

## 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;
- j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.
- k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

### 5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 3.9(N), cyclic and transient occurrences to ensure that components are maintained within the design limits.

### 5.5.6 Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its 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 exemption or relief has been authorized by the NRC.

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 per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), conduct a qualified in place UT examination over the volume from the inner bore

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5.5.7 Reactor Coolant Pump Flywheel Inspection Program (continued)

of the flywheel to the circle of one-half the outer radius or conduct a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels once every ten years coinciding with the inservice inspection schedule as required by ASME Section XI.

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.

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

- d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

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### 5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

### 5.5.16 Containment Leakage Rate Testing Program

- a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
  1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.
  2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.
- b. The peak calculated containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is 48 psig.
- c. The maximum allowable containment leakage rate,  $L_a$ , at  $P_a$ , shall be 0.20% of containment air weight per day.
- d. Leakage rate acceptance criteria are:
  1. Containment leakage rate acceptance criterion is  $\leq 1.0 L_a$ . During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are  $< 0.60 L_a$  for the Type B and Type C tests and  $\leq 0.75 L_a$  for Type A tests;

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5.5.16 Containment Leakage Rate Testing Program (continued)

2. Air lock testing acceptance criteria are:
  - a) Overall air lock leakage rate is  $\leq 0.05 L_a$  when tested at  $\geq P_a$ .
  - b) For each door, leakage rate is  $\leq 0.005 L_a$  when pressurized to  $\geq 10$  psig.
- e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.
- f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Reactor Vessel Head Closure Bolt Integrity

This program provides the requirements to support normal plant operation with one reactor vessel head closure bolt less than fully tensioned for one operating cycle. The provisions of this program shall be implemented when a head closure bolt becomes stuck in a partially inserted position such that the amount of thread engagement is not sufficient to take the tensioning loads without damage to the vessel threads or a bolt is not capable of being inserted into the bolt hole.

Prior to operation with one reactor vessel head closure bolt less than fully tensioned, the following conditions shall apply:

- a. The circumstances associated with the less than fully tensioned closure bolt will be verified to be bounded by the analysis that was referenced in the letter dated September 15, 2000 (WO 00-0036).
- b. A review of the results of the visual examinations performed on the closure bolts shall be performed to ensure that there is no indication of sufficient degradation of closure bolts that could affect the conclusions of Specification 5.5.17a. above.

Within 30 days following startup of the plant, a report shall be submitted to the Commission identifying the circumstances for operation with one reactor vessel head closure bolt less than fully tensioned.

Operation with the same reactor vessel head closure bolt less than fully tensioned shall be limited to one operating cycle (i.e., until the next refueling outage).

## 5.0 ADMINISTRATIVE CONTROLS

### 5.6 Reporting Requirements

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The following reports shall be submitted in accordance with 10 CFR 50.4.

#### 5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

#### 5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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