1.0 USE AND APPLICATION

1.1 Definitions

- NOTE -

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATIONS	CORE ALTERATIONS shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP 30, Supplement to Part 1, pages 192-212, table entitled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

LEAKAGE

LEAKAGE from the RCS shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

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

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MODE
- MODES

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:

a. Described in Chapter 14, Initial Test Program of the UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission (NRC).

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with the Low Temperature Overpressurization Protection System for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these limits is addressed in individual specifications.

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1520 MWt.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Shutdown	< 0.99	NA	≥ 350
4	Hot Standby[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

3.3 INSTRUMENTATION

3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

LCO 3.3.6 The CREATS actuation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each Function.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more Functions with one channel or train inoperable.	A.1	Place one CREATS train in emergency mode.	7 days
B.	One or more Functions with two channels or two trains inoperable.	B.1.1	Place one CREATS train in emergency mode.	Immediately
		AND		
		B.1.2	Enter applicable Conditions and Required Actions for one CREATS train made inoperable by inoperable CREATS actuation instrumentation.	Immediately
		OR		
		B.2	Place both CREATS trains in emergency mode.	Immediately
C.	Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
		AND		
		C.2	Be in MODE 5.	36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	Perform COT.	92 days
SR 3.3.6.3	- NOTE - Verification of setpoint is not required. Perform TADOT.	24 months
SR 3.3.6.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.5	Perform ACTUATION LOGIC TEST.	24 months

Table 3.3.6-1
CREATS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	LIMITING SAFETY SYSTEM SETTINGS[a]
1.	Manual Initiation	1, 2, 3, 4, (b)	2 trains	SR 3.3.6.3	NA
2.	Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, (b)	2 trains	SR 3.3.6.5	NA
3.	Control Room Radiation Intake Monitors	1, 2, 3, 4, (b)	2	SR 3.3.6.1 SR 3.3.6.2 SR 3.3.6.4	≤ .57 mR/hr
4.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) A channel is OPERABLE when both of the following conditions are met:

1. The absolute difference between the as-found Trip Setpoint (TSP) and the previous as-left TSP is within the COT Acceptance Criteria. The COT Acceptance Criteria is defined as:

   |as-found TSP - previous as-left TSP| ≤ COT uncertainty

   The COT uncertainty shall not include the calibration tolerance.

2. The as-left TSP is within the established calibration tolerance band about the nominal TSP. The nominal TSP is the desired setting and shall not exceed the Limiting Safety System Setting (LSSS). The LSSS, COT uncertainty, and the established calibration tolerance band are defined in accordance with the Ginna instrument setpoint methodology. The channel is considered operable even if the as-left TSP is non-conservative with respect to the LSSS provided that the as-left TSP is within the established calibration tolerance band.

(b) During movement of irradiated fuel assemblies

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	DOSE EQUIVALENT I-131 specific activity not within limit.		- NOTE - LCO 3.0.4 is not applicable.	
		A.1	Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm.	Once per 8 hours
		AND		
		A.2	Restore DOSE EQUIVALENT I-131 to within limit.	7 days
B.	Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 specific activity > 60 µCi/gm.	B.1	Be in MODE 3 with T_{avg} < 500°F.	8 hours
C.	Gross specific activity not within limit.	C.1	Be in MODE 3 with T_{avg} < 500°F.	8 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/$\bar{E}$ µCi/gm.	7 days
SR 3.4.16.2	- NOTE - Only required to be performed in MODE 1.	
	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm.	14 days AND Between 2 and 10 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.16.3	- NOTE - Only required to be performed in MODE 1.	
	Determine $\bar{E}$ from a reactor coolant sample.	Once within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. AND Every 184 days thereafter

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC), and NaOH Systems

LCO 3.6.6 Two CS trains, four CRFC units, and the NaOH system shall be OPERABLE.

- NOTE -

In MODE 4, both CS pumps may be in pull-stop for up to 2 hours for the performance of interlock and valve testing of motor operated valves (MOVs) 857A, 857B, and 857C. Power may also be restored to MOVs 896A and 896B, and the valves placed in the closed position, for up to 2 hours for the purpose of each test.

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

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One CS train inoperable.	A.1	Restore CS train to OPERABLE status.	72 hours
B.	NaOH system inoperable.	B.1	Restore NaOH System to OPERABLE status.	72 hours
C.	Required Action and associated Completion Time of Condition A or B not met.	C.1 AND	Be in MODE 3.	6 hours
		C.2	Be in MODE 5.	84 hours
D.	One or two CRFC units inoperable.	D.1	Restore CRFC unit(s) to OPERABLE status.	7 days
E.	Required Action and associated Completion Time of Condition D not met.	E.1 AND	Be in MODE 3.	6 hours
		E.2	Be in MODE 5.	36 hours

CONDITION		REQUIRED ACTION		COMPLETION TIME
F.	Two CS trains inoperable. OR Three or more CRFC units inoperable.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Perform SR 3.5.2.1 and SR 3.5.2.3 for valves 896A and 896B.	In accordance with applicable SRs.
SR 3.6.6.2	Verify each CS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.3	Verify each NaOH System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.4	Operate each CRFC unit for ≥ 15 minutes.	31 days
SR 3.6.6.5	Verify cooling water flow through each CRFC unit.	31 days
SR 3.6.6.6	Verify each CS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.7	Verify NaOH System solution volume is ≥ 3000 gal.	184 days
SR 3.6.6.8	Verify NaOH System tank NaOH solution concentration is ≥ 30% and ≤ 35% by weight.	184 days
SR 3.6.6.9	Perform required CRFC unit testing in accordance with the VFTP.	In accordance with the VFTP
SR 3.6.6.10	Verify each automatic CS valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months

	SURVEILLANCE	FREQUENCY
SR 3.6.6.11	Verify each CS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.12	Verify each CRFC unit starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.13	Verify each automatic NaOH System valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.14	Verify spray additive flow through each eductor path.	5 years
SR 3.6.6.15	Verify each spray nozzle is unobstructed.	10 years

3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Air Treatment System (CREATS)

LCO 3.7.9 Two CREATS Trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One CREATS train inoperable.	A.1	Restore CREATS train to OPERABLE status.	7 days
B.	Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 AND	Be in MODE 3.	6 hours
		B.2	Be in MODE 5.	36 hours
C.	Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	C.1	Suspend movement of irradiated fuel assemblies.	Immediately
D.	Two CREATS trains inoperable in MODE 1, 2, 3, or 4.	D.1	Enter LCO 3.0.3.	Immediately
E.	Two CREATS trains inoperable during movement of irradiated fuel assemblies.	E.1	Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.9.1	Operate each CREATS filtration train ≥ 15 minutes.	31 days
SR 3.7.9.2	Perform required CREATS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.9.3	Verify each CREATS train actuates on an actual or simulated actuation signal.	24 months

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

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

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain:

a. 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 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.

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),

   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 does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by the onsite review function and 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.

5.5.2 Primary Coolant Sources Outside Containment Program

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. The systems include Containment Spray, Safety Injection, and Residual Heat Removal in the recirculation configuration. 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 Deleted

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 ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

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 from the plant to unrestricted areas, conforming to 10 CFR 50, Appendix I and 40 CFR 141;

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 to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the plant 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 from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit Program

This program provides controls to track the reactor coolant system cyclic and transient occurrences specified in UFSAR Table 5.1-4 to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 2.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Inservice Testing Program

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

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

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

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

5.5.8 Steam Generator (SG) Tube Surveillance Program

Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program in accordance with the Nuclear Policy Manual. This inspection program shall define the specific requirements of the edition and Addenda of the ASME Boiler and Pressure Code, Section XI, as required by 10 CFR 50.55a(g). The program shall include the following:

a. The inspection intervals for SG tubes shall be specified in the Inservice Inspection Program.

b. SG tubes that have imperfections > 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.

c. SG sleeves that have imperfections > 30% through wall, as indicated by eddy current, shall be repaired by plugging.

5.5.9 Secondary Water Chemistry Program

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

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.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented. The test methods will be in accordance with Regulatory Guide 1.52, Revision 2, except as modified below.

a. Containment Recirculation Fan Cooler System

1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

b. Control Room Emergency Air Treatment System (CREATS)

1. Demonstrate the pressure drop across the combined HEPA filters, the prefilters, the charcoal adsorbers and the post-filters is < 11 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place DOP test of the HEPA filter bank shows a penetration and system bypass < 0.05%.

3. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 0.05%, when tested under ambient conditions.

4. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 1.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F), a relative humidity of 95%, and a face velocity of 61 ft/min.

c. SFP Charcoal Adsorber System

1. Demonstrate that the total air flow rate from the charcoal adsorbers shows at least 75% of that measured with a complete set of new adsorbers.

2. Demonstrate that an in-place Freon test of the charcoal adsorbers bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

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

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in waste gas decay tanks. The gaseous radioactivity quantities shall be determined following the methodology in NU REG-0133.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas decay tanks 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); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay 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.

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.

5.5.12 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 clear and bright appearance with proper color; and

b. Within 31 days following addition of the new fuel to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil.

5.5.13 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 UFSAR 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 UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13.b.1 or Specification 5.5.13.b.2 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 50.71e.

5.5.14 Safety Function Determination Program (SFDP)

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

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

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

a. A required system redundant to the supported system(s) 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 inoperable 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.

5.5.15 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 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq P_a$.

c. Mini-purge valve acceptance criteria is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.