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CHAPTER 5.0

ADMINISTRATIVE CONTROLS

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5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

- 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety and are not addressed in the Final Safety Analysis Report (FSAR) or Technical Specifications.

- 5.1.2 The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.
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5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

- a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the FSAR;
- b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;
- c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and
- d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

- a. An equipment operator shall be assigned when fuel is in the reactor and an additional equipment operator shall be assigned when the unit is in MODE 1, 2, 3, or 4.
- b. Shift crew composition may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

- c. A Radiation Protection Department technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.
- d. Not Used
- e. The operations manager or assistant operations manager shall hold an SRO license.
- f. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This position shall be assigned when the unit is in MODE 1, 2, 3, or 4, unless the Shift Manager or the Operating Supervisor meet the qualifications as required by the NRC.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

- 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978, with the following exceptions:
- 5.3.1.1 Shift Managers, Operating Supervisors, Reactor Operators, and Shift Technical Advisors shall meet or exceed the qualifications of ANSI/ANS 3.1-1981 as endorsed by Reg. Guide 1.8, Revision 2, with the same exceptions as contained in the current revision to the Operator Licensing Examiner Standards, NUREG-1021, ES-202.
- 5.3.1.2 The Radiation Protection Manager shall be a supervisor with line responsibility for operational health physics who meets or exceeds the qualifications of USNRC Regulatory Guide 1.8, September 1975, for a Radiation Protection Manager. The Radiation Protection Manager will be designated by the plant manager.
- 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).
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5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

- 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:
- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
 - b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;
 - c. Quality assurance for effluent and environmental monitoring;
 - d. Not Used; and
 - e. All programs specified in **Specification 5.5**.
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5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by [Specification 5.6.2](#) and [Specification 5.6.3](#).

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 change(s) together with the appropriate analyses or evaluations justifying the change(s), and
 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after the approval of the plant manager; and
- 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 in 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 (i.e., month and year) the change was implemented.

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5.5 Programs and Manuals (continued)

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the recirculation portion of the Containment Spray, Safety Injection, Chemical and Volume Control, and Residual Heat Removal. The program shall include the following:

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas, conforming to 10 CFR 50, Appendix I;
- e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;
- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;
- g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:
 - 1. For noble gases: A dose rate of ≤ 500 mrem/yr to the whole body and a dose rate of ≤ 3000 mrem/yr to the skin, and
 - 2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: A dose rate of ≤ 1500 mrem/yr to any organ.
- h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
- 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 to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

- 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 **FSAR, Section 3.9(N).1.1**, "Design Transients", 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 any tendon degradation, 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 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), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

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5.5 Programs and Manuals (continued)

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 applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

<u>ASME OM 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 and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program 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 OM 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:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 a 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, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 (3DP) 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. Leakage is not to exceed 1 gpm total for all four steam generators.
 3. The operational LEAKAGE performance criterion is specified in **LCO 3.4.13**, "RCS Operational LEAKAGE."

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

- c. Provisions for SG tube plugging 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 plugging 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. A degradation assessment 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 installation.

 - 2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

- (a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
- (b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
- (c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
- (d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

- 3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. 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.

5.5.11 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 Regulatory Guide 1.52, Rev. 2, and uses the test procedure guidance in Regulatory Guide 1.52, Revision 2, Positions C.5.a, C.5.c and C.5.d.

- 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 < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

- c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Filtration	2.0%	70%
Control Room Pressurization	2.0%	70%
Emergency Exhaust System	2.0%	70%

- d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration	5.4" WG	2000 cfm, ± 200 cfm
Control Room Pressurization	5.4" WG	500 cfm, +500,- 50 cfm
Emergency Exhaust System	5.4" WG	9000 cfm, ± 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI 510-1975 and corrected to design nameplate voltage settings.

ESF Ventilation System	Wattage
Control Room Pressurization	15 ± 2 KW
Emergency Exhaust System	37 ± 3 KW

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

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radwaste 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, Revision 0". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures, Revision 2".

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);
- 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 the outdoor liquid radwaste tanks listed below that are not

(continued)

5.5 Programs and Manuals

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

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 System is less than the quantities determined in accordance with the Standard Review Plan, Section 15.7.3:

- a. Reactor Makeup Water Storage Tank,
- b. Refueling Water Storage Tank,
- c. Condensate Storage Tank, and
- d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

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.

5.5.13 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 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 kinematic viscosity within limits for ASTM 2D fuel oil, and
 3. a water and sediment content within limits for ASTM 2D fuel oil.
- b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition of new fuel oil to storage tanks; and
- c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 31 days based on applicable ASTM D-2276 standards.
- d. The provisions of **SR 3.0.2** and **SR 3.0.3** are applicable to the Diesel Fuel Oil Testing Program test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
 1. a change in the TS incorporated in the license; or
 2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- 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;

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

- 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.

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.

(continued)

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

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.
 3. The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during the Refuel 14 outage (fall of 2005).
 4. The first Type A test performed after the October 26, 1999 Type A test shall be performed no later than October 25, 2014.
- b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 48.1 psig.
 - c. The maximum allowable containment leakage rate, L_a , at P_a , shall be 0.20% of the 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 C tests and $\leq 0.75 L_a$ for Type A tests;
 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 ≥ 10 psig.
 - e. The provisions of Technical Specification **SR 3.0.2** do not apply to the test frequencies in the Containment Leakage Rate Testing Program.
 - f. The provisions of Technical Specification **SR 3.0.3** are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals (continued)

5.5.17 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

- a. The definition of the CRE, CRE boundary, control building envelope (CBE), and the CBE Boundary.
- b. Requirements for maintaining the CRE and CBE boundaries in their design condition, including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air leakage past the CRE and CBE boundaries in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following exception is taken to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

1. The Tracer Gas Test based on the Brookhaven National Laboratory Atmospheric Tracer Depletion (ATD) Method is used to determine the unfiltered air leakage past the CRE and CBE boundaries. The ATD Method is described in AmerenUE letters dated December 15, 2004 (ULNRC-05104), June 6, 2006 (ULNRC-05298), July 16, 2007 (ULNRC-05427), and October 30, 2007 (ULNRC-05448).
- d. Measurement, at designated locations, of the CRE pressure relative to the outside atmosphere during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

(continued)

5.5 Programs and Manuals

5.5.17 Control Room Envelope Habitability Program (continued)

- e. The quantitative limits on unfiltered air leakage into CRE and CBE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air leakage measured by the testing described in paragraph c. The unfiltered air leakage limit for radiological challenges is the leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air leakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.
- f. The provisions of **SR 3.0.2** are applicable to the Frequencies for assessing CRE habitability, determining CRE and CBE unfiltered leakage, and measuring CRE pressure and assessing CRE and CBE as required by paragraphs c and d, respectively.

5.5.18 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
 - b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
 - c. The provisions of Surveillance Requirements **3.0.2** and **3.0.3** are applicable to the Frequencies established in the Surveillance Frequency Control Program.
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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Not Used.

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.

5.6.3 Radioactive Effluent Release Report

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 Part 50, Appendix I, Section IV.B.1.

5.6.4 Not used.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
1. Moderator Temperature Coefficient limits in [Specification 3.1.3](#),
 2. Shutdown Bank Insertion Limit for [Specification 3.1.5](#),
 3. Control Bank Insertion Limits for [Specification 3.1.6](#),
 4. Axial Flux Difference Limits for [Specification 3.2.3](#),
 5. Heat Flux Hot Channel Factor, $F_Q(Z)$, F_Q^{RTP} , $K(Z)$, $W(Z)$ and F_Q Penalty Factors for [Specification 3.2.1](#),
 6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}^N$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for [Specification 3.2.2](#),
 7. Shutdown Margin Limits for [Specifications 3.1.1](#), [3.1.4](#), [3.1.5](#), [3.1.6](#), and [3.1.8](#),
 8. Reactor Core Safety Limits Figure for [Specification 2.1.1](#),
 9. Overtemperature ΔT and Overpower ΔT Setpoint Parameters for [Specification 3.3.1](#), and
 10. Reactor Coolant System Pressure and Temperature DNB Limits for [Specification 3.4.1](#).
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY."
 2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION."
 3. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE."

(continued)

5.6 Reporting Requirements

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."
 5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."
 6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."
 7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."
 8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."
 9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER DT AND THERMAL OVERTEMPERATURE DT TRIP FUNCTIONS."
 10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."
 11. WCAP-10965-P-A Addendum 2-A, "Qualification of the New Pin Power Recovery Methodolgy."
 12. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."
 13. WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications."
 14. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON."
 15. WCAP-16045-P-A Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."
- 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.

(continued)

5.6 Reporting Requirements

- 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. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
 - 1. **Specification 3.4.3**, "RCS Pressure and Temperature (P/T) Limits," and
 - 2. **Specification 3.4.12**, "Cold Overpressure Mitigation System (COMS)."
- b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves".
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

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 Not used.

(continued)

5.6 Reporting Requirements (continued)

5.6.10 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 5.5.9**, "Steam Generator (SG) Program." The report shall include:

- a. The scope of inspections performed on each SG;
 - b. 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 degradation mechanism;
 - f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator; and
 - g. The results of condition monitoring, including the results of tube pulls and in-situ testing.
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5.0 ADMINISTRATIVE CONTROLS

5.7 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:

- 5.7.1 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 or group 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 rate information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or
 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(continued)

5.7 High Radiation Area

5.7.1 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)

- (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, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 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/Operating Supervisor or Radiation Protection Department Supervision, 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.

(Continued)

5.7 High Radiation Area

5.7.2 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)

- 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 or group 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
 - 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, or
 - 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably

(Continued)

5.7 High Radiation Area

5.7.2 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)

Achievable” principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

- e. Except for individual 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.
 - f. Such individual areas that are within a larger area, such as PWR containment, 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.
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