**Enclosure 4 of this letter contains proprietary information and should be withheld from public disclosure under 10 CFR 2.390**

**10 CFR 50.90**

Palo Verde Nuclear Generating Station

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102-05276-CDM/TNW/GAM
May 26, 2005

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

Dear Sirs:

**Subject: Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3
Docket Nos. STN 50-528, 50-529, and 50-530
Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity and Steam Generator Tube Inspection Length through the Tubesheet**

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) hereby requests to amend Operating Licenses NPF-41, NPF-51, and NPF-74 for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, respectively.

The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator (SG) tube integrity. The proposed TSs are consistent with those in NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

The proposed amendment also includes changes to the revised SG program in TS Section 5.5.9 to specify the SG tube inspection length through the SG tubesheet and establish plugging criteria in the inspected tubesheet region for the remaining original SGs containing Alloy 600 mill annealed (MA) tubes. This change is being proposed to establish conformance with the NRC position identified in Generic Letter (GL) 2004-01, as committed by APS in letter no. 102-05171 to NRC dated October 28, 2004. The



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proposed inspection length is based on the results of Westinghouse Electric Corporation analysis and testing described in report WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions" (Enclosure 4). Westinghouse has determined that this report contains proprietary information and requests that it be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390(a)(4). Enclosure 3 contains an affidavit from Westinghouse stating the reasons that WCAP-16208-P should be considered a proprietary document. A redacted, non-proprietary version of this report, WCAP-16208-NP, is provided as Enclosure 5.

The proposed changes specify inspection requirements both for SGs containing Alloy 600 MA tubes and SGs containing Alloy 690 thermally treated (TT) tubes. The SGs in Units 1 and 3 containing Alloy 600 MA tubes are scheduled to be replaced with Alloy 690 TT-tube SGs in Fall 2005 (Unit 1) and Fall 2007 (Unit 3). The Unit 2 SGs were replaced with Alloy 690 TT-tube SGs in fall 2003. Therefore, the only remaining Alloy 600 MA-tube SG inspection will be the Unit 3 SG inspection during its spring 2006 refueling outage. Since this proposed amendment specifies inspection requirements for SGs of specific tube material, and not by unit number, it can be implemented at one time for all three PVNGS units. Each unit will comply with the inspection requirements for the specific SG tube material type at the time of the inspection. Following completion of SG replacement in all three units (after Fall 2007), APS intends to submit an amendment request to delete the inspection requirements for Alloy 600 MA-tube SGs.

Enclosure 1 provides a notarized affidavit required by 10 CFR 50.30. Enclosure 2 provides a description of the proposed changes. Attachment 2A provides the existing PVNGS TS pages marked-up to show the proposed changes.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10CFR 50.91(b)(1).

APS requests approval of the proposed amendment by February 28, 2006, just prior to the Unit 3 refueling inspection when the last Alloy 600 MA SG tubes at PVNGS will be inspected before replacement. Also, an implementation period of 150 days is requested.

No commitments are being made to the NRC by this letter. If you have any questions, please contact Thomas N. Weber at (623) 393-5764.

Sincerely,

David Mauldin

CDM/TNW/GAM

Enclosures:

1. Notarized affidavit
2. APS' evaluation of the proposed changes
   Attachment 2A  Proposed Technical Specification changes (mark-up)
   Attachment 2B  Proposed Technical Specification changes (retyped)
   Attachment 2C  Proposed Technical Specification Bases changes (mark-up)
3. Westinghouse affidavit requesting that WCAP-16208-P, Revision 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390(a)(4)
4. WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions" (Proprietary)
5. WCAP-16208-NP, Revision 1 (Non-proprietary)

| | | |
|---|---|---|
| cc: | B. S. Mallett | NRC Region IV Regional Administrator |
| | M. B. Fields | NRC NRR Project Manager |
| | G. G. Warnik | NRC Senior Resident Inspector for PVNGS |
| | A. V. Godwin | Arizona Radiation Regulatory Agency (ARRA) |

## ENCLOSURE 1

## AFFIDAVIT

STATE OF ARIZONA )
) ss.
COUNTY OF MARICOPA )

I, David Mauldin, represent that I am Vice President, Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 26th Day Of May, 2005.

Susie Lynn Ergish

Notary Public

SUSIE LYNN ERGISH
Notary Public - Arizona
Maricopa County
My Comm. Expires Jul 14, 2007

Notary Commission Stamp

## ENCLOSURE 2
## ARIZONA PUBLIC SERVICE COMPANY'S EVALUATION

Subject: Application for Technical Specification Improvement Regarding Steam Generator (SG) Tube Integrity and SG Tube Inspection Length through the Tubesheet

## 1.0 INTRODUCTION

This proposed license amendment is comprised of two parts:

1. Incorporate steam generator (SG) integrity technical specification (TS) changes associated with the Consolidated Line Item Improvement Process (CLIIP) for TSTF-449, Revision 4; and

2. Proposed changes in TS Section 5.5.9 to specify the SG tube inspection length through the SG tubesheet and establish plugging criteria in the inspected tubesheet region for the remaining original SGs containing Alloy 600 mill annealed (MA) tubes to establish conformance with the NRC position identified in Generic Letter (GL) 2004-01, as committed by APS in letter no. 102-05171 to NRC dated October 28, 2004.

These two parts of the proposed amendment are discussed in separate subsections in this Enclosure.

### 1.1 Introduction: CLIIP Changes

This proposed license amendment revises the requirements in Technical Specifications (TS) related to steam generator tube integrity. The proposed TSs are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this technical specification improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

The proposed revised TSs are consistent with those in TSTF-449, Rev 4. However, since the current PVNGS Units 1, 2, and 3 TSs have variations from the current Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS; NUREG-1432, Revision 3), which is the starting point for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs. The variations to the changes in TSTF-449, Revision 4 are described in section 2.1.1 of this submittal.

### 1.2 Introduction: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

The proposed change to TS 5.5.9 includes changes to specify the SG tube inspection length through the SG tubesheet and establish plugging criteria in the inspected tubesheet region for the remaining original SGs containing Alloy 600 mill annealed (MA) tubes.

The SGs in Units 1 and 3 containing Alloy 600 MA tubes are scheduled to be replaced with Alloy 690 TT-tube SGs in Fall 2005 (Unit 1) and Fall 2007 (Unit 3). The Unit 2 SGs were replaced with Alloy 690 TT-tube SGs in fall 2003. Therefore, the only remaining Alloy 600 MA-tube SG inspection will be the Unit 3 SG inspection during its spring 2006 refueling outage. This proposed amendment can be implemented at one time for all three PVNGS units, and each unit will comply with the inspection requirements for the specific SG-type at the time of the inspection. Following completion of SG replacement in all three units (after Fall 2007), APS intends to submit an amendment request to delete the tubesheet-region inspection and plugging criteria for SGs with Alloy 600 MA tubes.

## 2.0 DESCRIPTION OF PROPOSED AMENDMENT

### 2.1 *Description of Proposed Amendment: CLIIP Changes*

Consistent with the NRC-approved Revision 4 of TSTF-449, the proposed TS changes include:

- Revised TS 1.1 definition of LEAKAGE
- Revised TS 3.4.14, "RCS [Reactor Coolant System] Operational Leakage"
- New TS 3.4.18, "Steam Generator (SG) Tube Integrity"
- Revised TS 5.5.9, "Steam Generator (SG) Program"
- Revised TS 5.6.8, "Steam Generator Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4 is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program (TS 5.5.14).

#### 2.1.1 Variations from the CLIIP

The proposed revised PVNGS TSs are consistent with the revised TSs in TSTF-449, Revision 4. However, since the current PVNGS Units 1, 2, and 3 TSs have differences from the latest revision (Rev. 3) of the Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS, NUREG-1432), which was used as the markup TSs for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs.

The proposed changes are consistent with TSTF-449, Revision 4, except for the following variations due to differences between the current PVNGS TSs and the ISTS used as the mark-up TSs in TSTF-449, and clarifications in the TS Bases:

- This proposed amendment does not include the TSTF-449 changes to the primary to secondary leakage limits in TS 3.4.14 since those changes have previously been made to the PVNGS TS. This submittal does not include the TSTF-449 changes to the Safety Analysis section in TS Bases B 3.4.14 since the current TS Bases reflects site-specific safety analysis. This submittal does include changes to the TS Bases B 3.4.14 LCO to be consistent with the Improved Standard Technical Specification Bases.

- This proposed amendment will delete Action C of TS 3.4.14 and its TS Bases to be consistent with the changes being made for TSTF-449. Action C is not in the CE Improved Standard Technical Specifications, NUREG-1432, but is unique to PVNGS. TS 3.4.14, Action C, which currently requires entering LCO 3.0.3 (plant shutdown and cooldown) when one or more SGs are inoperable, will not be needed because the new TS 3.4.18, Action B will accomplish the same action (plant shutdown and cooldown) when SG tube integrity is not maintained.

- This proposed amendment will revise the Notes in SR 3.4.14.1 to be consistent with the proposed Note to SR 3.4.14.2 and with the current revision of CE Improved Standard Technical Specifications, NUREG-1432. The changes to TS Bases B 3.4.14 for SRs 3.4.14.1 and 3.4.14.2 include discussions that are not in TSTF-449 but help clarify the purpose of the Notes.

- The option to sleeve tubes is being deleted from TS 5.5.9 in the proposed TS. There are no sleeves installed in the PVNGS SGs.

- The TS Bases changes for the new TS B 3.4.18 Applicable Safety Analysis reflect the site specific safety analysis for SG tube rupture.

## 2.2 Description of Proposed Amendment: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

The proposed change to TS 5.5.9 includes changes to specify the SG tube inspection length through the SG tubesheet and establish plugging criteria in the inspected tubesheet region for the remaining original SGs with Alloy 600 mill annealed (MA) tubes. This change is being proposed to establish conformance with the NRC position described in Generic Letter (GL) 2004-01, "Requirements for Steam Generator Tube Inspections."

The proposed inspection length in TS 5.5.9 is, "For the original SGs with Alloy 600 MA tubes, the section of tube from the tube-to-tubesheet weld to 12 inches below the bottom of the expansion transition for both the tube inlet and outlet may be excluded from inspection based on Westinghouse Report WCAP-16208-P, Revision 1, 'NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions.'" The proposed plugging criteria is, "For the original SGs with Alloy 600 MA tubes, all detected degradation in the portion of tube below the bottom of the expansion transition that is not excluded from inspection per the provisions of 5.5.9.d shall be removed from service."

## 3.0 BACKGROUND

### 3.1 Background: CLIIP Changes

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

### 3.2 Background: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

APS' review of NRC Generic Letter (GL) 2004-01, "Requirements for Steam Generator Tube Inspections," identified the need to clarify the TS to identify the extent of the tubes to be inspected within the thickness of the tubesheet for SGs with Alloy 600 mill annealed (MA) tubes. This proposed amendment includes a change to address that clarification.

## 4.0 REGULATORY REQUIREMENTS AND GUIDANCE

### 4.1 Regulatory Requirements and Guidance: CLIIP Changes

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126) the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

The proposed revised TSs are consistent with those in TSTF-449, Rev 4. However, since the current PVNGS Units 1, 2, and 3 TSs have variations from the current Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS, NUREG-1432), which is the starting point for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs. The variations are described in section 2.1.1 of this enclosure.

### 4.2 Regulatory Requirements and Guidance: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

NRC Generic Letter (GL) 2004-01, "Requirements for Steam Generator Tube Inspections," identified the NRC position regarding the need for TSs to specify the SG tube inspection length through the tubesheet for Alloy 600 MA tubes. This proposed change will establish conformance with the NRC position.

## 5.0 TECHNICAL ANALYSIS

### 5.1 Technical Analysis: CLIIP Changes

APS has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR

10298) as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. APS has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to PVNGS Units 1, 2, and 3 and justify this amendment for the incorporation of the changes to the PVNGS TS.

The proposed revised TSs are consistent with those in TSTF-449, Rev 4. However, since the current PVNGS Units 1, 2, and 3 TSs have variations from the current Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS, NUREG-1432), which is the starting point for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs. The variations are described in section 2.1.1 of this enclosure.

## 5.2 Technical Analysis: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

Westinghouse report WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions" (Enclosure 4), provides a technical basis for an inspection program within the tubesheet region of Alloy 600 MA-tube SGs in Combustion Engineering (CE) plants. Revision 0 of WCAP-16208-P was submitted to the NRC by Florida Power and Light Company (FPL) as a supporting enclosure to FPL letter L-2004-245, dated November 8, 2004, to amend the St. Lucie Unit 2 operating license to define the depth of the required SG tube inspections and clarify the plugging criteria in the tubesheet region. By letter dated March 31, 2005, FPL submitted responses to an NRC request for additional information that provided responses from Westinghouse answering questions regarding WCAP-16208-P, Revision 0. Westinghouse subsequently revised WCAP-16208-P, issued as Revision 1, to incorporate changes to reflect the responses to the NRC RAI submitted by FPL in their March 31, 2005 letter.

WCAP-16208-P, Revision 1 provides recommended inspection lengths in the tubesheet region based on results from a conservative test and analysis program to ensure that the requirements for tube structural and leakage integrity provided in NEI 97-06 are met. The WCAP-16208-P, Revision 1 recommended minimum inspection lengths below the tube-to-tubesheet joint expansion transition are 10.4 inches for PVNGS Unit 1 and 11.6 inches for Unit 3 (specified as Plants CE1 and CE3). The proposed inspection length requirement "from the tube-to-tubesheet weld to 12 inches below the bottom of the expansion transition" bounds the WCAP-16208-P recommended inspection lengths for both Unit 1 and Unit 3.

The WCAP-16208-P, Revision 1 report supports the conclusion that the surrounding tubesheet prevents tube rupture and provides resistance against axial pullout loads during normal and accident conditions. In addition, any primary-to-secondary leakage from tube degradation below the inspection length is an inconsequential contribution to the total leakage assumed for a steam line break accident and may be considered negligible. Consequently, any tube degradation that may go undetected below the inspection length would not affect structural or leakage margins.

The PVNGS Steam Generator Inspection Program requires a degradation assessment to determine susceptible areas of the tubing to be inspected and the appropriate eddy current techniques to detect and quantify the degradation within each area. Input data needed for the subsequent Condition Monitoring and Operational Assessment are considered as part of the tube integrity assessment required by PVNGS Steam Generator Inspection Program and satisfies the intent mandated in NEI 97-06, "Steam Generator Program Guidelines."

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Thus, structural integrity is maintained by the tubesheet constraint. However, a 360-degree circumferential crack or many axially oriented cracks could permit severing of the tube and tube pullout from the tubesheet under the axial forces on the tube from primary to secondary pressure differentials. However, Section 5 of WCAP-16208-P describes the testing that was performed to define the length of non-degraded tubing that is sufficient to compensate for the axial forces on the tube and thus prevent pullout. The operating conditions utilized in WCAP-16208-P included those at PVNGS (specified as Plants CE1 and CE3) and are summarized in Section 3 of the WCAP report.

Operating experience has demonstrated negligible normal operating leakage from primary water stress corrosion cracking (PWSCC) in expansion transitions. PWSCC in expansions in the tubesheet region would be even further leakage limited by the tight tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. The steam line break (SLB) accident conditions provide the most severe radiological hazards for postulated accidents involving loss of pressure or fluid in the secondary system. Section 3.2.3 in WCAP-16208-P provides the justification to neglect the total SLB leak rate contributed by cracks below the inspection length. Therefore, eddy current inspection in the area below the inspection length is not necessary to preclude normal operating or accident induced leakage.

Additional conservatism is shown in PVNGS operational assessments, which conservatively include postulated leakage in the inspection length without credit for the leak-limiting effect provided by the tubesheet.

## 6.0 REGULATORY ANALYSIS

### 6.1 Regulatory Analysis: CLIIP Changes

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

The proposed revised TSs are consistent with those in TSTF-449, Rev 4. However, since the current PVNGS Units 1, 2, and 3 TSs have variations from the current Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS, NUREG-1432), which is the starting point for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs. The variations are described in section 2.1.1 of this enclosure.

### 6.1.1 Verification and Commitments: CLIIP Changes

The following information is provided to support the NRC staff's review of this amendment application:

| Plant/Unit | **PVNGS Unit 1** | **PVNGS Unit 2** | **PVNGS Unit 3** |
|---|---|---|---|
| Steam Generator Model | CE System 80 | Replacement CE System 80 | CE System 80 |
| Effective full power years (EFPY) of service for currently installed SGs | 14.8 | 1.2 | 14.1 |
| Tubing material | Alloy 600 MA(1) | Alloy 690 TT | Alloy 600 MA(1) |
| Number of tubes per SG | 11,012 | 12,580 | 11,012 |
| Number and percentage of tubes plugged in each SG | SG 11: 873 (8%) tubes plugged;<br><br>SG 12: 1203 (11%) tubes plugged | SG 21: 19 (0.15%) tubes plugged;<br><br>SG 22: 20 (0.16%) tubes plugged | SG 31: 631 (6%) tubes plugged;<br><br>SG 32: 755 (7%) tubes plugged |
| Number of tubes repaired in each SG | None | None | None |
| Degradation mechanism identified | Upperbundle freespan axial ODSCC, ID and OD circumferential and axial cracking at tubesheet transition region, U-bend PWSCC, axial ODSCC at eggcrates and support wear | Support wear | Upperbundle freespan axial ODSCC, ID and OD circumferential and axial cracking at tubesheet transistion region, U-bend PWSCC, axial ODSCC at eggcrates and support wear |
| Current primary-to secondary leakage limits, evaluated at | 150 gpd per SG (TS 3.4.14) at room | 150 gpd per SG (TS 3.4.14) at | 150 gpd per SG (TS 3.4.14) at room |

| Plant/Unit | PVNGS Unit 1 | PVNGS Unit 2 | PVNGS Unit 3 |
|---|---|---|---|
| what temperature condition? | temperature | room temperature | temperature |
| Approved alternate tube repair criteria | None | None | None |
| Approved SG tube repair methods | This proposed amendment will delete the previously-approved tube repair method | This proposed amendment will delete the previously-approved tube repair method | This proposed amendment will delete the previously-approved tube repair method |
| Performance criteria for accident leakage | 0.5 gpm per SG | 0.5 gpm per SG | 0.5 gpm per SG |

(1) The SGs in Units 1 and 3 containing Alloy 600 MA tubes are scheduled to be replaced with Alloy 690 TT-tube SGs in Fall 2005 (Unit 1) and Fall 2007 (Unit 3). The only remaining Alloy 600 MA-tube SG inspection will be the Unit 3 SG inspection during its spring 2006 refueling outage.

### 6.2 Regulatory Analysis: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

NRC Generic Letter (GL) 2004-01, "Requirements for Steam Generator Tube Inspections," identified the NRC position regarding the need for TSs to specify the SG tube inspection length through the tubesheet for Alloy 600 MA tubes. This proposed change will establish conformance with the NRC position.

## 7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

### 7.1 No Significant Hazards Consideration: CLIIP Changes

APS has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. APS has concluded that the proposed determination presented in the notice is applicable to PVNGS Units 1, 2, and 3 and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a). APS is not proposing variations or deviations from the proposed no significant hazards consideration determination, except for the following:

> The PVNGS SG tube rupture analysis conservatively assumes a leakage of 1 gpm (1440 gpd) from the unaffected SG instead of the operational leakage limit (150 gpd) that is described in the model SE No Significant Hazards Consideration.

The proposed revised PVNGS TSs are consistent with those in TSTF-449, Rev 4. However, since the current PVNGS Units 1, 2, and 3 TSs have variations from the current Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS, NUREG-1432), which is the starting point for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs. The variations are described in section 2.1.1 of this enclosure.

### 7.2 No Significant Hazards Consideration: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

The proposed change to the proposed Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," would specify the SG tube inspection length through the SG tube sheet for the Palo Verde Nuclear Generating Station (PVNGS) Alloy 600 mill annealed (MA) tube SGs.

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

Response: No

The analysis that established the inspection length through the SG tube sheet for the PVNGS Alloy 600 MA-tube SGs took into account the reinforcing effect the tubesheet has on the external surface of an expanded SG tube. Tube-bundle integrity will not be adversely affected by the implementation of the revised tube inspection scope. SG tube burst or collapse cannot occur within the confines of the tubesheet; therefore, the tube burst and collapse criteria of draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are inherently met. Any degradation below the inspection length is shown by analyses and test results to be acceptable, thereby precluding an event with consequences similar to a postulated tube rupture event.

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Thus, structural integrity is maintained by the tubesheet constraint. However, a 360-degree circumferential crack or many axially oriented cracks could permit severing of the tube and tube pullout from the tubesheet under the axial forces on the tube from primary to secondary pressure differentials. Analysis and testing was performed to define the length of non-degraded tubing that is sufficient to compensate for the axial forces on the tube and thus prevent pullout. That length is bounded by the inspection length proposed in this change.

In conclusion, incorporation of the revised inspection scope into PVNGS TS maintains existing design limits and therefore, the proposed 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 performance based requirements are an improvement over the requirements imposed by the current TS.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary to secondary leakage that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

Tube-bundle integrity is expected to be maintained during all plant conditions upon implementation of the proposed tube inspection scope. Use of this scope does not introduce a new mechanism that would result in a different kind of accident from those previously analyzed. Even with the limiting circumstances of a complete circumferential separation of a tube occurring below the inspection length into the tubesheet, SG tube pullout is precluded and leakage is predicted to be maintained within the Updated Final Safety Analysis Report limits during all plant conditions.

Therefore, the proposed change does not create the possibility of a new or different type 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 is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TS.

Upon implementation of the revised inspection scope, operation with potential cracking below the Inspection Extent length in the expansion region of the SG tubing will meet the margin of safety as defined by Regulatory Guide (RG) 1.83 (Ref. 17), draft RG 1.121 (Ref. 11), and the requirements of General Design Criteria 14, 15, 31, and 32 of Appendix A to 10 CFR 50.

Therefore, the proposed change does not involve a significant reduction in a margin of safety and overall plant safety will be enhanced by the proposed changes.

## 8.0 ENVIRONMENTAL EVALUATION

### 8.1 Environmental Evaluation: CLIIP Changes

APS has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. APS has concluded that the staff's findings presented in that evaluation are applicable to PVNGS Units 1, 2, and 3 and the evaluation is hereby incorporated by reference for this application.

The proposed revised TSs are consistent with those in TSTF-449, Rev 4. However, since the current PVNGS Units 1, 2, and 3 TSs have variations from the current Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS, NUREG-1432), which is the starting point for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs.

### 8.2 Environmental Evaluation: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

A review has determined that the proposed amendment 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 amendment 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 amendment 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 amendment.

## 9.0 PRECEDENT

### 9.1 Precedent: CLIIP Changes

This application is being made in accordance with the CLIIP. APS is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4. APS is not proposing variations or deviations from the NRC staff's model SE published on March 2, 2005 (70 FR 10298), except for the following:

> The PVNGS SG tube rupture analysis conservatively assumes a leakage of 1 gpm (1440 gpd) from the unaffected SG instead of the operational leakage limit (150 gpd) that is described in the No Significant Hazards Consideration discussion in the model SE.

The proposed revised TSs are consistent with those in TSTF-449, Rev 4. However, since the current PVNGS Units 1, 2, and 3 TSs have variations from the current Combustion Engineering (CE) Improved Standard Technical Specifications (ISTS, NUREG-1432), which is the starting point for TSTF-449, there are some variations to the *changes* that are needed to incorporate TSTF-449 in the PVNGS TSs.

### 9.2 Precedent: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

The proposed change to the SG Program in TS 5.5.9 includes changes to specify the SG tube inspection length through the SG tubesheet and establish plugging criteria in the inspected tubesheet region for the remaining original SGs with Alloy 600 MA tubes. This change is being proposed to establish conformance with the NRC position described in Generic Letter (GL) 2004-01, "Requirements for Steam Generator Tube Inspections. "

This proposed change is similar to a proposed amendment submitted to the NRC by Florida Power and Light Company (FPL) in FPL letter L-2004-245, dated November 8, 2004, to amend the St. Lucie Unit 2 operating license to define the depth of the required SG tube inspections and clarify the plugging criteria in the tubesheet region. This FPL amendment request included WCAP-16208-P, Revision 0 as a supporting enclosure. By letter dated March 31, 2005, FPL submitted responses to an NRC request for additional information that provided responses from Westinghouse answering questions regarding WCAP-16208-P, Revision 0. Westinghouse subsequently revised WCAP-16208-P, issued as Revision 1, to incorporate changes to reflect the responses to the NRC RAI submitted by FPL in their March 31, 2005 letter.

## 10.0 REFERENCES

### 10.1 References: CLIIP Changes

Federal Register Notices:

> Notice for Comment published on March 2, 2005 (70 CFR 10298)
>
> Notice of Availability published on May 6, 2005 (70 FR 24126)

## 10.2 References: SG Tube Inspection Length through the Tubesheet (Alloy 600 Tubes)

1. NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections."

2. Westinghouse report WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions," dated May 2005.

Attachment 2A

Proposed Technical Specification Changes (mark-up)

Pages:

ii
1.1-4 and 5
3.4.14-1 and 2
3.4.18-1 and 2 (new)
5.5-6 through 5.5-24
5.6-6

# PALO VERDE NUCLEAR GENERATING STATION
# IMPROVED TECHNICAL SPECIFICATIONS
# TABLE OF CONTENTS

1.1 Definitions (continued)

| | |
|---|---|
| ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME | The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC. |
| $K_{n-1}$ | $K_{n-1}$ is the K effective calculated by considering the actual CEA configuration and assuming that the fully or partially inserted full strength CEA of highest worth is fully withdrawn. |
| LEAKAGE | LEAKAGE shall be:<br>a. <u>Identified LEAKAGE</u><br>1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;<br>2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or<br>3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE). |

(continued)

## 1.1 Definitions

**LEAKAGE (continued)**

b. Unidentified LEAKAGE

All LEAKAGE that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE   primary to secondary LEAKAGE

LEAKAGE (except ~~SG LEAKAGE~~) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

**MODE**

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, cold leg reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

**NEUTRON RATED THERMAL POWER (NRTP)**

The indicated neutron flux at RTP.

**OPERABLE - OPERABILITY**

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

**PHYSICS TESTS**

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14, Initial Test Program of the UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

(continued)

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.14 RCS Operational LEAKAGE

LCO 3.4.14 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one ~~SG~~ steam generator (SG)

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE. | A.1 Reduce LEAKAGE to within limits. | 4 hours |
| B. Required Action and associated Completion Time of Condition A not met.<br><br><u>OR</u><br><br>Pressure boundary LEAKAGE exists.<br><br><u>OR</u><br><br>Primary to secondary LEAKAGE not within limit. | B.1 Be in MODE 3.<br><br><u>AND</u><br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |
| ~~C. One or more SGs inoperable.~~ | ~~C.1 Enter LCO 3.0.3.~~ | ~~Immediately~~ |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.14.1 | after establishment<br>-------------------NOTE----------------- (5)<br>1. Not required to be performed ~~in MODE 3 or 4~~ until 12 hours of steady state operation.<br>2. Not applicable to primary to secondary LEAKAGE.<br>----------------------------------------<br>Perform RCS water inventory balance. | ~~-----NOTE------ Only required to be performed during steady state operation~~<br>72 hours |
| SR 3.4.14.2 | ~~Verify SG tube integrity is in accordance with the Steam Generator Tube Surveillance Program.~~<br>Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.<br>------------NOTE-------------<br>Not required to be performed until 12 hours after establishment of steady state operation<br>----------------------------- | ~~In accordance with the Steam Generator Tube Surveillance Program~~<br>72 hours |

**INSERT NEW TS 3.4.18**

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair 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 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 |
| | 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. |
| B. Required Action and associated Completion Time of Condition A not met.<br>OR<br>SG tube integrity not maintained. | B.1 Be in MODE 3. | 6 hours |
| | AND | |
| | B.2 Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.18.1 | Verify SG tube integrity in accordance with the Steam Generator Program. | In accordance with the Steam Generator Program. |
| SR 3.4.18.2 | 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. |

Replace with INSERT 5.5.9

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the Inservice Inspection of steam generator tubes and tube sleeves to ensure that structural integrity of this portion of the RCS is maintained. The program shall include the following:

5.5.9.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting tubes and tube sleeves in at least the minimum number of steam generators specified in Table 5.5.9-1.

5.5.9.2a Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20% and not sleeved in that area).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 5.5.9.4.a.10.) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:

(continued)

## INSERT 5.5.9 (sheet 1 of 3)

5.5.9 Steam Generator (SG) Program

A Steam Generator (SG) 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 an 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 and 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 SG 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) 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 0.5 gpm per SG and 1 gpm through both SGs.

## INSERT 5.5.9 continued (sheet 2 of 3)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

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. For the original SGs with Alloy 600 MA tubes, all detected degradation in the portion of tube below the bottom of the expansion transition that is not excluded from inspection per the provisions of 5.5.9.d shall be removed from service.

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. For the original SGs with Alloy 600 MA tubes, the section of tube from the tube-to-tubesheet weld to 12 inches below the bottom of the expansion transition for both the tube inlet and outlet may be excluded from inspection based on Westinghouse Report WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions." In addition to meeting the requirements of d.1, d.2a, d.2b, 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.

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

2a. Original SGs with Alloy 600 MA tubes: Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

2b. Replacement SGs with Alloy 690 TT tubes: 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

## INSERT 5.5.9 continued (sheet 3 of 3)

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 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.

5.5 Programs and Manuals (continued)

5.5.9.2a Steam Generator Tube Sample Selection and Inspection (continued)

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each inspection sample shall be classified into one of the following three categories (this classification shall apply to the inspection of tubes and treated exclusive of the sleeve inspections in 5.5.9.2b):

| Category | Inspection Results |
|---|---|
| C-1 | Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective. |
| C-2 | One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes. |
| C-3 | More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective. |

----------------------NOTE----------------------
In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.
------------------------------------------------

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2b <u>Steam Generator Tube Sleeve Sample Selection and Inspection</u> - The steam generator tube sleeve minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-3. The inservice inspection of steam generator tube sleeves shall be performed at the frequencies specified in 5.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.4. The tube sleeves selected for each inservice inspection shall include at least 20% of the total number of tube sleeves in all steam generators; the tube sleeves selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tube sleeves inspected shall be from these critical areas. Where the number of sleeves in the critical areas represents less than 50% of the initial sample, all sleeves in the critical areas shall be inspected.

b. The first sample of tube sleeves selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All tube sleeves that previously had detectable wall penetrations (greater than 20%).

   2. Tube sleeves in those areas where experience has indicated potential problems.

   3. A tube sleeve inspection (pursuant to Specification 5.5.9.4.a.8.) shall be performed on each selected tube sleeve. If any selected tube sleeve does not permit the passage of the eddy current probe for a tube sleeve inspection, this shall be recorded and an adjacent tube sleeve shall be selected and subjected to a tube sleeve inspection.

The results of each inspection sample shall be classified into one of the following three categories (this classification shall apply to the inspection of sleeves and treated exclusive of the tube inspections in 5.5.9.2a):

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2b Steam Generator Tube Sleeve Sample Selection and Inspection (continued)

| Category | Inspection Results |
| --- | --- |
| C-1 | Less than 5% of the total tube sleeves inspected are degraded tube sleeves and none of the inspected tube sleeves are defective. |
| C-2 | One or more tube sleeves, but not more than 1% of the total tube sleeves inspected are defective, or between 5% and 10% of the total tube sleeves inspected are degraded tube sleeves. |
| C-3 | More than 10% of the total tube sleeves inspected are degraded tube sleeves or more than 1% of the inspected tube sleeves are defective. |

-------------------------NOTE------------------
In all inspections, previously degraded tube sleeves must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.
-----------------------------------------------

5.5.9.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes and tube sleeves shall be performed at the following frequencies:

a. The first inservice inspection of the steam generator tubes shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.3 Inspection Frequencies (continued)

b. The inservice inspection of steam generator tube sleeves shall be performed at the following frequencies:

1. Steam generator tube sleeves shall be inspected prior to initial operation. The operating period before the initial inservice inspection shall not be shorter than six months nor longer than 24 months. The inspections of tube sleeves shall be configured to ensure that each individual tube sleeve is inspected at least once in 60 months.

2. If the results of the inservice inspection of steam generator tube sleeves conducted in accordance with Table 5.5.9-3 fall in category C-3, the inspection frequency shall be increased to ensure that each remaining tube sleeve is inspected at least once in 30 months. The increase in inspection frequency shall apply until the subsequent inspection satisfies the criteria for Category C-1.

c. If the results of the inservice inspection of a steam generator conducted in accordance with Tables 5.5.9-2 and 5.5.9-3 at 40 month intervals fall into Category C-3, the inspection frequency for the applicable tube or sleeve inspection shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.3.a (tubes) or 5.5.9.3.b.3 (sleeves); the interval may then be extended to a maximum of once per 40 months (tubes) or 30 months (sleeves).

d. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 5.5.9-2 and 5.5.9-3 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.

2. A seismic occurrence greater than the Operating Basis Earthquake.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.3 Inspection Frequencies (continued)

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

5.5.9.4 Acceptance Criteria

a. As used in this Specification

1. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

4. Degraded Tube or Sleeve means a tube or sleeve containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

5. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair imperfection depths specified below are in percentage of nominal wall thickness:

| | |
|---|---|
| a. Original tube wall | 40% |
| b. ABB-CE leak tight sleeve wall | 35% |

(continued)

5.5 Programs and Manuals (continued)

5.5.9.4 Acceptance Criteria (continued)

7. Preservice Inspection in the context of new steam generators means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

   Preservice Inspection for steam generator tubes repaired by tube sleeving means an inspection of the full length of the pressure boundary portion of the sleeved area performed by eddy current techniques prior to service to establish a baseline condition of the sleeved area. The sleeved area includes the pressure retaining portions of the parent tube in contact with the sleeve, the sleeve-to-tube weld and the pressure retaining portion of the sleeve.

8. Sleeve Inspection for sleeves selected in accordance with table 5.5.9-3 means an inspection of the sleeved area, including the pressure retaining portions of the parent tube in contact with the sleeve, the sleeve-to-tube weld and the pressure retaining portion of the sleeve.

9. Tube or Tubing means that portion of the tube that forms the primary system to secondary system pressure boundary.

10. Tube Inspection for tubes selected in accordance with Table 5.5.9-2 means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg, excluding sleeved areas.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.4 Acceptance Criteria (continued)

11. Tube Repair or Sleeve refers to welded sleeving, as described in Combustion Engineering, Inc. (CE or ABB-CE) report CEN-630-P, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," Revision 02, June 1997, which is used to maintain a tube in service or to return a previously plugged tube to service. Returning a previously plugged tube to service includes the removal of the tube plugs that were installed as a preventive or corrective measure and performing a tube inspection of the tube in accordance with Technical Specification 5.5.9.4.a.8.

12. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.3.d., above.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions required by Tables 5.5.9-2 and 5.5.9-3, including the following:

1. Plug or repair all defective tubes and all tubes containing through-wall cracks.

2. Plug all tubes containing any defective sleeves and all tubes containing any sleeves with through-wall cracks.

(continued)

TABLE 5.5.9-1
MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

| | | |
|---|---|---|
| Preservice Inspection | No | Yes |
| No. of Steam Generators per Unit | Two | Two |
| First Inservice Inspection | All | One |
| Second & Subsequent Inservice Inspection | One* | One* |

TABLE NOTATION

*The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

## TABLE 5.5.9-2
## STEAM GENERATOR TUBE INSPECTION

| 1ST SAMPLE INSPECTION | | | 2ND SAMPLE INSPECTION | | 3RD SAMPLE INSPECTION | |
|---|---|---|---|---|---|---|
| Sample Size | Result | Action Required | Result | Action Required | Result | Action Required |
| A minimum of S Tubes per S.G | C-1 | None | N.A. | N.A. | N.A. | N.A. |
| | C-2 | Plug or repair defective tubes and inspect additional 2S tubes in this S.G. | C-1 | None | N.A. | N.A. |
| | | | C-2 | Plug or repair defective tubes and inspect additional 4S tubes in this S.G. | C-1 | None |
| | | | | | C-2 | Plug or repair defective tubes |
| | | | | | C-3 | Perform action for C-3 result of first sample |
| | | | C-3 | Perform action for C-3 result of first sample | N.A. | N.A. |
| | C-3 | Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G.<br><br>Notification to NRC pursuant to 10 CFR 50.72 (b)(2) | All other S.G.s are C-1 | None | N.A. | N.A. |
| | | | Some S.G.s C-2 but no additional S.G. are C-3 | Perform action for C-2 result of second sample | N.A. | N.A. |
| | | | Additional S.G. is C-3 | Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to 10 CFR 50.72 (b)(2) | N.A. | N.A. |

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

TABLE 5.5.9-3
STEAM GENERATOR SLEEVE INSPECTION

| 1ST SAMPLE INSPECTION | | | 2ND SAMPLE INSPECTION | |
|---|---|---|---|---|
| Sample Size | Result | Action Required | Result | Action Required |
| A minimum of 20% of the sleeves per S.G. | C-1 | None | N.A. | N.A. |
| | C-2 | Plug tubes containing defective sleeves and inspect all remaining installed sleeves in this S.G. | C-1 | None |
| | | | C-2 | Plug tubes containing defective sleeves |
| | | | C-3 | Perform action for C-3 result of first sample |
| | C-3 | Inspect all installed sleeves in this S.G., plug tubes containing defective sleeves and inspect 100% of the installed sleeves in the other S.G.<br><br>Notification to NRC pursuant to 10 CFR 50.72 (b)(2) | Other S.G. is C-1 | None |
| | | | Other S.G. is C-2 | Plug tubes containing defective sleeves |
| | | | Other S.G. is C-3 | Inspect all sleeves in each S.G. and plug tubes containing defective sleeves.<br><br>Notification to NRC pursuant to 10 CFR 50.72(b)(2) |

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 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.

5.5.11 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, Revision 2, and in accordance with Regulatory Guide 1.52, Revision 2 and ANSI N510-1980 at the system flowrate specified below ± 10%.

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 with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, at the system flowrate specified as follows ± 10%:

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

| ESF Ventilation System | Flowrate |
|---|---|
| Control Room Essential Filtration System (CREFS) | 28,600 CFM |
| Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS) | 6,000 CFM |

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 Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

| ESF Ventilation System | Flowrate |
|---|---|
| CREFS | 28,600 CFM |
| ESF PREACS | 6,000 CFM |

c. Demonstrate for each of the ESF systems that a charcoal adsorber sample, when obtained in accordance with the application of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, as described in Section 1.8 of the UFSAR, shows the methyl iodide penetration less than or equal to the value specified below, when tested in accordance with ASTM D3803-1989, at a temperature of 30°C and to the relative humidity specified as follows:

| ESF Ventilation System | Penetration | RH |
|---|---|---|
| CREFS | ≤ 2.5% | 70% |
| ESF PREACS | ≤ 2.5% | 70% |

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than or equal to the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

| ESF Ventilation System | Delta P | Flowrate |
|---|---|---|
| CREFS | 4.8 inches water gauge | 28,600 CFM |
| ESF PREACS | 5.2 inches water gauge | 6,000 CFM |

e. Demonstrate that the heaters for each of the ESF systems dissipate the following specified value when tested in accordance with ANSI N510-1980:

| ESF Ventilation System | Wattage |
|---|---|
| ESF PREACS | > 19 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 control for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

5.5 Programs and Manuals (continued)

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

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

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 all outdoor liquid radwaste 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 connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

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 as referenced in the UFSAR. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

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 within limits when tested in accordance with ASTM D1796;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3.

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.

(continued)

5.5 Programs and Manuals (continued)

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

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 limitations and remedial or compensatory actions may be identified to 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 no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to 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.

5.5.16 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 Code Section XI, Subsection IWL, except where relief has been authorized by the NRC. The containment concrete visual examination may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

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 Code Section XI, Subsection IWE, except where relief has been authorized by the NRC.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 52.0 psig for Units 1 and 3, and 58.0 psig for Unit 2. The containment design pressure is 60 psig.

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

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance 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 $\leq 0.01\ L_a$ when pressurized to $\geq$ 14.5 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

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

5.6 Reporting Requirements (continued)

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "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.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

Replace with Insert 5.6.8

~~Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged and/or repaired in each steam generator shall be reported to the Commission in a Special Report.~~

~~The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:~~

~~a. Number and extent of tubes inspected.~~

~~b. Location and percent of wall-thickness penetration for each indication of an imperfection.~~

~~c. Identification of tubes plugged and/or repaired.~~

~~Results of steam generator tube and sleeve inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

## INSERT 5.6.8

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.

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

**Attachment 2B**

**Proposed Technical Specification Changes (retyped)**

**Pages:**

**ii**
**1.1-4 and 5**
**3.4.14-1 and 2**
**3.4.18-1 and 2 (new)**
**5.5-6 through 5.5-16**
**5.6-6**

# TABLE OF CONTENTS

1.1 Definitions (continued)

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

$K_{n-1}$

$K_{n-1}$ is the K effective calculated by considering the actual CEA configuration and assuming that the fully or partially inserted full strength CEA of highest worth is fully withdrawn.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE).

(continued)

1.1 Definitions

| | |
|---|---|
| LEAKAGE (continued) | b. Unidentified LEAKAGE<br><br>All LEAKAGE that is not identified LEAKAGE;<br><br>c. Pressure Boundary LEAKAGE<br><br>LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall. |
| MODE | A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, cold leg reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel. |
| NEUTRON RATED THERMAL POWER (NRTP) | The indicated neutron flux at RTP. |
| OPERABLE - OPERABILITY | A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). |
| PHYSICS TESTS | PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:<br><br>a. Described in Chapter 14, Initial Test Program of the UFSAR;<br><br>b. Authorized under the provisions of 10 CFR 50.59; or<br><br>c. Otherwise approved by the Nuclear Regulatory Commission. |

(continued)

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.14 RCS Operational LEAKAGE

LCO 3.4.14 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE. | A.1 Reduce LEAKAGE to within limits. | 4 hours |
| B. Required Action and associated Completion Time of Condition A not met.<br><br>OR<br><br>Pressure boundary LEAKAGE exists.<br><br>OR<br><br>Primary to secondary LEAKAGE not within limit. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.14.1 | -------------------NOTES------------------<br>1. Not required to be performed until 12 hours after establishment of steady state operation.<br><br>2. Not applicable to primary to secondary LEAKAGE<br>-----------------------------------------<br><br>Perform RCS water inventory balance. | 72 hours |
| SR 3.4.14.2 | -------------------NOTE-------------------<br>Not required to be performed until 12 hours after establishment of steady state operation.<br>-----------------------------------------<br><br>Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG. | 72 hours |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair 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 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 |
| | 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. |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Be in MODE 3. | 6 hours |
| OR | AND | |
| SG tube integrity not maintained. | B.2 Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.18.1 | Verify SG tube integrity in accordance with the Steam Generator Program. | In accordance with the Steam Generator Program. |
| SR 3.4.18.2 | 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. |

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program

A Steam Generator (SG) 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 an 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 and 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 SG 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) 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,

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

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 0.5 gpm per SG and 1 gpm through both SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

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. For the original SGs with Alloy 600 MA tubes, all detected degradation in the portion of the tube below the bottom of the expansion transition that is not excluded from inspection per the provisions of 5.5.9.d shall be removed from service.

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. For the original SGs with Alloy 600 MA tubes, the section of tube from the tube-to-tubesheet weld to 12 inches below the bottom of the expansion transition for both the tube inlet and outlet may be excluded from inspection based on Westinghouse Report WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions." In addition to meeting the requirements of d.1, d.2a, d.2b, 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.

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

2a. Original SGs with Alloy 600 MA tubes: Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

2b. Replacement SGs with Alloy 690 TT tubes: 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 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)

## 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 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.

### 5.5.11 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, Revision 2, and in accordance with Regulatory Guide 1.52, Revision 2 and ANSI N510-1980 at the system flowrate specified below ± 10%.

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 with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, at the system flowrate specified as follows ± 10%:

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

| ESF Ventilation System | Flowrate |
|---|---|
| Control Room Essential Filtration System (CREFS) | 28,600 CFM |
| Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS) | 6,000 CFM |

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 Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

| ESF Ventilation System | Flowrate |
|---|---|
| CREFS | 28,600 CFM |
| ESF PREACS | 6,000 CFM |

c. Demonstrate for each of the ESF systems that a charcoal adsorber sample, when obtained in accordance with the application of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, as described in Section 1.8 of the UFSAR, shows the methyl iodide penetration less than or equal to the value specified below, when tested in accordance with ASTM D3803-1989, at a temperature of 30°C and to the relative humidity specified as follows:

| ESF Ventilation System | Penetration | RH |
|---|---|---|
| CREFS | ≤ 2.5% | 70% |
| ESF PREACS | ≤ 2.5% | 70% |

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than or equal to the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

| ESF Ventilation System | Delta P | Flowrate |
|---|---|---|
| CREFS | 4.8 inches water gauge | 28,600 CFM |
| ESF PREACS | 5.2 inches water gauge | 6,000 CFM |

e. Demonstrate that the heaters for each of the ESF systems dissipate the following specified value when tested in accordance with ANSI N510-1980:

| ESF Ventilation System | Wattage |
|---|---|
| ESF PREACS | > 19 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 control for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

5.5 Programs and Manuals (continued)

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

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

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 all outdoor liquid radwaste 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 connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

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 as referenced in the UFSAR. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

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 within limits when tested in accordance with ASTM D1796;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3.

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.

(continued)

5.5 Programs and Manuals (continued)

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

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 limitations and remedial or compensatory actions may be identified to 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 no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to 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.

5.5.16 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 Code Section XI, Subsection IWL, except where relief has been authorized by the NRC. The containment concrete visual examination may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

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 Code Section XI, Subsection IWE, except where relief has been authorized by the NRC.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 52.0 psig for Units 1 and 3, and 58.0 psig for Unit 2. The containment design pressure is 60 psig.

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

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance 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 $\leq 0.01\ L_a$ when pressurized to $\geq 14.5$ psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

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

5.6 Reporting Requirements (continued)

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "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.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 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.

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

Attachment 2C

Proposed Technical Specification Bases Changes (mark-up)

Pages:

B 3.4.4-2
B 3.4.5-2 and 3
B 3.4.6-3
B 3.4.7-4
B 3.4.14-4 through B 3.4.14-7
B 3.4.18-1 through B 3.4.18-8 (new)

BASES

APPLICABLE SAFETY ANALYSES (continued)

aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

The reactor coolant pumps provide sufficient forced circulation flow through the reactor coolant system to assure adequate heat removal from the reactor core during power operation. The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain a departure from nucleate boiling ratio (DNBR) above the DNBR Safety Limit during all normal operations and anticipated transients. The safety analyses that are of most importance to RCP operation are the total loss of reactor coolant flow, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

RCS Loops – MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36 (C)(2)(ii).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. SG, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2.

(continued)

BASES

LCO

The purpose of this LCO is to require two RCS loops to be available for heat removal, thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable (≥ 25% wide range water level) of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The Note permits a limited period of operation without RCPs. All RCPs may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration is prohibited because an even concentration distribution throughout the RCS cannot be ensured. The intent is to stop any known or direct positive reactivity additions to the RCS due to dilution. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The 10 degrees F is considered the actual value of the necessary difference between RCS core outlet temperature and the saturation temperature associated with RCS pressure to be maintained during the time the pumps would be de-energized. The instrument error associated with determining this difference is 27 degrees F. (The only restriction for instrumentation use is with pressurizer pressure less than or equal to 350 psia, and in that situation the narrow range pressurizer pressure instrumentation must be used.) Therefore, the indicated value of the difference between RCS core outlet temperature and the saturation temperature associated with RCS pressure must be greater than or equal to 37 degrees F in order to use the provisions of the Note allowing the pumps to be de-energized.

In MODE 3 it is sometimes necessary to stop all RCPs (e.g.,to perform surveillance or startup testing, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop (loop 1 or loop 2) consists of at least one associated OPERABLE RCP and an associated SG that is OPERABLE ~~in accordance with the Steam Generator Tube~~

(continued)

BASES

LCO (continued)

Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4 "RCS Loops-MODES 1 and 2";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core.

B.1

If restoration is not possible within 72 hours, the unit must be placed in MODE 4 within 12 hours. In MODE 4, the plant may be placed on the SDC System. The Completion Time of 12 hours is compatible with required operation to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

**LCO (continued)**

Note 2 requires, that before an RCP may be started with any RCS cold leg temperature ≤ 214°F during cooldown, or ≤ 291°F during heatup, that secondary side water temperature (saturation temperature corresponding to SG pressure) in each SG is < 100°F above each of the RCS cold leg temperatures. The numerical values for RCS cold leg temperature at which this Note is applicable do not account for all instrument uncertainty. Use of an indicated value of 217°F or below during cooldown and 294°F or below during heatup ensures that the actual limits will not be exceeded. These values, which include appropriate instrument uncertainty, are established within the applicable plant procedures.

Satisfying the above condition will preclude a large pressure surge in the RCS when the RCP is started.

Note 3 restricts RCP operation to no more than 2 RCPs with RCS cold leg temperature ≤ 200°F, and no more than 3 RCPs with RCS cold leg temperature >200°F but ≤ 500°F. Satisfying these conditions will maintain the analysis assumptions of the flow induced pressure correction factors due to RCP operation (Ref. 1)

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program and has the minimum water level specified in SR 3.4.6.2.

Similarly, for the SDC System, an OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal. RCPs and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow, if required.

**APPLICABILITY**

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4 "RCS Loops-MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";

(continued)

BASES

LCO (continued)

Note 5 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of SDC trains from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC trains.

An OPERABLE SDC train is composed of an OPERABLE SDC pump (CS or LPSI) capable of providing flow to the SDC heat exchanger for heat removal.

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow (current Section XI), if required. An ~~OPERABLE~~ SG can perform as a heat sink when it is OPERABLE ~~in accordance with the SG Tube Surveillance Program~~ and has the minimum water level specified in SR 3.4.7.2.

The RCS loops may not be considered filled until two conditions needed for operation of the steam generators are met. First, the RCS must be intact. This means that all removable portions of the primary pressure boundary (e.g., manways, safety valves) are securely fastened. Nozzle dams are removed. All manual drain and vent valves are closed, and any open system penetrations (e.g., letdown, reactor head vents) are capable of remote closure from the control room. An intact primary allows the system to be pressurized as needed to achieve the subcooling margin necessary to establish natural circulation cooling. When the RCS is not intact as described, a loss of SDC flow results in blowdown of coolant through boundary openings that also could prevent adequate natural circulation between the core and steam generators. Secondly, the concentration of dissolved or otherwise entrained gases in the coolant must be limited or other controls established so that gases coming out of solution in the SG U-tubes will not adversely affect natural circulation. With these conditions met, the SGs are a functional method of RCS heat removal upon loss of the operating SDC train. The ability to feed and steam SGs at all times is not required when RCS temperature is less than 210°F because significant loss of SG inventory through boiling will not occur during time anticipated to take corrective action. The required SG level provides sufficient time to either restore the SDC train or implement a method for feeding and steaming the SGs (using non-class components if necessary).

(continued)

BASES

LCO
(continued)

LCO 3.4.15, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

d. Primary to Secondary LEAKAGE through Any One SG

Replace with Insert

The maximum allowable operational primary to secondary LEAKAGE through any one SG of 150 gpd is based on operating experience as an indication of one or more propagating tube leak mechanisms. This operational limit is significantly less than the initial conditions assumed in the safety analyses. The Steam Generator Tube Surveillance Program described in TS Section 5.5.9 ensures that the structural integrity of the SG tubes is maintained. The 150 gpd leakage rate limit provides additional assurance against tube rupture at normal and faulted conditions and provides additional assurance that cracks will not propagate to burst prior to detection by leakage monitoring methods and commencement of plant shutdown. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

(continued)

## TS Bases B 3.4.14, LCO, Section d (INSERT)

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

BASES

ACTIONS

A.1

Unidentified LEAKAGE~~,~~ or identified LEAKAGE~~, or primary to secondary LEAKAGE~~ in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, or if unidentified~~,~~ or identified~~, or primary to secondary~~ LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

~~C.1~~

~~If one or more SGs are inoperable, due to SR 3.4.14.2, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

~~Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

~~The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4, until 12 hours of steady state operation near operating pressure have elapsed. This means that once steady state operating conditions are established, 12 hours is allowed for completing the Surveillance if the Surveillance Frequency interval was exceeded in MODE 5 or 6. Further discussion of SR note format is found in Section 1.4, Frequency.~~

~~The Note in the Frequency column allows for SR 3.4.14.1 nonperformance due to planned or unplanned power manipulations. This Note is not intended to allow power manipulations solely for the purpose of avoiding SR 3.4.14.1 performance. Steady state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.~~

Revise as shown in SR 3.4.14.1 Bases Revision

(continued)

## TS Bases SR 3.4.14.1 (REVISION)

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) ~~and near operating pressure~~. ~~Therefore,~~ This surveillance is modified by two notes. Note 1 states that this SR is not required to be performed ~~in MODES 3 and 4,~~ until 12 hours ~~of~~ after establishing steady state operation. ~~near operating pressure have elapsed.~~ This means that once steady state operating conditions are established, 12 hours is allowed for completing the Surveillance ~~if the Surveillance Frequency interval was exceeded in MODE 5 or 6~~. When required by the Frequency, and after steady state operating conditions are established, the surveillance must be completed prior to the end of 12 hours of steady state operation. If steady state operating conditions have not been established for 12 hours, this surveillance is not required until steady state operation is established for 12 hours. This SR is not required to be completed prior to changing MODES if steady state operation has not been established for 12 hours. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Further discussion of SR note format is found in Section 1.4, Frequency.

~~The~~ Note 1 ~~in the Frequency column~~ allows for SR 3.4.14.1 nonperformance due to planned or unplanned ~~power manipulations~~ transients. This Note is not intended to allow ~~power manipulations~~ transients solely for the purpose of avoiding SR 3.4.14.1 performance. Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. ~~and a Note requires the Surveillance to be met when steady state is established.~~ For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. ~~A Note under the Frequency column states that this SR is required to be performed during steady state operation.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

Revise as shown in Revision

~~An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.16. "RCS Leakage Detection Instrumentation."~~

~~The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.~~

SR 3.4.14.2

Replace with Insert SR 3.4.14.2 BASES Revision

~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Section 15.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

## TS Bases SR 3.4.14.2 (INSERT)

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.18, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. This means that once steady state operating conditions are established, 12 hours is allowed for completing the Surveillance. When required by the Frequency, and after steady state operating conditions are established, the surveillance must be completed prior to the end of 12 hours of steady state operation. If steady state operating conditions have not been established for 12 hours, this surveillance is not required until steady state operation is established for 12 hours. This SR is not required to be completed prior to changing MODES if steady state operation has not been established for 12 hours. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Further discussion of SR note format is found in Section 1.4, Frequency.

The Note allows for SR 3.4.14.2 nonperformance due to planned or unplanned transients. This Note is not intended to allow transients solely for the purpose of avoiding SR 3.4.14.2 performance. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

(continued)

BASES

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 bounding primary to secondary LEAKAGE rate equal to one gallon per minute (1440 gallons per day) in the unaffected SG 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 only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

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 of 0.5 gallon per minute (gpm) from each SG or 1 gpm from both SGs, or is assumed to increase to those levels as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level is assumed to be equal to the LCO 3.4.17, "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. 2), 10 CFR 100 (Ref. 3) 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 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

(continued)

LCO
(continued)

tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

(NOTE: Specification 5.5.9, "Steam Generator Program," Provision (d), specifies a SG tube inspection length for the original SGs with Alloy 600 MA tubes as the length of the tube from 12 inches below the hot-leg expansion transition at the tube inlet to 12 inches below the cold-leg expansion transition at the tube outlet.)

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "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 when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are

(continued)

BASES

LCO
(continued)

classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref.5).

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 0.5 gpm from each SG or 1 gpm total from both SGs. 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 3.4.14, "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 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.

(continued)

BASES

APPLICABILITY (continued)

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 3.4.18.2. 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 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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

tube(s). 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 3.4.18.1

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 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

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SURVEILLANCE REQUIREMENTS

SR 3.4.18.1 (continued)

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

The Steam Generator Program defines the Frequency of SR 3.4.18.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). 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.

SR 3.4.18.2

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 5.5.9 are intended to ensure that the 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.

(continued)

BASES

REFERENCES

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

# ENCLOSURE 3

## Westinghouse Affidavit Requesting That WCAP-16208-P, Revision 1 Be Withheld From Public Disclosure In Accordance With The Provisions Of 10 CFR 2.390(A)(4)

Westinghouse

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Our ref: CAW-05-1997

May 18, 2005

**APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE**

**Subject: Revision 1 to WCAP-16208-P, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions"**

Reference: Westinghouse Report WCAP-16208-P Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions," May 2005.

Westinghouse, via the Arizona Public Service Company (APS), transmits the above-referenced proprietary document for which withholding is requested. Affidavit CAW-05-1997, which accompanies this letter and is signed by the owner of the proprietary information, Westinghouse Electric Company LLC, sets forth the basis on which this information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR Section 2.390(b)(4) of the Commission's regulations. This letter also authorizes use of the accompanying affidavit by APS.

In conformance with the requirements of 10 CFR 2.390, Westinghouse confirms that the information contained within the referenced document is proprietary. The justification for claiming this document as proprietary is identified in Sections (4)(ii)(a) through (4)(ii)(f) of the enclosed affidavit.

Communication with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-1997, and be addressed to the undersigned.

Very truly yours,

James A. Gresham
James A. Gresham
Manager
Regulatory Compliance and Plant Licensing

Enclosure:

# AFFIDAVIT

STATE OF CONNECTICUT )

) ss: WINDSOR, CT

COUNTY OF HARTFORD )

Before me, the undersigned authority, personally appeared James A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company, LLC

Sworn to and subscribed before me
this 18th day of May 2005.

Notary Public

My commission expires May 31, 2008.

(1) I, James A. Gresham, depose and say that I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC ("Westinghouse"), and as such I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system for classification of proprietary information, which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use of this information by our competitors would put Westinghouse at a competitive disadvantage by reducing their expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure of this information would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is contained in Westinghouse Report WCAP-16208-P Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions," May 2005.

The information is part of a model that will enable Westinghouse to support utilities with CE NSSS plants in the identification and application of a steam generator tubesheet inspection model, including:

(a) The identification of important factors relevant to the determination of the recommended steam generator tubesheet inspection length, and

(b) Development of a generic methodology for the applicability of the inspection length model to utilities with CE NSSS plants.

(vii) Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology that was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar advanced nuclear power plant designs and to provide licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

## PROPRIETARY INFORMATION NOTICE

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, Westinghouse confirms that the information in Westinghouse Report WCAP-16208-P Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions," May 2005 is proprietary. The justification for claiming this information as proprietary is indicated in Sections (4)(ii)(a) through (4)(ii)(f) of affidavit CAW-05-1997 accompanying this transmittal.

## COPYRIGHT NOTICE