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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

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

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

ACTION:

NOTE: Separate entry is allowed for each steam generator tube

- a. With one or more steam generator tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program, within 7 days verify tube integrity of the affected tube(s) is maintained until the next inspection, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

AND

Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or steam generator tube inspection.

- b. With steam generator tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVELLANCE REQUIREMENTS

4.4.5.1 Verify steam generator tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected steam generator tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a steam generator tube inspection.

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SOUTH TEXAS - UNITS 1 & 2

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(Next page is 3/4 4-19)

Unit 1 – Amendment No. 164
Unit 2 – Amendment No. 154

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 gpm UNIDENTIFIED LEAKAGE,
- c. 150 gallons per day of primary-to-secondary leakage through any one steam generator,
- d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and
- e. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

- a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With Reactor Coolant System operational UNIDENTIFIED or IDENTIFIED LEAKAGE greater than the above limits, reduce leakage to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.1 Note: this requirement is not applicable to primary-to-secondary leakage (refer to 4.4.6.2.3).

Reactor Coolant System operational leakage shall be demonstrated to be within each of the above limits by:

- a. Monitoring the containment atmosphere gaseous radioactivity and particulate radioactivity channels at least once per 12 hours;
- b. Monitoring the containment normal sump inventory and discharge at least once per 12 hours;
- c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and ⁽¹⁾
- d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

- a. At least once per 18 months,
- b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,
- c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and
- d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve except for valves XRH0060 A, B, C, and XRH0061 A, B, C.

4.4.6.2.3 Primary-to-secondary leakage shall be verified \leq 150 gallons per day through any one steam generator at least once per 72 hours. ⁽¹⁾

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

(1) Not required to be performed until 12 hours after establishment of steady state operation.

6.0 ADMINISTRATIVE CONTROLS

6.8 Procedures, Programs, and Manuals

6.8.3.n (continued)

- 2) The ODCM shall also contain descriptions of the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radiological Effluent Release Report required by Specifications 6.9.1.3 and 6.9.1.4.
- 3) Licensee-initiated changes to the ODCM:
 - a) Shall be documented and records of reviews performed shall be retained.

This documentation shall contain:

 1. Sufficient information to support the changes together with the appropriate analyses or evaluations justifying the changes and
 2. A determination that the changes maintain the levels of radioactive effluent control required by 10 CFR 20.1 302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.
 - b) Shall become effective after approval of the plant manager.
 - c) Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (month and year) the change was implemented.

o. Steam Generator Program

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

6.0 ADMINISTRATIVE CONTROLS

6.8 Procedures, Programs, and Manuals

6.8.3.o (continued)

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.
- b. Performance criteria for SG tube integrity. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.
 1. Structural integrity performance criterion. All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 (3 Δ P) against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
 2. Accident induced leakage performance criterion. The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm total for all four SGs in one unit.

6.0 ADMINISTRATIVE CONTROLS

6.8 Procedures, Programs, and Manuals

6.8.3.o (continued)

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."
- c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged
- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.
 2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

6.0 ADMINISTRATIVE CONTROLS

6.8 Procedures, Programs, and Manuals

6.8.3.o (continued)

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

6.0 ADMINISTRATIVE CONTROLS
6.9 Reporting Requirements

6.9.1.6 (continued)

- c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any mid-cycle revisions or supplements, shall be provided to the NRC upon issuance for each reload cycle.

6.9.1.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.3.o. The report shall include:

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

6.9.2 Not Used