

1.1 Definitions

---

LEAKAGE  
(continued)

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

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary-to-secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

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

OPERABLE-OPERABILITY

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

PHYSICS TESTS

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

---

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

- LCO 3.4.13            RCS operational LEAKAGE shall be limited to:
- a.     No pressure boundary LEAKAGE;
  - b.     1 gpm unidentified LEAKAGE;
  - c.     10 gpm identified LEAKAGE; and
  - d.     150 gallons per day primary-to-secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

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

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1</p> <p>-----NOTE-----</p> <ol style="list-style-type: none"> <li>1. Not required to be performed until 12 hours after establishment of steady state operation.</li> <li>2. Not applicable to primary-to-secondary LEAKAGE.</li> </ol> <p>-----</p> <p>Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.</p>	<p>72 hours</p>
<p>SR 3.4.13.2</p> <p>-----NOTE-----</p> <p>Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>-----</p> <p>Verify primary-to-secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG.</p>	<p>72 hours</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 STEAM GENERATOR (SG) TUBE INTEGRITY

LCO 3.4.17 SG tube integrity shall be maintained

AND

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

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	<u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met.  <u>OR</u>	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours
SG tube integrity not maintained		

**SURVEILLANCE REQUIREMENTS**

<b>SURVEILLANCE</b>		<b>FREQUENCY</b>
SR 3.4.17.1	Verify steam generator tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection.

5.7 Procedures, Programs, and Manuals (continued)

---

5.7.2.12 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 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 in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
  2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. For design basis accidents that have a faulted steam generator, accident induced leakage is not to exceed 1.0 gallon per minute (gpm) for the faulted steam generator and 150 gallons per day (gpd) for the non-faulted steam generators. For design basis accidents that do not have a faulted steam generator, accident induced leakage is not to exceed 150 gpd per steam generator.
  3. The operational leakage performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

5.7 Procedures, Programs, and Manuals

---

5.7.2.12 Steam Generator (SG) Program (continued)

- 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 SGs shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.
  
    - 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
  
  - e. Provisions for monitoring operational primary-to-secondary LEAKAGE.
- 

(continued)

5.7 Procedures, Programs, and Manuals (continued)

---

5.7.2.13 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.14 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2; ASME N510-1989, and the exceptions noted for each ESF system in Table 6.5 of the FSAR.

- a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Table 6.5 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Reactor Building Purge	< 1.00%	14,000 cfm $\pm$ 10%
Emergency Gas Treatment	< 0.05%	4,000 cfm $\pm$ 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm $\pm$ 10%
Control Room Emergency	< 1.00%	4,000 cfm $\pm$ 10%

(continued)

5.7 Procedures, Programs, and Manuals

---

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

- b. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass within acceptance criterion when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Table 6.5 of the FSAR, and ASME N510-1989 at the system flowrate specified below:

ESF VENTILATION SYSTEM	ACCEPTANCE CRITERIA	FLOW RATE
Reactor Building Purge	< 1.00%	14,000 cfm $\pm$ 10%
Emergency Gas Treatment	< 0.05%	4,000 cfm $\pm$ 10%
Auxiliary Building Gas Treatment	< 0.05%	9,000 cfm $\pm$ 10%
Control Room Emergency	< 1.00%	4,000 cfm $\pm$ 10%

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, and the exceptions noted for each ESF system in Table 6.5 of the FSAR, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of  $\leq 30^{\circ}\text{C}$  and greater than or equal to the relative humidity specified below.

---

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

ESF VENTILATION SYSTEM	METHYL IODIDE PENETRATION	RELATIVE HUMIDITY
Reactor Building Purge	< 10%	95%
Emergency Gas Treatment	< 0.175%	70%
Auxiliary Building Gas Treatment	< 0.175%	70%
Control Room Emergency	< 1.0%	70%

- d. Demonstrate for each of the ESF systems that the pressure drop across the entire filtration unit is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, the exceptions noted for each ESF system in Table 6.5 of the FSAR, and ASME N510-1989 at the system flowrate specified below.

ESF VENTILATION SYSTEM	PRESSURE DROP	FLOW RATE
Reactor Building Purge	< 4.7 inches water	14,000 cfm $\pm$ 10%
Emergency Gas Treatment	< 7.6 inches water	4,000 cfm $\pm$ 10%
Auxiliary Building Gas Treatment	< 7.6 inches water	9,000 cfm $\pm$ 10%
Control Room Emergency	< 3.5 inches water	4,000 cfm $\pm$ 10%

(continued)

5.7 Procedures, Programs, and Manuals

---

5.7.2.14 Ventilation Filter Testing Program (VFTP) (continued)

- e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF VENTILATION SYSTEM	AMOUNT OF HEAT
Emergency Gas Treatment	20 ± 2.0 kW
Auxiliary Building Gas Treatment	50 ± 5.0 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.7.2.15 Explosive Gas and Storage Tank Radioactivity Monitoring Program

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

The program shall include:

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

---

(continued)

5.7 Procedures, Programs, and Manuals

---

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

- b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of > 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and
- c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20.1302(b)(2)(i), at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.7.2.16 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to the 7 day storage tanks by determining that the fuel oil has:
  - 1. an API gravity or an absolute specific gravity within limits,
  - 2. a flash point and kinematics viscosity within limits for ASTM 2D fuel oil, and
  - 3. a clear and bright appearance with proper color;

---

(continued)

5.7 Procedures, Programs, and Manuals

---

5.7.2.16 Diesel Fuel Oil Testing Program (continued)

- b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to the 7 day storage tanks; and
- c. Total particulate concentration of the fuel oil in each of the four interconnected tanks which constitute a 7 day storage tank is  $\leq 10$  mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.7.2.17 (removed from Technical Specifications)

5.7.2.18 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.

---

(continued)

5.7 Procedures, Programs, and Manuals

---

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

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.

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.

5.7.2.19 Containment Leakage Rate Testing Program

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 (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The Fall 2007 end date for conducting the 10 year interval containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than Fall 2012.

The peak calculated containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is 15.0 psig.

The maximum allowable containment leakage rate,  $L_a$ , at  $P_a$ , is 0.25% of the primary containment air weight per day.

---

(continued)

5.7 Procedures, Programs, and Manuals

---

5.7.2.19 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

- a. Containment overall 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 combined Type B and Type C tests, and  $\leq 0.75 L_a$  for Type A tests.
- b. Air lock testing acceptance criteria are:
  - 1) Overall air lock leakage rate is  $\leq 0.05 L_a$  when tested at  $\geq P_a$ .
  - 2) For each door, leakage rate is  $\leq 0.01 L_a$  when pressurized to  $\geq 6$  psig.

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

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

---

5.0 ADMINISTRATIVE CONTROLS

5.8 Safety Function Determination Program (SFDP)

---

---

(moved to 5.7.2.18)

---

---

5.0 ADMINISTRATIVE CONTROLS

5.9 Reporting Requirements

---

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

5.9.1 DELETED

5.9.2 Annual Radiological Environmental Operating Report

-----NOTE-----

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

-----

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 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 the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. 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.

---

(continued)

5.9 Reporting Requirements

---

5.9.3 Radioactive Effluent Release Report

-----NOTE-----

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

-----

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.9.4 DELETED

---

(continued)

5.9 Reporting Requirements (continued)

---

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

- LCO 3.1.4 Moderator Temperature Coefficient
- LCO 3.1.6 Shutdown Bank Insertion Limit
- LCO 3.1.7 Control Bank Insertion Limits
- LCO 3.2.1 Heat Flux Hot Channel Factor
- LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
- LCO 3.2.3 Axial Flux Difference
- LCO 3.9.1 Boron Concentration

- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.

The approved analytical methods are specifically those described in the following documents:

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.
- 2a. WCAP-12945-P-A, Volume I (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).
- b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985. Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

---

(continued)

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)
  4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).
  5. WCAP-15088-P, Rev. 1, "Safety Evaluation Supporting A More Negative EOL Moderator Temperature Coefficient Technical Specification for the Watts Bar Nuclear Plant," July 1999, (W Proprietary), as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 20 (Methodology for Specification 3.1.4 - Moderator Temperature Coefficient.).
  6. Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM<sup>✓</sup>™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM<sup>✓</sup>™," Revision 0, May 2000; as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 31.
  7. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).
  8. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).
  9. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.9 Reporting Requirements (continued)

---

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

- a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
- LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits  
LCO 3.4.12 Cold Overpressure Mitigation System (COMS)
- b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC. The acceptability of the analytical methods is documented in NRC letter, "WATTS BAR UNIT 1 - ACCEPTANCE FOR REFERENCING OF PRESSURE TEMPERATURE LIMITS METHODOLOGY AND PRESSURE TEMPERATURE LIMITS REPORT (TAC M89048)", September 22, 1995 and "EXEMPTION FROM THE REQUIREMENTS OF 10 CFR Part 50.60, ACCEPTANCE CRITERIA FOR FRACTURE PREVENTION MEASURES FOR LIGHTWATER NUCLEAR POWER REACTORS FOR NORMAL OPERATION - WATTS BAR NUCLEAR PLANT (TAC NO. M99063)." September 29, 1997. Specifically, the analytical methods are described in the following references:
1. Letter, W. J. Museler to NRC, regarding request for exemption from 10 CFR 50.60, March 10, 1994.
  2. Letter, D. E. Nunn to NRC, regarding heatup and cooldown curves for normal operation (submitting WCAP-14176 and WCAP-14040, Rev. 1), December 23, 1994.
  3. Letter, R. R. Baron to NRC, responding to NRC July 11, 1995, request for additional information, July 31, 1995.
  4. Letter, R. R. Baron to NRC providing more information regarding cold overpressure mitigating system setpoints, September 8, 1995.
  5. Letter, J. A. Scalice to NRC, regarding request for exemption from 10 CFR 50.60, concerning use of Code Case N-514 to determine LTOP setpoints, dated June 20, 1997.
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

---

(continued)

5.9 Reporting Requirements (continued)

---

5.9.7 EDG Failures Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.9.8 PAMS Report

When a Report is required by Condition B or G 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.

5.9.9 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.7.2.12, Steam Generator (SG) Program. The report shall include:

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

5.0 ADMINISTRATIVE CONTROLS

5.10 Record Retention

---

---

(removed from Technical Specifications)

---

---

5.0 ADMINISTRATIVE CONTROLS

5.11 High Radiation Area

---

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.11.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

- a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.
- b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual entering such an area shall possess:
  1. A radiation monitoring device that continuously displays radiation dose rates in the area; or
  2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
  3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

---

(continued)

5.11 High Radiation Area

---

5.11.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
  - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
  - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.
- e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

---

(continued)

5.11 High Radiation Area (continued)

---

5.11.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or, continuously guarded door or gate that prevents unauthorized entry, and, in addition:
  1. All such door and gate keys shall be maintained under the administrative control of the Shift Manager, radiation protection manager, or his or her designee.
  2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.
- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual entering such an area shall possess:
  1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

---

(continued)

5.11 High Radiation Area

---

5.11.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
  3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
    - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
    - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.
  4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.
- e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individual's, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

---

(continued)

5.11 High Radiation Area

---

5.11.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

- f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.
-