

5.5 Programs and Manuals

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5.5.4 Radioactive Effluent Controls Program (continued)

- i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and
- j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Table 5.2-2a, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC. The first performance of the IWL requirements for containment sample tendon force measurements and tendon wire and strand sample examinations will be performed by the end of 2006.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel at least once per 10 years by conducting either:

- a. An in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius: or
- b. A surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

(continued)

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5.5.7 Reactor Coolant Pump Flywheel Inspection Program (continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program.

5.5.8 Inservice Testing Program

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

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

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

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Program

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

- a. Provisions for condition monitoring assessments. Condition monitoring

(continued)

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### 5.5.9 Steam Generator (SG) Program (continued)

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. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
  1. Structural integrity performance criterion: All inservice 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 (3ΔP) 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 three SGs.
  3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

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(continued)

5.6 Reporting Requirements

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5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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

- a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.
- b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.
- c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

5.6.8 PAM Report

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

5.6.9 Deleted

5.6 Reporting Requirements

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5.6.10 Steam Generator (SG) Tube Inspection Report

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

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

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the requirements of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC (Ref. 4).

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REFERENCES

1. 10 CFR 50, Appendix J, Option B.
  2. FSAR, Chapter 15.
  3. FSAR, Section 6.2.
  4. Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a.
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