revise the Specification 5.5.8, "Steam Generator (SG) Program," 5.6.7, "Steam Generator Tube Inspection Report," and LCO 3.4.20, "Steam Generator (SG) 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 a 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. The proposed change to reporting requirements and clarifications of the existing requirements have no affect on the probability or consequences of SGTR.

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. In summary, the safety function of a SG is maintained by ensuring the integrity of its 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, Dominion 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.

3.2 Applicable Regulatory Requirements/Criteria

During the initial plant licensing of North Anna Power Station Units 1 and 2, it was demonstrated that the design of the reactor coolant pressure boundary met the regulatory requirements in place at that time. North Anna Power Station, Units 1 and 2,

were issued construction permits in February 1971, based on the station design being in conformance with the *General Design Criteria for Nuclear Power Plants,* published in 1966. The General Design Criteria (GDC) included in Appendix A to 10 CFR Part 50 did not become effective until February, 1971. The units were not subject to GDC requirements (Reference SECY-92-223 dated September 18, 1992.) However, the following information demonstrates compliance with GDC 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. Specifically, Section 3.1.1 of the UFSAR discusses the design of the station relative to the design criteria published in 1971. The GDC state that the Reactor Coolant Pressure Boundary (RCPB) shall have "an extremely low probability of abnormal leakage . . . and gross rupture" (GDC 14/UFSAR 3.1.10), "shall be designed with sufficient margin" (GDCs 15/UFSAR 3.1.10 and 31/ UFSAR 3.1.27), shall be of "the highest quality standards practical" (GDC 30/UFSAR 3.1.26), and shall be designed to permit "periodic inspection and testing . . . to assess . . . structural and leak tight integrity" (GDC 32/ UFSAR 3.1.28). Structural integrity refers to maintaining adequate margins against burst, and collapse of the SG tubing. There are no changes to the SG design that impact these general design criteria.

The TS plugging limits ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions. The reactor coolant pressure boundary is designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. Reactor coolant pressure boundary components have provisions for the inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. Structural integrity refers to maintaining adequate margins against burst, and collapse of the SG tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits during all plant conditions.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction, and operation of safety related components. The pertinent requirements of this appendix apply to activities affecting the safety related functions of these components. These requirements are described in Criteria IX, XI, and XVI of Appendix B and include control of special processes, inspection, testing, and corrective action.

Under 10 CFR 50.65, the Maintenance Rule, licensees classify SGs as risk significant components because they are relied upon to remain functional during and after design basis events. SGs are to be monitored under 10 CFR 50.65(a)(2) against industry established performance criteria. Meeting the performance criteria of NEI 97-06, Revision 3, provides reasonable assurance that the SG tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary.

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.

Attachment 2

Marked-up Technical Specifications Pages

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 And 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.20 Steam Generator (SG) Tube Integrity

LCO 3.4.20 SG tube integrity shall be maintained.

<u>AND</u>

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

plugging

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~~ 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 SG tube 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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.20.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.20.2	Verify that each inspected SG tube that satisfies the tube ~~repair~~ 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

5.5.7 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 specified in the ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda as follows:

ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
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 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 Code for Operation and Maintenance of Nuclear Power Plants shall be construed to supersede the requirements of any TS.

5.5.8 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 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

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

a. (continued)

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 in-service 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 cool down ~~and~~ all anticipated transients included in the design specifications and design basis accidents. This includes retaining ~~a~~ safety factor of 3.0 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 for all SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

5.5 Programs and Manuals

plugging

5.5.8 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~~ 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.

degradation

installation

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

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.~~

Replace with Insert A

3. If crack indications 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.

affected or potentially affected

results in more frequent inspections

Insert A for TS 5.5.8 - SG Program - (690 tubes)

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 and that may satisfy the applicable plugging criteria, 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

5.6.7 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 5.5.8, "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, and

f. ~~Total number and percentage of tubes plugged to date,~~ The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing~~, and~~

~~h. The effective plugging percentage for all plugging in each SG.~~

Attachment 3

Proposed Technical Specification Pages

**Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 And 2**

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.20 Steam Generator (SG) Tube Integrity

LCO 3.4.20 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the 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
A. One or more SG tubes satisfying the tube 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 SG tube inspection.	7 days
	AND	
	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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.20.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.20.2	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

5.5.7 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 specified in the ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda as follows:

ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
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 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 Code for Operation and Maintenance of Nuclear Power Plants shall be construed to supersede the requirements of any TS.

5.5.8 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 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

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

a. (continued)

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 in-service 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 cool down), and all anticipated transients included in the design specification, design basis accidents. This includes retaining a safety factor of 3.0 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 for all SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

5.5 Programs and Manuals

5.5.8 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. A 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 and that may satisfy the applicable plugging criteria, 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

(continued)

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. (continued)

   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.

3. If crack indications are found in any SG tube, then the next inspection for each affected or 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.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

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.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in general conformance with the frequencies and requirements of Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance

(continued)

5.5 Programs and Manual

5.5.10 Ventilation Filter Testing Program (VFTP)

a. (continued)

with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)	1000 ± 10% cfm
Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)	Nominal accident flow for a single train actuation

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

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 in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
MCR/ESGR EVS	1000 ± 10% cfm
ECCS PREACS	Nominal accident flow for a single train actuation

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than the

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

c. (continued)

value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and relative humidity specified below.

ESF Ventilation System	Penetration	RH
MCR/ESGR EVS	2.5%	70%
ECCS PREACS	5%	70%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
MCR/ESGR EVS	4 inches W.G.	1000 ± 10% cfm
ECCS PREACS	5 inches W.G.	≤ 39,200 cfm

e. Demonstrate that the heaters for each of the ESF systems dissipate ≥ the value specified below when tested in accordance with ASME N510-1975.

ESF Ventilation System	Wattage
MCR/ESGR EVS	3.5 kW

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

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste 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

(continued)

5.5 Programs and Manuals

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

Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste 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 each of the following outdoor tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains liquid radwaste ion exchanger system is less than the amount that would result in concentrations greater than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, excluding tritium, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents:

    1. Refueling Water Storage Tank;

    2. Casing Cooling Storage Tank;

    3. PG Water Storage Tank;

    4. Boron Recovery Test Tank; and

    5. Any Outside Temporary Tank.

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 Programs and Manuals

5.5.12 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. water and sediment ≤ 0.05%.

b. Within 31 days following addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days; and

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing Frequencies.

5.5.13 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

(continued)

5.5 Programs and Manuals

5.5.13 Technical Specifications (TS) Bases Control Program (continued)

b. (continued)

2. a change to the UFSAR 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 UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13b 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).

5.5.14 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;

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, no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident

(continued)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exception:

   NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the October 9, 1999 Type A test shall be performed no later than October 9, 2014.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.7 psig. The containment design pressure is 45 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

(continued)

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

1. Prior to entering a MODE where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

   $\leq 0.60\ L_a$ for the Type B and Type C tests on a Maximum Path Basis and $\leq 0.75\ L_a$ for Type A tests.

   During operation where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

   $\leq 1.0\ L_a$ for overall containment leakage rate and $\leq 0.60\ L_a$ for the Type B and Type C tests on a Minimum Path Basis.

2. Overall air lock leakage rate testing acceptance criterion is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.16 Main Control Room/Emergency Switchgear Room (MCR/ESGR) Envelope Habitability Program

A MCR/ESGR Envelope Habitability Program shall be established and implemented to ensure that MCR/ESGR envelope habitability is maintained such that, with an OPERABLE MCR/ESGR EVS, MCR/ESGR envelope 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 MCR/ESGR envelope under design basis accident conditions without

(continued)

5.5 Programs and Manuals

5.5.16 Main Control Room/Emergency Switchgear Room Envelope Habitability Program (MCR/ESGR) (continued)

personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent for the duration of the accident. The program shall include the following elements:

a. The definition of the MCR/ESGR envelope and the MCR/ESGR envelope boundary.

b. Requirements for maintaining the MCR/ESGR envelope boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the MCR/ESGR envelope into the MCR/ESGR envelope 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 MCR/ESGR envelope habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

   The following is an exception to Section C.2 of Regulatory Guide 1.197, Revision 0:

   - 2.C.1 Licensing Bases - Vulnerability assessments for radiological, hazardous chemical and smoke, and emergency ventilation system testing were completed as documented in the UFSAR. The exceptions to the Regulatory Guides (RG) referenced in RG 1.196 (i.e., RG 1.52, RG 1.78, and RG 1.183), which were considered in completing the vulnerability assessments, are documented in the UFSAR/current licensing basis. Compliance with these RGs is consistent with the current licensing basis as described in the UFSAR.

d. Measurement, at designated locations, of the MCR/ESGR envelope pressure relative to all external areas adjacent to the MCR/ESGR envelope boundary during the pressurization mode of operation by one train of the MCR/ESGR EVS, 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 assessment of the MCR/ESGR envelope boundary.

(continued)

5.5 Programs and Manuals

5.5.16 Main Control Room/Emergency Switchgear Room Envelope Habitability Program (MCR/ESGR) (continued)

e. The quantitative limits on unfiltered air inleakage into the MCR/ESGR envelope. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of design basis accident consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of MCR/ESGR envelope 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 MCR/ESGR envelope habitability, determining MCR/ESGR envelope unfiltered inleakage, and measuring MCR/ESGR envelope pressure and assessing the MCR/ESGR envelope boundary as required by paragraphs c and d, respectively.

5.5.17 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specification are performed at interval 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.

5.6 Reporting Requirements

5.6.7 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 5.5.8, "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 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.

Attachment 4

Marked-up Technical Specification Bases Pages

(For Information Only)

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 And 2

BASES

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.

plugging

During an ~~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.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant

(continued)

BASES

APPLICABILITY (continued)

SG integrity limits are not provided in MODES 5 and 6 since RCS conditions are far less challenging than 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 separate Conditions entry is permitted 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~~ plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.20.2. An evaluation of SG tube integrity of the 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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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.

SURVEILLANCE REQUIREMENTS

SR 3.4.20.1

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.20.1 (continued)

function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.20.1. 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 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. Insert B

SR 3.4.20.2

plugging

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ criteria is removed from service by plugging. The tube ~~repair~~ criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube ~~repair~~ 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 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. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.

Insert B for TS Bases - Surveillance Requirements

If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.8 until subsequent inspections support extending the inspection interval.