Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

October 24, 2001
NOC-AE-01001142
10CFR50.90

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Proposed Amendment to Relocate Various Technical Specifications
to the Technical Requirements Manual

Pursuant to 10 CFR 50.90, STP Nuclear Operating Company (STPNOC) requests the following amendment of Operating Licenses NPF-76 and NPF-80 for South Texas Project Units 1 and 2. The proposed change will relocate various Technical Specifications and their associated Bases descriptions to the Technical Requirements Manual to include:

3/4.1.3.3 "Position Indication System, Shutdown" and its Special Test Exception
3/4.3.3.2 "Movable Incore Detectors"
3/4.3.3.11 "Explosive Gas Monitoring Instrumentation"
3/4.4.7 "Chemistry"
3/4.4.9.2 "Pressure/Temperature Limits – Pressurizer"
3/4.4.11 "Reactor Vessel Head Vents"
3/4.7.2 "Steam Generator Pressure/Temperature Limitation"
3/4.7.10 "Sealed Source Contamination"
3/4.9.3 "Decay Time"
3/4.9.5 "Communications"
3/4.9.7 "Crane Travel - Fuel Handling Building"
3/4.10.5 "Special Test Exception, Position Indication System, Shutdown"
3/4.11.2.5 "Explosive Gas Mixture"

In addition, this proposed amendment corrects various typographical and page numbering errors, deletes an outdated one-time exception, and makes minor format changes to improve consistency.

This proposed amendment is consistent with NUREG-1431, "Standard (Improved) Technical Specifications - Westinghouse Plants."

STPNOC has reviewed the proposed amendment pursuant to 10CFR50.92 and determined that it involves a no significant hazards consideration. In addition, STPNOC has determined that the proposed amendment satisfies the criteria of 10CFR51.22(c)(9) for categorical exclusion from

A001

the requirement for an environmental assessment. The STP Plant Operations Review Committee and Nuclear Safety Review Board have reviewed and approved the proposed amendment.

In accordance with 10CFR50.91(b), STPNOC is notifying the State of Texas of this request for a license amendment by providing a copy of this letter and its attachments.

STPNOC requests that the proposed amendment be reviewed and approved by the Nuclear Regulatory Commission by June 30, 2002. In addition, STPNOC requests 6 months for implementation following NRC approval of this amendment request. The Safety Evaluation, and the proposed revised pages of the Technical Specifications are included as attachments to this letter. The marked-up Bases are provided for information.

If there are any questions regarding the proposed amendment, please contact R. D. Piggott at (361) 972-7438 or me at (361) 972-8757.

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

Executed on: 10/24/01

J. J. Sheppard
Vice President,
Engineering & Technical Services

RDP/

Attachments:

1. Licensee's Evaluation
2. Proposed Technical Specification Changes (Mark-up)
   (Bases sections provided for information)
3. Proposed Technical Specification Pages (Re-Typed)

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Attention: Document Control Desk
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ATTACHMENT 1

LICENSEE'S EVALUATION

LICENSEE'S EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses NPF-76 and NPF-80 for South Texas Project Units 1 and 2. The proposed change will relocate various Technical Specifications to the Technical Requirements Manual (TRM):

3/4.1.3.3	"Position Indication System, Shutdown" and its Special Test Exception
3/4.3.3.2	"Movable Incore Detectors"
3/4.3.3.11	"Explosive Gas Monitoring Instrumentation"
3/4.4.7	"Chemistry"
3/4.4.9.2	"Pressure/Temperature Limits – Pressurizer"
3/4.4.11	"Reactor Vessel Head Vents"
3/4.7.2	"Steam Generator Pressure/Temperature Limitation"
3/4.7.10	"Sealed Source Contamination"
3/4.9.3	"Decay Time"
3/4.9.5	"Communications"
3/4.9.7	"Crane Travel - Fuel Handling Building"
3/4.10.5	"Special Test Exception, Position Indication System, Shutdown"
3/4.11.2.5	"Explosive Gas Mixture"

Their associated Bases will also be relocated to the TRM to be consistent with relocation of the specifications. In addition, this proposed amendment corrects various typographical and page numbering errors, deletes an outdated one-time exception, and makes minor format changes to improve consistency.

The proposed changes are consistent with NUREG-1431, "Standard (Improved) Technical Specifications - Westinghouse Plants." Any changes to these requirements in the TRM will be made in accordance with 10CFR50.59.

2.0 PROPOSED CHANGE

STPNOC proposes to relocate the following Technical Specification and associated Bases to the STP Technical Requirements Manual. In addition, some administrative changes are proposed to correct various typographical errors and changes to improve consistency. This requires:

2.1 TS 3/4.1.3.3, Position Indication System - Shutdown and its exception in TS 3/4.10.5

- Relocating the existing Position Indication System Technical Specification 3/4.1.3.3 and Special Test Exception 3/4.10.5 including the portions of the associated Bases in the Technical Requirements Manual.

- Deleting the Position Indication System - Shutdown references found in the Technical Specification Index;

2.2 TS, 3/4.3.3.2, "Movable Incore Detectors"

- Relocating the existing Movable Incore Detectors Technical Specification 3/4.3.3.2 and the associated Bases to the Technical Requirements Manual;

- Removing the references to TS 3.3.3.2 from TS 4.2.4.2 action a and TS 4.2.4.2 action b [Quadrant Power Tilt Ratio within limits]. Specifically, for TS surveillance 4.2.4.2.a, remove "(Specification 3.3.3.2.a does not apply)" and for TS surveillance 4.2.4.2.b, replace "subject to the requirements of Specification 3.3.3.2" with "with a full incore map". Reference to Specification 3.3.3.2 is detail that is no longer applicable with the relocation of TS 3/4.3.3.2 to the TRM. Details concerning use of incore detectors to verify Quadrant Power Tilt Ratio will be added to the TS Bases 3/4.2.4 for clarification.

- Deleting the Movable Incore Detectors references found in the Technical Specification Index;

2.3 3/4.3.3.11, "Explosive Gas Monitoring Instrumentation" and 3/4.11.2.5 "Explosive Gas Mixture"

- Relocating the existing Explosive Gas Monitoring Instrumentation and Explosive Gas Mixture Technical Specifications 3/4.3.3.11, 3/4.11.2.5, including Table 3.3-13 and Table 4.3-9, and their associated Bases to the Technical Requirements Manual. <u>Note:</u> The requirement to issue a Special Report under the conditions described in the specification will be retained. However, reference to TS 6.9.2 will be removed when this requirement is relocated to the TRM. This is consistent with NUREG-1431. STPNOC is also requesting a License Amendment under separate submittal that revises Section 6.0 of Technical Specifications, Administrative Controls where TS 6.9.2 has been deleted. (See Section 7.0, Reference 2);

- Deleting the Explosive Gas Monitoring Instrumentation and Explosive Gas Mixture references found in the Technical Specification Index including Table 3.3-13 and Table 4.3-9;

2.4 TS 3/4.4.7, "Chemistry"

- Relocating the existing Chemistry Technical Specification 3/4.4.7, including Table 3.4-2 and Table 4.4-3, and the associated Bases to the Technical Requirements Manual;

- Deleting the Chemistry references found in the Technical Specification Index including Table 3.4-2 and Table 4.4-3;

2.5 3/4.4.9.2, "Pressure/Temperature Limits - Pressurizer"

- Relocating the existing Pressure/Temperature Limits - Pressurizer Technical Specification 3/4.4.9.2 and the portion of the associated Bases regarding pressurizer temperature limits to the Technical Requirements Manual;

- Deleting the Temperature Limits - Pressurizer references found in the Technical Specification Index;

2.6 TS 3/4.4.11, "Reactor Vessel Head Vents"

- Relocating the existing Reactor Vessel Head Vents Technical Specification 3/4.4.11 and the associated Bases to the Technical Requirements Manual;

- Deleting the Reactor Vessel Head Vents references found in the Technical Specification Index;

2.7 TS 3/4.7.2, "Steam Generator Pressure/Temperature Limitation"

- Relocating the existing Steam Generator Pressure/Temperature Limitation Technical Specification 3/4.7.2 and the associated Bases including the portion of 3/4.4.9 (item 3) to the Technical Requirements Manual;

- Deleting the Steam Generator Pressure/Temperature Limitation references found in the Technical Specification Index;

2.8 3/4.7.10, "Sealed Source Contamination"

- Relocating the existing Sealed Source Contamination Technical Specification 3/4.7.10, the associated Bases to the Technical Requirements Manual. Note: The applicable records retention requirements of Technical Specification 6.10.2 , items (f) and (g) concerning sealed sources will be maintained in the TSs until STPNOC License Amendment Request which revises Section 6.0 of Technical Specifications, Administrative Controls is approved. This amendment request will relocate the records requirements of TS 6.10.2 to the OQAP. (See Section 7.0, Reference 2).

- Deleting the Sealed Source Contamination references found in the Technical Specification Index;

2.9 3/4.9.3, "Decay Time"

- Relocating the existing Decay Time Technical Specification 3/4.9.3 and associated Bases to the Technical Requirements Manual;

- Deleting the Decay Time references found in the Technical Specification Index;

2.10 3/4.9.5, "Communications"

- Relocating the existing Refueling Operations – Communications Technical Specification 3/4.9.5 and associated Bases to the Technical Requirements Manual;

- Deleting the Refueling Operations – Communications references found in the Technical Specification Index;

2.11 3/4.9.7, "Crane Travel - Fuel Handling Building"

- Relocating the existing Crane Travel - Fuel Handling Building Technical Specification 3/4.9.7 and associated Bases to the Technical Requirements Manual;

- Deleting the Crane Travel - Fuel Handling Building references found in the Technical Specification Index;

2.12 Administrative Changes

The proposed changes are completely administrative in nature:

- On Index page "iv,"
  - The title for Figure 3.1-2a should read "BOL MTC VERSUS POWER LEVEL"
  - The page the entry for Technical Specification 3/4.1.2, "Boration Systems," should be changed to read "(This specification number is not used.)" and the listed subsections should be deleted along with their page numbers.
  - Reference to FIGURE 3.1-3 should be deleted.
- On Index page "vi,"
  - The references to Chemical Detection Systems, TABLE 3.3-11, Radioactive Liquid Effluent Monitor Instrumentation, associated Tables 3.3-12 and 4.3-8, unused TS 3/4.3.4 and page numbers should be removed.
- On Index page "viii,"
  - TABLE 4.4-5 and the page number should be deleted
  - the page number for FIGURE 3.4-4 should read "3/4 4-37"
  - the page number for TS 3/4.5.4 should be deleted (page 3/4 5-9)
  - the page number for "Containment Cooling System" should read "3/4 6-16"
- On Index page "ix,"
  - the page number for Section 3/4.6.3 should read "3/4 6-17"
  - the page number for "Hydrogen Analyzers" should read "3/4 6-18"

- the page for "Electric Hydrogen Recombiners" should read "3/4 6-19"

- On Index page "x,"
  - the reference to TABLE 4.8-1 and the page number should be deleted
  - the entry for Technical Specification 3/4.8.4, "Electrical Equipment Protective Devices," should be changed to read "(This specification number is not used.)" and the listed subsections should be deleted along with their page numbers
- On Index page "xi,"
  - the individual items under TS 3/4.11.1 and TS 3/4.11.2 that have been deleted should be removed
  - TS 3/4.11.3 and 3/4.11.4 should both read "(This specification number is not used.)"
- On Index page "xiii,"
  - the page number for Basis 3/4.4.2 should read "B 3/4 4-1a"
- On Index page "xiv,"
  - the page number for Basis 3/4.7.8 should read "B 3/4 7-5a"
- On Index page "xv,"
  - the page number for Basis 3/4.7.14 should read "B 3/4 7-7"
  - the entry for Basis 3/4.8.4 should read "Not used" and the page number for that Basis should be deleted
  - the page number for Basis 3/4.9.6 should be deleted
  - the page number for Basis 3/4.9.4 should read "B 3/4 9-1a"
  - the page number for Basis 3/4.9.12 should read "B 3/4 9-3a"
- On page 3/4 0-3,
  - the asterisk at the end of paragraph 4.0.5 and the last paragraph on the page should be deleted because this was a one-time exception for Unit 1 that ended in March 1999
  - "AMENDMENT" in the footer should read "Amendment" in two places
- Page 3/4 1-9 should be changed to reflect that Technical Specification 3/4.1.2 is not used and correct reference to page numbers that were deleted.
- On page 3/4 3-17 (TS 4.3.2.1),
  - "Table 4.3.2" in paragraph 4.3.2.1 should read "Table 4.3-2"
- On page 3/4 3-48,
  - the heading for the fourth column in TABLE 4.3-2 should read "DIGITAL "OR" ANALOG CHANNEL OPERATIONAL TEST"
  - the entry for item 10e should read "See Item 8. above..."

- Page 3/4 3-75 should be changed to reflect that Technical Specifications 3/4.3.7 through 3/4.3.11 and 3/4.3.4 are not used. Also add a note to show that blank pages 3/4 3-76 through 3/4 3-84 are deleted. Note: with the removal of TS 3/4.3.3.11 (refer to section 2.3 above) pages 3/4 3-76 through 3/4 3-84 will be blank and can be deleted.
- delete blank pages 3/4 3-76 through 3/4 3-84.
- The last Unit 1 Amendment No. should be restored (not stricken through) on pages 3/4 4-13, 4-13a, 4-16, 4-16a, 4-16b, 4-18, and 4-18a.
- On pages 3/4 4-18, 4-18a, "Table 4.4-3" should be in all capital letters.
- Page 3/4 7-21 should be changed to reflect that Technical Specifications 3/4.7.9 through 3./4.7.13 are not used. Also add a note to show that blank pages 3/4 7-22 through 3/4 7-32 are deleted. Note: with the removal of TS 3/4.7.10 (refer to section 2.8 above) pages 3/4 7-22 through 3/4 7-32 will be blank and can be deleted.
- delete blank pages 3/4 7-22 through 3/4 7-32

Refer to Attachment 2 for the actual page mark-up of Technical Specification changes.

3.0 BACKGROUND

The proposed changes relocate the requirements of above noted Technical Specifications including Surveillance requirements. These Technical Specifications do not meet the criteria for inclusion in Technical Specifications as identified in 10CFR50.36(c)(2)(ii) or 10CFR50.36(c)(3) with the exception of TS 3/4.9.3, "Decay Time".

TS 3/4.9.3, "Decay Time" satisfies Criterion 2 of 10CFR50.36(c)(2)(ii) in that it is a process variable that is an initial condition to a design basis accident (DBA) that presents a challenge to the integrity of a fission product barrier. However, the activities necessary prior to commencing movement of irradiated fuel (e.g., containment entry, reactor vessel head removal) ensures that there will always be > 42 hours (Decay Time limit) of sub-criticality before movement of any irradiated fuel. The Industry/NRC agreed during the development of NUREG-1431 that this Limiting Condition for Operation could be relocated since it is not required to be in Technical Specifications to provide adequate protection of the health and safety of the public. Hence, this Specification along with the others identified should be relocated to the TRM, consistent with NUREG-1431.

The proposed changes will facilitate future changes to these requirements without obtaining NRC approval. Any changes to these requirements will require a 10CFR50.59 evaluation.

3.1 TS 3/4.1.3.3, "Position Indication System – Shutdown" and Special Test Exception TS 3/4.10.5

These Specifications provide operability requirements for the rod position indication system. The system indication is required to be capable of determining the control rod position within 12 steps for the shutdown and control banks position in MODES 3, 4, and 5 with the Reactor Trip System breakers closed (capable of rod withdrawal). This requirement provides adequate assurance that rod position indication during shutdown conditions is accurate, and that the indication is available to the operator to detect inoperable, misaligned, or mispositioned rods. The Special Test Exception permits the Position Indication Systems to be inoperable during rod drop time measurements in order to obtain required data to determine the rod drop time. The rod position indication signals at the control board display are momentarily interrupted during the rod drop test.

With the reactor sub-critical and no significant fission power being produced, the requirement to have indication of the rod position is unnecessary. The safety functions associated with the shutdown or control rods are the ability of the rods to be tripped and negative reactivity that can be inserted from rods that are withdrawn. The rod indication Specification in MODES 3, 4, or 5, is not required and only necessary when the reactor is critical, to ensure proper power distribution.

The rod position monitoring system is described in UFSAR section 7.7.1.3.2. The rod deviation and rod bottom alarms are derived from these signals as described in UFSAR 7.7.1.3.4 and 7.7.1.3.5.

3.2 TS 3/4.3.3.2, "Movable Incore Detectors"

This Specification ensures the operability of movable incore detector instrumentation during conditions where the instrumentation is required to monitor the flux distribution within the core. The detectors are used for periodic surveillance of the power distribution (TS 4.2.2.2.a [Heat Flux Hot Channel Factor - $F_Q(Z)$], TS 4.2.3.2 [Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$], TS 4.2.4.2 [Quadrant Power Tilt Ratio]) and calibration of the excore detectors (Table 4.3-1, 4.3.1.1.2.a.2 [notes 3 and 6]). The movable incore detector instrumentation is also required to complete various TS Limiting Conditions for Operation (LCO) to include inoperable control rod TS 3.1.3.1 [action b.3.c], inoperable rod position indicator bank TS 3.1.3.2 [action a.1], $F_Q(Z)$ limit exceeded TS 3.2.2 [action b], $F^N_{\Delta H}$ limit exceeded TS 3.2.3 [action b].

The detectors are not assumed in any design basis accident analysis and do not mitigate an accident. This requirement is not necessary to ensure safe reactor operation as summarized in WCAP-11618.

The movable incore detector system is described in UFSAR 7.7.1.9. Use of the system to verify flux distribution within the core is discussed in UFSAR Chapter 4.3.

3.3 TS 3/4.3.3.11, "Explosive Gas Monitoring Instrumentation" and TS 3/4.11.2.5 "Explosive Gas Mixture"

The explosive gas monitoring instrumentation is provided to help ensure that the concentration of potentially explosive waste gas mixtures contained in the gaseous waste processing system is adequately monitored in order to maintain it below the explosive gas mixture limit. The explosive gas mixture limit is provided to ensure that the concentration of potentially explosive gas mixtures is maintained below the flammability limit of oxygen. The gaseous waste processing system automatically shuts down on high oxygen concentration, preventing entry of high levels into the system. The concentration of oxygen in the gaseous waste processing system is not an initial assumption of any design basis accident or transient analysis.

The gaseous waste processing system is designed in accordance with 10CFR50, Appendix I as described in UFSAR 11.3.

3.4 TS 3/4.4.7, "Chemistry"

This Specification provides limits on the concentration of dissolved oxygen, chloride, and fluoride in the reactor coolant system (RCS). RCS water chemistry is monitored routinely for a variety of reasons. One reason is to reduce the possibility of failures in the reactor coolant system pressure boundary caused by corrosion. Poor reactor coolant water chemistry contributes to the long-term degradation of system materials of construction. The effects of exceeding the limits are time and temperature dependent. The chemistry monitoring activity has a long-term preventative purpose rather than mitigating. The surveillance requirements provide assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3.5 TS 3/4.4.9.2, "Pressure/Temperature Limits - Pressurizer"

This Specification provides pressurizer heat-up and cool-down rates and the maximum spray water temperature differential during auxiliary spray operations. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to cyclic induced failure in the pressure boundary of the pressurizer. The heat-up and cool-down rate temperature limits are placed on the pressurizer to limit the cyclic, thermal loading on critical areas in the pressure boundary. The limits on the rate of change of temperature have been established using approved methodology, to preclude operation in an unanalyzed condition.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure".

3.6 TS 3/4.4.11, "Reactor Vessel Head Vents"

This Specification assures that the reactor vessel head vent paths are OPERABLE and closed in MODES 1, 2, 3, 4. The reactor vessel head vents are provided to exhaust non-condensable gases and steam from the reactor coolant system that could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a Loss of Coolant Accident (LOCA). The Surveillance requirements assure that each reactor vessel head vent path is demonstrated to be OPERABLE at least once every 18 months. The function, capabilities and testing requirements are consistent with the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements", action plan item II.B.1. (Reference UFSAR 5.4.15 and 7A).

3.7 TS 3/4.7.2, "Steam Generator Pressure/Temperature Limitation"

The steam generator (SG) pressure and temperature limits ensure that pressure induced stresses in the SGs do not exceed the maximum allowable fracture toughness stress limits. These pressure and temperature limits are based on maintaining the SG nil-ductility reference temperature, RT_{NDT}, sufficient to prevent brittle fracture.

The reactor coolant system pressure boundary, including the steam generators, meet the design requirements of 10CFR50, Appendix A, GDC 30 and 31. (Reference UFSAR 3.1.2.4.1 and 3.1.2.4.2)

3.8 TS 3/4.7.10, "Sealed Source Contamination"

This Specification provides limitations on sealed source contamination to ensure the total body and individual organ irradiation doses do not exceed allowable intake limits in the event of ingestion or inhalation. The maximum limitation of < 0.005 microCuries of removable contamination on each sealed source is based on 10CFR70.39(a)(3) limits for plutonium. This requirement and the associated surveillance requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

3.9. TS 3/4.9.3, "Decay Time"

This Specification requires the reactor to be sub-critical at least 42 hours prior to the movement of irradiated fuel assemblies in the reactor vessel. This ensures that sufficient time will elapse to allow the radioactive decay of the short-lived fission products. The activities necessary prior to commencing movement of irradiated fuel (e.g., containment entry, reactor vessel head removal, cavity flood-up) ensure that there will always be > 42 hours (Decay Time limit) of sub-criticality before movement of any irradiated fuel, therefore this requirement is not needed in Technical Specifications.

3.10 TS 3/4.9.5, "Communications"

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in facility status or core reactivity conditions during CORE ALTERATIONS. The communications allow for coordinating of activities that require interaction between the control room and containment personnel.

3.11 TS 3/4.9.7, "Crane Travel - Fuel Handling Building"

This restriction prohibits loads over the fuel assemblies in the fuel storage pool in excess of the nominal weight of a fuel assembly (including control rod assembly and handling tool) unless carried by the Fuel Handling Building crane 15-ton hoist (single-failure proof). This ensures that in the event the load is dropped, the activity release will be limited to that contained in a single fuel assembly and any possible distortion of the fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses. Administrative controls support these limits for moving loads over the spent fuel pool however crane travel is not monitored during operation but checked on a periodic basis. The deterministic criteria for inclusion in Technical Specifications is therefore, not satisfied.

3.12 Administrative Changes

The changes are completely administrative in nature. The proposed Technical Specification changes correct various typographical and page-numbering errors, delete an outdated one-time exception, and make minor format changes to improve consistency. No actual plant equipment or accident analyses will be affected by these proposed changes.

4.0 TECHNICAL ANALYSIS

The changes requested are administrative changes only. No actual plant equipment or safety analyses will be affected by these changes. The operational limits affected by these changes will not be relaxed, merely relocated. The UFSAR will not be affected by these changes where the design and operational characteristics will remain unchanged by the proposed request.

These systems, equipment or limits contained in these Specifications were each found to be a non-significant risk contributor to core damage frequency and offsite releases as summarized in WCAP-11618 - Westinghouse Owners Group MERITS Program – Phase II Task 5 (Reference 1).

The evaluations performed in support of this License Amendment Request will result in administrative changes to the Updated Final Safety Analysis (UFSAR) to reference the requirements of the Technical Requirements Manual instead of Technical Specifications. Design, operation, and control of the applicable equipment/systems will continue to meet the requirements described in the UFSAR.

These changes are consistent with NUREG-1431, "Standard (Improved) Technical Specifications - Westinghouse Plants".

4.1 TS 3/4.1.3.3, "Position Indication System – Shutdown" and Special Test Exception TS 3/4 10.5

With the reactor sub-critical and no significant fission power being produced, the requirement to have rod position indication is unnecessary. The safety functions associated with the shutdown or control rods are the ability of the rods to be tripped and negative reactivity that can be inserted from rods that are withdrawn. The rod indication Specification in MODES 3, 4, or 5, is not required. Worst-case failure modes of the position indication system are postulated in the accident analyses in UFSAR Chapter 15 concluding that there is no clad damage and no release of fission products to the RCS. (UFSAR 7.7.2)

4.2 TS 3/4.3.3.2, "Movable Incore Detectors"

The detectors are used for periodic surveillance of the power distribution, and calibration of the excore detectors, but are not assumed in any design basis accident analysis and do not mitigate an accident.

4.3 TS 3/4.3.3.11, "Explosive Gas Monitoring Instrumentation" and TS 3/4.11.2.5 "Explosive Gas Mixture"

The explosive gas monitoring instrumentation and explosive gas mixture limit are provided to ensure that the gaseous waste processing system is not operated with high oxygen concentrations. The concentration of oxygen in the gaseous waste processing system is not an initial assumption of any design basis accident or transient analysis. The system is not required to function post-accident and is isolated on phase 'A' Containment isolation.

4.4 TS 3/4.4.7, "Chemistry"

Poor RCS water chemistry contributes to the long-term degradation of system materials of construction. The effects of exceeding the limits are time and temperature dependent. The chemistry monitoring activity is of a long-term preventative purpose rather than mitigating. The surveillance requirements provide assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

4.5 TS 3/4.4.9.2, "Pressure/Temperature Limits - Pressurizer"

The heat-up and cool-down rate temperature limits are placed on the pressurizer to limit the cyclic, thermal loading on critical areas in the pressure boundary. The limits on the rate of change of temperature have been established using approved methodology, to preclude operation in an unanalyzed condition. These limits are not initial condition assumptions of a design basis accident or transient.

The pressurizer operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME BPV Code, Section III, Appendix G requirements.

4.6 TS 3/4.4.11, "Reactor Vessel Head Vents"

The reactor vessel head vents are provided to exhaust non-condensable gases and steam from the reactor coolant system that could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a LOCA. The function, capabilities and testing requirements are consistent with the requirements of NUREG-0737 as described in UFSAR sections 5.4.15 and 7A. Operation of the reactor vessel head vents is not part of a primary success path. The operation of these vents is an operator action after the event has occurred, and is only required when there is indication that natural circulation is not occurring. A break in the reactor vessel head vent piping is bounded by the small break LOCA analysis as described in the UFSAR 5.4.15.3.

4.7 TS 3/4.7.2, "Steam Generator Pressure/Temperature Limitation"

The steam generator (SG) pressure and temperature limits ensure that pressure induced stresses in the SGs do not exceed the maximum allowable fracture toughness stress limits. These pressure and temperature limits are based on maintaining the SG nil-ductility reference temperature, RT_{NDT}, sufficient to prevent brittle fracture.

These limits are not initial condition assumptions of a design basis accident or transient. These limits represent operating restrictions, but these restrictions would not preclude an unanalyzed accident or transient.

4.8 TS 3/4.7.10, "Sealed Source Contamination"

This Specification and associated surveillance requirements assure periodic inventory of sealed sources and monitoring for loose contamination to ensure that total body and individual organ irradiation doses as a result of leakage do not exceed allowable intake limits. The limit is based on 10CFR70.39(a)(3) limits for plutonium. This requirement and the associated surveillance requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation. These controls are also described in UFSAR section 12.5.3.7.

4.9 TS 3/4.9.3, "Decay Time"

This Specification requires the reactor to be sub-critical at least 42 hours prior to the movement of irradiated fuel assemblies in the reactor vessel to ensure that sufficient time will elapse to allow the radioactive decay of the short-lived fission products. The activities necessary prior to commencing movement of irradiated fuel (e.g., containment entry, reactor vessel head removal, cavity flood-up) ensure that there will always be > 42 hours (Decay Time limit) of sub-criticality before movement of any irradiated fuel. The Industry/NRC agreed during the development of NUREG-1431 that this Limiting Condition for Operation could be relocated since it is not required to be in Technical Specifications to provide adequate protection of the health and safety of the public.

The safety analysis for the Fuel Handling Accident assumes the accident occurs 42 hours after shutdown as described in UFSAR 15.7.4, consistent with Regulatory Guide 1.25, Assumptions Used For Evaluating The Potential Radiological Consequences Of A Fuel Handling Accident In The Fuel Handling And Storage Facility For Boiling And Pressurized Water Reactors.

4.10 TS 3/4.9.5, "Communications"

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in facility status or core reactivity conditions during CORE ALTERATIONS. However, the refueling system design accident or transient response does not take credit for communications. The communications allow activities to be coordinated that require interaction between the control room and containment personnel.

4.11 TS 3/4.9.7, "Crane Travel - Fuel Handling Building"

This restriction prohibits loads over the fuel assemblies in the spent fuel pool in excess of the nominal weight of a fuel assembly (including control rod assembly and handling tool) unless carried by the Fuel Handling Building (FHB) crane 15-ton hoist (single-failure proof). This ensures that in the event the load is dropped, the activity release will be limited to that contained in a single fuel assembly and any possible distortion of the fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses. Administrative controls support these limits for moving loads over the spent fuel pool however crane travel is not monitored during operation but checked on a periodic basis. The deterministic criteria for inclusion in Technical Specifications is therefore, not satisfied.

The FHB 15/2-ton crane is used for equipment handling with the capability of travelling over the spent fuel pool. The handling of fuel assemblies in the spent fuel pool is done with fuel handling tools suspended from the Fuel Handling Machine hoist. The FHB 15/2 ton crane is described in UFSAR sections 9.1.4.2.4.12 and 9.1.4.3.1.6. The Fuel-Handling Machine is described in UFSAR 9.1.4.3.1.3.

4.12 Administrative Changes

The changes to correct typographical errors and format changes are completely administrative in nature and will not affect the heath and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Determination

Pursuant to 10 CFR 50.92, it has been determined that this proposed amendment involves no significant hazards consideration. This determination was made by applying the Nuclear Regulatory Commission established standards contained in 10CFR50.92. These standards assure that operation of South Texas Project in accordance with this request consider the following:

1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   This request involves relocation of information to the Technical Requirements Manual and administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   This request involves relocation of information to the Technical Requirements Manual and administrative changes only. The proposed change does not alter the performance of the equipment or the manner in which the equipment will be operated. The equipment will still be verified by test, if applicable, in accordance with applicable surveillance requirements. Changing the location of these requirements and surveillances from Technical Specifications to the Technical

Requirements Manual will not create any new accident initiators or scenarios. Since the proposed changes only allow activities that are presently approved and conducted, no possibility exists for a new or different kind of accident from those previously evaluated.

3) Will the change involve a significant reduction in a margin of safety?

Response: No

This request involves relocation of information to the Technical Requirements Manual and administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety systems settings, or will not relax the bases for any limiting conditions of operation. Therefore, the proposed changes will not impact the margin of safety.

Conclusion

Based on the above analysis, STPNOC concludes that the proposed amendment to relocate these requirements from Technical Specifications to the TRM present no significant hazards consideration under the standards set forth in 10CFR50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

These changes are administrative changes only, therefore, the equipment will continue to meet the existing regulatory requirements as described in Sections 5.2.1 through 5.2.9 below.

The proposed changes relocate the requirements of above noted Technical Specifications including Surveillance requirements. The Technical Specifications do not meet the criteria for inclusion in Technical Specifications as identified in 10CFR50.36(c)(2)(ii) or 10CFR50.36(c)(3) with the exception of TS 3/4.9.3, "Decay Time".

TS 3/4.9.3, "Decay Time" satisfies Criterion 2 of 10CFR50.36(c)(2)(ii) in that it is a process variable that is an initial condition to a design basis accident that presents a challenge to the integrity of a fission product barrier. The Industry/NRC agreed during the development of NUREG-1431, that this Limiting Condition for Operation could be relocated since it is not required to be in Technical Specifications to provide adequate protection of the health and safety of the public. Hence, this Specification along with the others identified should be relocated to the TRM, consistent with NUREG-1431.

5.2.1 TS 3/4.1.3.3, "Position Indication System – Shutdown" and Special Test Exception TS 3/4 10.5

The plant control systems including the rod position indicating system are designed to conform to 10CFR50, Appendix A, General Design Criterion (GDC) 13 as described in UFSAR 7.7.2.

5.2.2 TS 3/4.3.3.2, "Movable Incore Detectors"

The plant control systems including the movable incore detection system are designed to conform to 10CFR50, Appendix A, GDC 13 as described in UFSAR 7.7.2.

5.2.3 TS 3/4.3.3.11, "Explosive Gas Monitoring Instrumentation" and TS 3/4.11.2.5 "Explosive Gas Mixture"

The gaseous waste processing system is designed in accordance with 10CFR50, Appendix I as described in UFSAR 11.3. Materials for piping, valves, and components handling radioactive gases conform to the requirements of specifications for materials listed in Section II of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, 1974, as per BTP ETSB 11-1, Rev. 1. (UFSAR 11.3.2.8.1)

5.2.4 TS 3/4.4.7, "Chemistry"

The RCS Chemistry specifications and guidelines described in UFSAR 5.2.3.2.1 are designed to reduce the possibility of failures in the reactor coolant system pressure boundary caused by corrosion.

5.2.5 TS 3/4.4.9.2, "Pressure/Temperature Limits - Pressurizer"

Pressurizer temperature limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The temperature and pressure changes during heat-up and cool-down are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure" as described in UFSAR 5.4.10. The reactor coolant system pressure boundary, including the pressurizer, meet the design requirements of 10CFR50, Appendix A, GDC 30, Quality of Reactor Coolant Pressure Boundary and 31, Fracture Prevention of Reactor Coolant Pressure Boundary. (UFSAR 3.1.2, Criterion Conformance, UFSAR 3.9.1.4.5, Analysis of Primary Components.)

5.2.6 TS 3/4.4.11, "Reactor Vessel Head Vents"

The reactor vessel head vent system provides for venting the reactor vessel head by using only safety-related equipment in accordance with the requirements of Three Mile Island (TMI) action plan item II.B.1 of NUREG-0737. The system also provides a safety grade letdown path for safety grade cold shutdown (Appendix 5.4.A). The reactor vessel head vent system satisfies applicable requirements and industry standards, including ASME Code classification, safety classification, single-failure criteria, and environmental qualification as described in UFSAR sections 5.4.15 and 7A.

5.2.7 TS 3/4.7.2, "Steam Generator Pressure/Temperature Limitation"

The steam generator (SG) pressure and temperature limits ensure that pressure induced stresses in the SGs do not exceed the maximum allowable fracture toughness stress limits. Like the pressurizer temperature limits of TS 3/4.4.9.2, the SG limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements. The reactor coolant system pressure boundary, including the SGs, meet the design requirements of 10CFR50, Appendix A, GDC 30 and 31.

5.2.8 TS 3/4.7.10, "Sealed Source Contamination"

This Specification and associated surveillance requirements assure sealed sources are periodically monitored for loose contamination to ensure that total body and individual organ irradiation doses as a result of leakage do not exceed allowable intake limits. The limit is based on 10CFR70.39(a)(3) limits for plutonium. These controls are described in UFSAR section 12.5.3.7.

5.2.9 TS 3/4.9.3, "Decay Time"

This Specification requires the reactor to be sub-critical at least 42 hours prior to the movement of irradiated fuel assemblies in the reactor vessel to ensure that sufficient time will elapse to allow the radioactive decay of the short-lived fission products. The activities necessary prior to commencing movement of irradiated fuel (e.g., containment entry, reactor vessel head removal, cavity flood-up) ensure that there will always be > 42 hours (Decay Time limit) of sub-criticality before movement of any irradiated fuel. The Industry/NRC agreed during the development of NUREG-1431 that this Limiting Condition for Operation could be relocated since it is not required to be in Technical Specifications to provide adequate protection of the health and safety of the public.

The 42 hours of decay assures in part that adequate shielding is provided to plant personnel to limit exposure from spent fuel assemblies as described in UFSAR

9.1.2.1 and required by GDC 61. The safety analyses for the Fuel Handling Accident assume the accident occurs 42 hours after shutdown as described in UFSAR 15.7.4, consistent with Regulatory Guide 1.25.

5.2.10 TS 3/4.9.5, "Communications"

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in facility status or core reactivity conditions during CORE ALTERATIONS. The communications systems used by refueling personnel are described in UFSAR 9.5.2.2.1.

5.2.11 TS 3/4.9.7, "Crane Travel - Fuel Handling Building"

This restriction prohibits loads over the fuel assemblies in the spent fuel pool in excess of 2500 lbs unless carried by the Fuel Handling Building (FHB) crane 15-ton hoist. In the event the load is dropped, the activity released would be limited to that contained in a single fuel assembly and any possible distortion of the fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses. Administrative controls support these limits for moving loads over the spent fuel pool however crane travel is not monitored during operation but checked on a periodic basis. The deterministic criteria for inclusion in Technical Specifications is therefore, not satisfied. The design and operation of the FHB 15/2 ton crane is described in UFSAR sections 9.1.4.2.4.12, and 9.1.4.3.1.6. The Fuel-Handling Machine is described in UFSAR 9.1.4.3.1.3.

5.12 Administrative Changes

The proposed changes are to correct various typographical and page-numbering errors, to delete an outdated one-time exception, and to make minor format changes to improve consistency. The changes are completely administrative in nature, therefore, the changes do not impact the health and safety of the public.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by the operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10CFR51.22, an evaluation of this request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR51.22(c)(9) and (c)(10) of the regulations.

This request will have no adverse radiation impact upon the environment. It has been determined that the proposed changes involve:

1. No significant hazards consideration,
2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and
3. No significant increase in individual or cumulative occupational radiation exposures.

Therefore, this request for revision of the Technical Specifications meets the criteria of 10CFR51.22 for categorical exclusion from the requirement for an environmental assessment.

7.0 REFERENCES

1. WCAP-11618 - Westinghouse Owners Group "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program – Phase II Task 5, Criteria Application", dated November 1987.
2. NOC-AE-01001146, "Proposed Amendment to South Texas Project Technical Specifications to Revise Administrative Requirements".

8.0 PRECEDENTS

As noted above, the relocation of these Technical Specifications to the Technical Requirements Manual is consistent with NUREG 1431 which has been implemented by a number of Westinghouse Plants".

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (MARK-UP)

(Bases sections provided for information only)

Note to Reviewer: Pages 3/4 4-16b included in the marked-up pages for this proposed amendment request, is also being proposed to be revised in accordance with a separate amendment request in Letter NOC-AE-01001146 which revises Section 6.0 of Technical Specifications, Administrative Controls where TS 6.9.2 has been deleted. (See Attachment 1, section 7.0, Reference 2).

Note to Reviewer: Pages 3/4 3-48 included in the marked-up pages for this proposed amendment request, is also being proposed to be revised in accordance with a separate amendment request in Letter NOC-AE-01001055 which modifies Technical Specification Table 4.3-2.

Relocation to TRM and Administrative Changes

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

BASES

BASES

Administrative Change

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowed outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:~~*~~

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i);

~~*The Inservice testing requirement for exercise testing in the closed direction for the following listed valves shall not be required until the next plant shutdown to Mode 5 of sufficient duration to allow the testing or until the next refueling outage scheduled in March 1999. This exception shall apply to the following Unit 1 valves only: 1-CC-0319, 1-CV-0034A, 1-CV-0034B, 1-CV-0034C, 1-CV-0034D,1-CV-0026, 1-FP-0943, and 1-IA-0541.~~

Administrative Change

3/4.1.2 (This specification is not used.)

Pages 3/4 1-~~9~~ 10 through 3/4 1-15 have been deleted.

Relocation to TRM

~~POSITION INDICATION SYSTEMS – SHUTDOWN~~

~~LIMITING CONDITION FOR OPERATION~~

3.1.3.3 (This specification not used) ~~One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.~~

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

~~ACTION:~~

~~With less than the above required position indicator (s) OPERABLE, immediately open the Reactor Trip System breakers.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.1.3.3 Each of the above required digital rod position indicator (s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at least once per 18 months.~~

~~* With the Reactor Trip System breakers in the closed position.~~
~~** See Special Test Exceptions Specification 3.10.5.~~

Relocation to TRM
Editorial change to Surveillance 4.2.4.2 due to relocation of TS 3/4.3.2.2 Movable Incore Detectors to the TRM

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoint within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ (by F_{xy} evaluation) and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by either:

a. Using the four pairs of symmetric thimble locations ~~(Specification 3.3.3.2.a does not apply)~~, or

b. Using the movable incore detection system to monitor the QUADRANT POWER TILT RATIO ~~subject to the requirements of Specification 3.3.3.2~~ **with a full incore map**.

* See Special Test Exceptions Specification 3.10.2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table ~~4.3.2~~ 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one train so that:

a. Each logic train is verified at least once per 36 months,

b. Each actuation train is verified at least once per 54 months*, and

c. One channel per function so that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

---

*If an ESFAS instrumentation channel is inoperable due to response times exceeding the required limits, perform an engineering evaluation to determine if the verification failure is a result of degradation of the actuation relays. If degradation of the actuation relays is determined to be the cause, increase the ENGINEERED SAFETY FEATURES RESPONSE TIME surveillance frequency such that all trains are verified at least once per 36 months.

Administrative Change
See Note to Reviewer on page 1 of Attachment 2.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Control Room Ventilation (Continued)								
b. Safety Injection	See Item 1. above all Safety Injection Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(6)	N.A.	N.A.	All
d. Control Room Intake Air Radioactivity-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	All
e. Loss of Power	See Items 8. above for all Loss of Power Surveillance Requirements.							
11. FHB HVAC								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(6)	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool.

Note: With the removal of Technical Specification 3/4.3.3.2, as part of this submittal it allows deletion of blank pages 3/4 3-55 through 3/4 7-60 within Section 3/4.3

Relocation to TRM

~~LIMITING CONDITION FOR OPERATION~~

3.3.3.2 through 3.3.3.4 (These specifications are not used) ~~The Movable Incore Detection System shall be OPERABLE with:~~

Pages 3/4 3-55 through 3/4 3-60 have been deleted.

~~a. At least 75% of the detector thimbles,~~

~~b. A minimum of two detector thimbles per core quadrant, and~~

~~c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.~~

~~APPLICABILITY: When the Movable Incore Detection System is used for:~~

~~a. Recalibration of the Excore Neutron Flux Detection System, or~~

~~b. Monitoring the QUADRANT POWER TILT RATIO, or~~

~~c. Measurement of~~ $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}

~~ACTION:~~

~~With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:~~

~~a. Recalibration of the Excore Neutron Flux Detection System, or~~

~~b. Monitoring the QUADRANT POWER TILT RATIO, or~~

~~c. Measurement of~~ $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}

Note: With the removal of Technical Specification 3/4.3.3.11, as part of this submittal it allows deletion of blank pages 3/4 3-76 through 3/4 7-84 within Section 3/4.3

Relocation to TRM and Administrative Changes

~~EXPLOSIVE GAS MONITORING INSTRUMENTATION~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.3.3.11~~ 3.3.3.7 through 3.3.3.11 and 3.3.4 (These specifications are not used) ~~The explosive gas monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.~~

Pages 3/4 3-76 through 3/4 7-84 have been deleted.

~~APPLICABILITY: As shown in Table 3.3-13~~

~~ACTION:~~

~~a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.~~

~~b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.~~

~~c. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.3.3.11 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST or DIGITAL CHANNEL OPERATIONAL TEST, as applicable, at the frequencies shown in Table 4.3-9.~~

Relocation to TRM

This page will be deleted

~~TABLE 3.3-13~~

~~EXPLOSIVE GAS MONITORING INSTRUMENTATION~~

~~INSTRUMENT~~	~~MINIMUM CHANNELS OPERABLE~~	~~APPLICABILITY~~	~~ACTION~~
~~1. GASEOUS WASTE PROCESSING SYSTEM~~			
~~Explosive Gas Monitoring System~~			
~~Oxygen Monitor (Process)~~	~~1~~	~~**~~	~~51~~

Relocation to TRM

This page will be deleted

~~TABLE 3.3-13 (Continued)~~

~~TABLE NOTATIONS~~

~~* (Not Used)~~
~~** During GASEOUS WASTE PROCESSING SYSTEM operation.~~

~~ACTION STATEMENTS~~

~~ACTION 47 - (Not used)~~

~~ACTION 48 - (Not used)~~

~~ACTION 49 - (Not used)~~

~~ACTION 50 - (Not used)~~

~~ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this GASEOUS WASTE PROCESSING SYSTEM may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.~~

~~ACTION 52 - (Not used)~~

~~ACTION 53 - (Not used)~~

Relocation to TRM

This page will be deleted

~~TABLE 4.3-9~~

~~EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

~~INSTRUMENT~~	~~CHANNEL CHECK~~	~~SOURCE CHECK~~	~~CHANNEL CALIBRATION~~	~~ANALOG OR DIGITAL CHANNEL OPERATIONAL TEST~~	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~
~~1. GASEOUS WASTE PROCESSING SYSTEM~~					
~~Explosive Gas Monitoring System~~					
~~Oxygen Monitor (Process)~~	~~D~~	~~N.A.~~	~~Q(5)~~	~~M~~	~~**~~

Relocation to TRM

This page will be deleted

~~TABLE 4.3-9 (Continued)~~

~~TABLE NOTATIONS~~

~~* (Not used)~~

~~** During GASEOUS WASTE PROCESSING SYSTEM operation.~~

~~(1) (Not Used)~~

~~(2) (Not Used)~~

~~(3) (Not Used)~~

~~(4) (Not Used)~~

~~(5) The CHANNEL CALIBRATION shall include the use of a standard gas sample containing a nominal two volume percent oxygen, balance nitrogen.~~

3) A tube inspection (pursuant to Specification 4.4.5.4a.9) 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.

4) For Model E steam generators only, indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Table 4.4-2 or Table 4.4-3) during each inservice inspection may be subjected to a partial tube inspection provided:

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, and

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

d. For Model E steam generators only, implementation of the steam generator tube/tube support plate repair criteria requires a 100-percent bobbin coil inspection for the flow distribution baffle plate intersections, for the hot-leg tube support plate intersections, and for the cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

1) All intersections with mechanically induced dent signals greater than 5 volts identified by bobbin coil inspection shall be inspected by rotating pancake coil (or equivalent).

2) All intersections with large mixed residuals that could potentially mask flaw responses at or above the voltage repair limits shall be inspected by rotating pancake coil (or equivalent).

3) At the flow distribution baffle intersections, at the cold-leg support plate intersections, and at the hot-leg support plate intersections with support plates L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), tubes with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (defined in 4.4.5.4.a.11) shall be inspected by rotating pancake coil (or equivalent).

Administrative Change

SURVEILLANCE REQUIREMENTS (Continued)

4) At the hot-leg support plate intersections with support plates C, F, and J (as identified in Figure 5.1 of WCAP-15163, Revision 1), all tubes with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 3 volts shall be inspected by rotating pancake coil (or equivalent) eddy current probe. An additional 100 tube intersections with support plates C, F, and J with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage less than 3 volts (100 total over all steam generators, not necessarily selected at random) shall be inspected by rotating pancake coil (or equivalent).

The results of each sample inspection shall be classified into one of the following three categories.

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

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

10) Preservice Inspection 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 shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

11) For Model E steam generators only, Tube Support Plate Plugging Limit is used for the disposition of a mill annealed alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates.

At the flow distribution baffle intersections, at the cold leg support plate intersections, and at the hot leg support plate intersections with support plates L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in a), b), c) and d) below:

a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit (Note 1), will be allowed to remain in service.

b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1), will be repaired or plugged, except as noted in 4.4.5.4.a. 11.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1) but less than or equal to the upper repair voltage limit (Note 2), may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with bobbin voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

d) If an unscheduled mid-cycle inspection is performed, the mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c. The mid-cycle repair limits will be determined from the equations for mid-cycle repair limits of NRC Generic Letter 95-05, Attachment 2, page 3 of 7. Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing.

Note 2: The upper voltage repair limit (V_{URL}) is calculated for each inspection according to the methodology in Generic Letter 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle. Voltage growth rate shall be the larger of the average growth rates experienced in the two prior cycles, but not less than 30% per effective full power year.

For Unit 2 Cycle 9 only, at the hot leg support plate intersections with support plates C, F, and J (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in e), f), and g) below:

e) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage less than or equal to 3.0 volts may remain in service.

f) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 3.0 volts shall be plugged or repaired regardless of whether or not a rotating pancake coil inspection detects degradation.

g) If one or more indications in the tube support plate intersections are confirmed by non-destructive examination to extend beyond the edge of the tube support plate, the 3-volt alternate repair criteria shall not be used in any steam generator. Exceptions to this requirement may be allowed for those indications that are determined by the NRC staff to be physically insignificant for the purposes of safety and risk assessment. Approval for the use of the 3-volt alternate repair criteria may be granted by the staff in writing on a one-time basis, following the staff review and consideration of the factors related to the crack extensions that are found.

12) Tube Repair refers to a process that reestablishes tube serviceability for Model E steam generators only. Acceptable tube repair will be performed in accordance with the methods described in Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, "April 1995 and WCAP-14653, "Specific Application of Laser Welded Sleeves for South Texas Project Power Plant Steam Generators, "June 1996, including post-weld stress relief;

Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

Administrative Change
See Note to reviewer on page 1 of Attachment 2.

SURVEILLANCE REQUIREMENTS (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions [plug or (for Model E steam generators only) repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks] required by Table 4.4-2 and Table 4.4-3.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which-fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Model E steam generators, implementation of the voltage-based repair criteria to tube support plate intersections, notify the Staff prior to returning the steam generators to service should any of the following conditions arise:

1) If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle. The calculation shall be done using:

a) The methodology of Generic Letter 95-05 for intersections at the flow distribution baffles, at the applicable cold leg support plates, and at the hot-leg support plates L through R; and

Administrative Change

TABLE ~~Table~~ 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug 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. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	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. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	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.

Administrative Change

TABLE ~~Table~~ 4.4-3

MODEL E STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes [1]	C-1	None	N. A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective repaired tubes and inspect 20% of the repaired tubes in each other S.G.	All other S.G.s are C-1	None
			Some SGs. C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample
		Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	Additional S. G. is C-3	Inspect all repaired tubes in each S. G. and plug defective repaired tubes. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)

(1) Each repair method is considered a separate population for determination of scope expansion.

Relocation to TRM

~~3/4.4.7 CHEMISTRY~~

~~LIMITING CONDITION FOR OPERATION~~

3.4.7 (This specification not used) ~~The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.~~

~~APPLICABILITY: At all times.~~

Pages 3/4 4-24 and 3/4 4-25 have been deleted

~~ACTION:~~

~~MODES 1, 2, 3, and 4:~~

~~a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and~~

~~b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~At All Other Times:~~

~~With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.~~

Relocation to TRM

This Page to be deleted

~~TABLE 3.4-2~~

~~REACTOR COOLANT SYSTEM~~

~~CHEMISTRY LIMITS~~

~~PARAMETER~~	~~STEADY-STATE LIMIT~~	~~TRANSIENT LIMIT~~
~~Dissolved Oxygen*~~	~~< 0.10 ppm~~	~~≤ 1.00 ppm~~
~~Chloride~~	~~< 0.15 ppm~~	~~≤ 1.50 ppm~~
~~Fluoride~~	~~≤ 0.15 ppm~~	~~≤ 1.50 ppm~~

~~*Limit not applicable with T_{avg} less than or equal to 250°F.~~

Relocation to TRM

This Page to be deleted

~~TABLE 4.4-3~~

~~REACTOR COOLANT SYSTEM~~

~~CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS~~

~~PARAMETER~~	~~SAMPLE AND ANALYSIS FREQUENCY~~
~~Dissolved Oxygen*~~	~~At least once per 72 hours~~
~~Chloride~~	~~At least once per 72 hours~~
~~Fluoride~~	~~At least once per 72 hours~~

~~*Not required with T_{avg} less than or equal to 250°F.~~

REACTOR COOLANT SYSTEM

~~PRESSURIZER~~

~~LIMITING CONDITION FOR OPERATION~~

3.4.9.2 (This specification not used) ~~The pressurizer temperature shall be limited to:~~

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

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

~~c. A maximum spray water temperature differential of 621°F.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

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

~~SURVEILLANCE REQUIREMENTS~~

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

Relocation to TRM

~~3/4.4.11 REACTOR VESSEL HEAD VENTS~~

~~LIMITING CONDITION FOR OPERATION~~

3.4.11 (This specification not used) ~~Two reactor vessel head vent paths each consisting of two vent valves and a control valve powered from emergency busses shall be OPERABLE and closed.~~

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

~~ACTION:~~

~~a. With one of the above reactor vessel head vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuators of all the vent valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b. With two reactor vessel head vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:~~

~~a. Verifying all manual isolation valves in each vent path are locked in the open position,~~

~~b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and~~

~~c. Verifying flow through the reactor vessel head vent paths during venting.~~

~~3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION~~

~~LIMITING CONDITION FOR OPERATION~~

3.7.2 (This specification not used) ~~The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~With the requirements of the above specification not satisfied:~~

~~a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and~~

~~b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.~~

Note: With the removal of Technical Specification 3/4.7.10, as part of this submittal it allows deletion of numerous blank pages within Section 3/4.7.

Relocation to TRM and Administrative Changes

3/4.7.9 ~~(Not Used)~~ through 3/4.7.13 (These specification numbers are not used.)

Pages 3/4 7-22 through 3/4 7-32 have been deleted.

Note: With this Specification deleted, it allows removal of blank pages 3/4 7-22 through 3/4 7-32.

Relocation to TRM and Administrative Changes

~~3/4.7.10 SEALED SOURCE CONTAMINATION~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.10 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.~~ [3/4.7.9 through 3/4.7.13 (These specification numbers are not used.)]

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:~~

~~1. Decontaminate and repair the sealed source, or~~

~~2. Dispose of the sealed source in accordance with Commission Regulations.~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.10.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:~~

~~a. The licensee, or~~

~~b. Other persons specifically authorized by the Commission or an Agreement State.~~

~~The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.~~

~~4.7.10.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.~~

~~a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:~~

~~1) With a half-life greater than 30 days (excluding Hydrogen 3), and~~

~~2) In any form other than gas.~~

(Next page is 3/4 7-33)

Relocation to TRM and Administrative Changes

This page deleted

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and~~

~~c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.~~

~~4.7.10.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.~~

Relocation to TRM

~~3/4.9.3 DECAY TIME~~

~~LIMITING CONDITION FOR OPERATION~~

3.9.3 (This specification not used.) ~~The reactor shall be subcritical for at least 42 hours.~~

~~APPLICABILITY: During movement of irradiated fuel in the reactor vessel.~~

~~ACTION:~~

~~With the reactor subcritical for less than 42 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.~~

~~LIMITING CONDITION FOR OPERATION~~

~~4.9.3 The reactor shall be determined to have been subcritical for at least 42 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.~~

REFUELING OPERATIONS

~~3/4.9.5 COMMUNICATIONS~~

~~LIMITING CONDITION FOR OPERATION~~

3.9.5 (This specification not used) ~~Direct communications shall be maintained between the control room and personnel at the refueling station.~~

~~APPLICABILITY: During CORE ALTERATIONS.~~

~~ACTION:~~

~~When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.~~

~~SURVEILLANCE REQUIREMENTS~~

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

Relocation to TRM

REFUELING OPERATIONS

~~3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING~~

~~LIMITING CONDITION FOR OPERATION~~

3.9.7 (This specification not used)~~ Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool except when carried by the single failure-proof 15-ton hoist of the FHB crane.~~

~~APPLICABILITY: With fuel assemblies in the spent fuel pool.~~

~~ACTION:~~

~~a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.7 Loads shall be verified less than or equal to 2,500 pounds prior to movement over fuel assemblies in the spent fuel pool unless they are carried by the single-failure-proof 15-ton hoist of the FHB crane.~~

Relocation to TRM
(Test exception for TS 3/4.1.3.3)

~~3/4.10.5 POSITION INDICATION SYSTEM – SHUTDOWN~~

~~LIMITING CONDITION FOR OPERATION~~

3.10.5 (This specification not used) ~~The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;~~

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

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

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

~~ACTION:~~

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

~~SURVEILLANCE REQUIREMENTS~~

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

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

~~b. Within 24 steps during rod motion.~~

~~* This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.~~

Relocation to TRM

3/4.11.2 GASEOUS EFFLUENTS

~~EXPLOSIVE GAS MIXTURE~~

~~LIMITING CONDITION FOR OPERATION~~

3.11.2.5 (This specification not used) ~~The concentration of oxygen in the GASEOUS WASTE PROCESSING SYSTEM inlet shall be limited to less than or equal to 3% by volume.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

a. ~~With the concentration of oxygen in the GASEOUS WASTE PROCESSING SYSTEM inlet exceeding the limit, restore the concentration to within the limit within 48 hours.~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~4.11.2.5 The concentration of oxygen in the GASEOUS WASTE PROCESSING SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases entering the GASEOUS WASTE PROCESSING SYSTEM with the oxygen monitor required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.~~

Proposed change to Bases 3/4.2.4
Clarification supports administrative change to Surveillance requirement 4.2.4.2.where an exception was provided for TS 3.3.2.2.a Movable Incore Detector operability.

3/4.2.4 QUADRANT POWER TILT RATIO - BASES section in shaded area to be added for clarification.

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

When using two sets of four symmetric thimbles to verify QUADRANT POWER TILT RATIO, the eight designated locations are the only detector thimbles required to be OPERABLE.

3/4.3.3.2 MOVABLE INCORE DETECTORS (Not Used) - BASES section in shaded areas to be Relocated to TRM

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $FN_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detector System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

The following text will also be ADDED to the BASES for Clarification:

The two sets of four symmetric thimbles are a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. When using two sets of four symmetric thimbles to verify QUADRANT POWER TILT RATIO, the eight designated locations are the only detector thimbles required to be OPERABLE.

3/4.3.3.11 EXPLOSIVE GAS MONITORING (Not Used) - BASES section in shaded areas to be Relocated to TRM

This instrumentation includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS WASTE PROCESSING SYSTEM.

3/4.4.7 CHEMISTRY (Not Used) - BASES section in shaded areas to be Relocated to TRM

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with containment concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS - BASES section in shaded areas to be Relocated to TRM

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3/4.4.9 PRESSURE/TEMPERATURE LIMITS - BASES section in shaded areas to be Relocated to TRM

Note: Items 3 and 4 and last paragraph of TS Bases 3/4.4.9 section to be deleted from the TS Bases and relocated to the TRM.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 621°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years (EFPY) of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT} Therefore, $_{\Delta}RT_{NDT}$ and an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the materials in question were computed using the method described in Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 32 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of $_{\Delta}RT_{NDT}$ determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H and new values of $_{\Delta}RT_{NDT}$ will be computed using the method described in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials". The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued) - BASES section in shaded areas to be Relocated to TRM

capsule. The heatup and cooldown curves must be recalculated when the $_{\Delta}RT_{NDT}$, determined from the surveillance capsule exceeds the calculated $_{\Delta}RT_{NDT}$ for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, $_{\Delta}RT_{NDT}$, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_1, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR}, is obtained from the reference and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individua1 basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup-ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.11 REACTOR VESSEL HEAD VENTS (Not Used) - BASES section in shaded areas to be Relocated to TRM

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of at least two reactor vessel head vent paths ensures that the capability exists to perform this function.

The valve redundancy of the reactor vessel head vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor vessel head vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION (Not Used) - BASES section in shaded areas to be Relocated to TRM

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NTD} of 10°F and are sufficient to prevent brittle fracture.

3/4.7.10 SEALED SOURCE CONTAMINATION (Not Used) - BASES section in shaded areas to be Relocated to TRM

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a) (3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.9.3 DECAY TIME (Not Used) - BASES section in shaded areas to be Relocated to TRM

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses for the rapid refueling design.

3/4.9.5 COMMUNICATIONS (Not Used) - BASES section in shaded areas to be Relocated to TRM

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING (Not Used) - BASES section in shaded areas to be Relocated to TRM

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool, unless handled by the single-failure-proof main hoist of the FHB 15-ton crane, ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.10.5 POSITION INDICATION SYSTEM – SHUTDOWN (Not Used) - BASES section in shaded areas to be Relocated to TRM

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.5 EXPLOSIVE GAS MIXTURE (Not Used) - BASES section in shaded areas to be Relocated to TRM

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the GASEOUS WASTE PROCESSING SYSTEM is maintained below the flammability limit of oxygen. The concentration of oxygen in the inlet header to the GASEOUS WASTE PROCESSING SYSTEM is continuously monitored and a high level alarm isolates the GASEOUS WASTE PROCESSING SYSTEM. Provision is made to manually purge the system with nitrogen and/or isolate the source of oxygen. Maintaining the concentration of oxygen below its flammability limit (4% by volume) provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.Because of analyzer variabilities, a safety margin of 1% by volume is applied. Therefore, the limiting condition for operation is maintaining oxygen concentration below 3% by volume.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION PAGES (RE-TYPED)

Note to Reviwer: Reconstituted Bases pages are not included in this submittal. STPNOC will submit these pages to the NRC under a separate submittal when the changes described in the enclosed License Amendment Request are approved by the NRC.

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BASES

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowed outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i);

3/4.1.2 (This specification not used)

Pages 3/4 1-10 through 3/4 1-15 have been deleted.

3.1.3.3 (This specification not used)

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoint within the next 4 hours.

c. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ (by F_{xy} evaluation) and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by either:

a. Using the four pairs of symmetric thimble locations, or

b. Using the movable incore detection system to monitor the QUADRANT POWER TILT RATIO with a full incore map.

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* See Special Test Exceptions Specification 3.10.2.

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one train so that:

a. Each logic train is verified at least once per 36 months,

b. Each actuation train is verified at least once per 54 months*, and

c. One channel per function so that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

*If an ESFAS instrumentation channel is inoperable due to response times exceeding the required limits, perform an engineering evaluation to determine if the verification failure is a result of degradation of the actuation relays. If degradation of the actuation relays is determined to be the cause, increase the ENGINEERED SAFETY FEATURES RESPONSE TIME surveillance frequency such that all trains are verified at least once per 36 months.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Control Room Ventilation	(Continued)							
b. Safety Injection	See Item 1. above all Safety Injection Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(6)	N.A.	N.A.	All
d. Control Room Intake Air Radioactivity-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	All
e. Loss of Power	See Item 8. above for all Loss of Power Surveillance Requirements.							
11. FHB HVAC								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(6)	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool

3.3.3.2 through 3.3.3.4 (These specifications are not used)

Pages 3/4 3-55 through 3/4 3-60 have been deleted.

3.3.3.7 through 3.3.3.11 and 3.3.4 (These specification numbers are not used)

Pages 3/4 3-76 through 3/4 3-84 have been deleted.

3) A tube inspection (pursuant to Specification 4.4.5.4a.9) 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.

4) For Model E steam generators only, indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Table 4.4-2 or Table 4.4-3) during each inservice inspection may be subjected to a partial tube inspection provided:

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, and

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

d. For Model E steam generators only, implementation of the steam generator tube/tube support plate repair criteria requires a 100-percent bobbin coil inspection for the flow distribution baffle plate intersections, for the hot-leg tube support plate intersections, and for the cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

1) All intersections with mechanically induced dent signals greater than 5 volts identified by bobbin coil inspection shall be inspected by rotating pancake coil (or equivalent).

2) All intersections with large mixed residuals that could potentially mask flaw responses at or above the voltage repair limits shall be inspected by rotating pancake coil (or equivalent).

3) At the flow distribution baffle intersections, at the cold-leg support plate intersections, and at the hot-leg support plate intersections with support plates L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), tubes with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (defined in 4.4.5.4.a.11) shall be inspected by rotating pancake coil (or equivalent).

4) At the hot-leg support plate intersections with support plates C, F, and J (as identified in Figure 5.1 of WCAP-15163, Revision 1), all tubes with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 3 volts shall be inspected by rotating pancake coil (or equivalent) eddy current probe. An additional 100 tube intersections with support plates C, F, and J with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage less than 3 volts (100 total over all steam generators, not necessarily selected at random) shall be inspected by rotating pancake coil (or equivalent).

The results of each sample inspection shall be classified into one of the following three categories.

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

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

10) Preservice Inspection 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 shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

11) For Model E steam generators only, Tube Support Plate Plugging Limit is used for the disposition of a mill annealed alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates.

At the flow distribution baffle intersections, at the cold leg support plate intersections, and at the hot leg support plate intersections with support plates L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in a), b), c) and d) below:

a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit (Note 1), will be allowed to remain in service.

b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1), will be repaired or plugged, except as noted in 4.4.5.4.a. 11.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1) but less than or equal to the upper repair voltage limit (Note 2), may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with bobbin voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

d) If an unscheduled mid-cycle inspection is performed, the mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c. The mid-cycle repair limits will be determined from the equations for mid-cycle repair limits of NRC Generic Letter 95-05, Attachment 2, page 3 of 7. Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing.

Note 2: The upper voltage repair limit (V_{URL}) is calculated for each inspection according to the methodology in Generic Letter 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle. Voltage growth rate shall be the larger of the average growth rates experienced in the two prior cycles, but not less than 30% per effective full power year.

For Unit 2 Cycle 9 only, at the hot leg support plate intersections with support plates C, F, and J (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in e), f), and g) below:

e) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage less than or equal to 3.0 volts may remain in service.

f) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 3.0 volts shall be plugged or repaired regardless of whether or not a rotating pancake coil inspection detects degradation.

g) If one or more indications in the tube support plate intersections are confirmed by non-destructive examination to extend beyond the edge of the tube support plate, the 3-volt alternate repair criteria shall not be used in any steam generator. Exceptions to this requirement may be allowed for those indications that are determined by the NRC staff to be physically insignificant for the purposes of safety and risk assessment. Approval for the use of the 3-volt alternate repair criteria may be granted by the staff in writing on a one-time basis, following the staff review and consideration of the factors related to the crack extensions that are found.

12) Tube Repair refers to a process that reestablishes tube serviceability for Model E steam generators only. Acceptable tube repair will be performed in accordance with the methods described in Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, "April 1995 and WCAP-14653, "Specific Application of Laser Welded Sleeves for South Texas Project Power Plant Steam Generators, "June 1996, including post-weld stress relief;

Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

SURVEILLANCE REQUIREMENTS (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions [plug or (for Model E steam generators only) repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks] required by Table 4.4-2 and Table 4.4-3.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

   1) Number and extent of tubes inspected,

   2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

   3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which-fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Model E steam generators, implementation of the voltage-based repair criteria to tube support plate intersections, notify the Staff prior to returning the steam generators to service should any of the following conditions arise:

   1) If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle. The calculation shall be done using:

      a) The methodology of Generic Letter 95-05 for intersections at the flow distribution baffles, at the applicable cold leg support plates, and at the hot-leg support plates L through R; and

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug 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. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	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. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	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 4.4-3

MODEL E STEAM GENERATOR REPAIRED TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes (1)	C-1	None	N. A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective repaired tubes and inspect 20% of the repaired tubes in each other S.G. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	All other S.G.s are C-1	None
			Some S.G.s. C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample
			Additional S. G. is C-3	Inspect all repaired tubes in each S. G. and plug defective repaired tubes. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)

(1) Each repair method is considered a separate population for determination of scope expansion.

3/4.4.7 (This specification not used)

Pages 3/4 4-24 and 3/4 4-25 have been deleted

3.4.9.2 (This specification not used)

3.4.11 (This specification not used)

3.7.2 (This specification not used)

3/4.7.9 through 3/4.7.13 (These specification numbers are not used)

Pages 3/4 7-22 through 3/4 7-32 have been deleted.

3.9.3 (This specification not used)

REFUELING OPERATIONS

3.9.5 (This specification not used)

3.9.7 (This specification not used)

SPECIAL TEST EXCEPTIONS

3/4.10.5 (This specification not used)

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

3.11.2.5 (This specification not used)