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CHAPTER 16.0

TECHNICAL REQUIREMENTS

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CHAPTER 16.0

TECHNICAL REQUIREMENTS

See the Callaway Plant Technical Specifications (NUREG-1058), Appendix A to NRC License No. NPF-30, for the retained Specifications. Requirements contained in this chapter were relocated in accordance with the NRC Final Policy Statement on Technical Specification Improvements, 58FR39132 dated July 22, 1993.

16.0 TECHNICAL REQUIREMENTS

16.0.1 GENERAL OPERATIONAL REQUIREMENTS

Within the context of the following Requirements, equipment covered by these requirements do not perform functions meeting the four criteria of 10CFR50.36(c)(2)(ii). Exceptions to the following requirements shall be stated in the individual Requirements.

16.0.1.1 The Limiting Conditions for Operation (LCOs) contained in the following Requirements shall be met for the MODES and operational conditions specified under the Applicability section except as noted under Sections 16.0.1.2, 16.0.1.3, and 16.0.1.6.

16.0.1.2 Upon making the determination that the LCO cannot be met, the guidance contained in the ACTION section of the Requirement shall be implemented, excepted as provided in Section 16.0.1.5. Completion of the prescribed actions is not required if compliance with the LCO is restored within the time limits allowed in the ACTION section or if the LCO is no longer applicable, unless otherwise stated.

16.0.1.3 When an LCO is not met, except as provided in the associated ACTION guidance, ACTION shall be implemented as determined by the Senior Director, Nuclear Operations or his designee. This requirement is not applicable in MODE 5 or 6. Exceptions to this requirement are stated in the individual Requirements.

16.0.1.4 Entry into a higher OPERATIONAL MODE or condition covered by these Requirements shall not be made if the LCO and associated ACTIONS cannot be met except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. The Senior Director, Nuclear Operations or his designee may allow entry into an OPERATIONAL MODE or condition while relying on the guidance of the ACTION section to meet the LCO. Exceptions are stated in the individual Requirements. This shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION guidance or that are part of a shutdown of the unit. This requirement is only applicable for entry into a MODE or other specified condition in the Applicability for MODES 1, 2, 3, and 4.

16.0.1.5 Equipment removed from service or declared non-functional to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its FUNCTIONALITY or the FUNCTIONALITY of other equipment. This is an exception to Section 16.0.1.2 for the system returned to service under administrative control to perform the testing required to demonstrate FUNCTIONALITY.

16.0.1.6 Test Exception 16.10.1 allows Section 16.1.3.1 requirements to be suspended to permit performance of special tests and operations. Compliance with the 16.10.1 LCO is optional. When the 16.10.1 LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When the 16.10.1 LCO is not desired

to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with Section 16.1.3.1.

16.0.2 GENERAL SURVEILLANCE REQUIREMENTS

These Surveillance Requirements apply to equipment that, within the context of the following Requirements, do not perform functions meeting the four criteria of 10CFR50.36(c)(2)(ii). Exceptions to these requirements shall be stated in the individual sections.

16.0.2.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for the individual LCOs unless stated otherwise in the individual Surveillance Requirement.

16.0.2.2 Each Surveillance Requirement shall be performed within the specified surveillance interval plus 25% unless otherwise approved by the Senior Director, Nuclear Operations or his designee. If a Surveillance requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance. If an ACTION requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Requirement are stated in the individual Requirements.

16.0.2.3 Failure to perform a Surveillance Requirement within the allowed time interval defined in Section 16.0.2.2, and without prior approval by the Senior Director, Nuclear Operations or his designee, constitutes noncompliance with the LCO. However, application of the ACTION guidance may be delayed as deemed appropriate by the Senior Director, Nuclear Operations or his designee to allow completion of the Surveillance Requirements. Surveillance Requirements do not have to be performed on non-functional equipment or variables outside specified limits.

16.0.2.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the LCO have been performed within the time limits allowed by Section 16.0.2.2 or unless approved in accordance with Section 16.0.1.4. This shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION guidance or that are part of a shutdown of the unit. Section 16.0.2.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4..

16.0.2.5 Deleted

16.0.3 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout Chapter 16.

ACTION

ACTION shall be that part of a Chapter 16 Technical Requirement which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE (AFD)

AFD shall be the difference in normalized flux signals between the top and bottom halves of an excore neutron detector.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameters that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel FUNCTIONALITY. 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, or total channel steps.

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 FUNCTIONALITY of all devices in the channel required for channel FUNCTIONALITY. The COT may be performed by means of any series of sequential, overlapping, or total channel steps. The COT shall include adjustments, as necessary, of the required alarm, interlock and Trip Setpoints required for channel FUNCTIONALITY such that the Setpoints are within the necessary range and accuracy.

CORE ALTERATION

CORE ALTERATION 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 unit 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 Technical Specification 5.6.5. Plant operation within these limits is addressed in individual Requirements.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotope I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from:

1) Table III of TID-14844, AEC, 1962, “Calculation of Distance Factors for Power and Test Reactor Sites, ‘ or

2) Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or

3) International Commission on Radiological Protection (ICRP) Publication 30, “Limits for Intakes of Radionuclides by Workers,” Supplement to Part 1, pages 192-212, Table titled “Committed Dose Equivalent in Target Organs or Tissues per intake of Unit Activity,” 1979, or

4) Table 2.1 of EPA Federal Guidance Report No. 11, EPA-520/1-88-020, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1988.

ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components in Table16.3-2 provided that the

components and methodology for verification have been previously reviewed and approved by NRC.

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 16.0-1.

FUNCTIONAL - FUNCTIONALITY

The term FUNCTIONAL/FUNCTIONALITY shall be used for SSCs not described in Technical Specifications, but which warrant programmatic controls to ensure that SSC availability and reliability are maintained. In general, these SSCs and the related controls are included in programs related to 10 CFR 50 Appendix B and 10 CFR 50.65 (i.e. the “maintenance rule”). Additionally, SSCs not described in Technical Specifications may fall within the scope of the term FUNCTIONAL because they perform functions described in the Final Safety Analysis Report (FSAR), emergency plan, fire protection plan, regulatory commitments, or other elements of the Current Licensing Basis (CLB).

A system, subsystem, train, component, or device shall be FUNCTIONAL or have FUNCTIONALITYwhen it is capable of performing its function(s) as set forth in the CLB, and when all necessary attendent instrumentation, controls, normal and 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 function(s). These CLB function(s) may include the capability to perform a necessary and related support function for an SSC(s) controlled by TSs.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls Program required by Technical Specification 5.5.4, (2) the Radiological Environmental Monitoring Program required by Section 16.11.4, and (3) descriptions of the information that should be

included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Technical Specifications 5.6.2 and 5.6.3.

OPERABLE - OPERABILITY

The term OPERABLE/OPERABILITY, as used in this Chapter, refers to an SSC's ability to perform its specified safety function as required by the Callaway Technical Specifications. The following definition of OPERABLE-OPERABILITY is taken directly from the Technical Specifications and repeated here for reference:

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).

OPERATIONAL MODE - MODE

An OPERATIONAL MODE (i.e., 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 16.0-2 with fuel in the reactor vessel.

PROCESS CONTROL PROGRAM

The PROCESS CONTROL PROGRAM shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3565 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components in Table 16.3-1 provided that the components and methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster control assemblies are fully inserted except for the single rod cluster control assembly of highest reactivity worth which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCS must be accounted for in the determination of SDM. In MODES 1 and 2, the fuel and moderator temperatures are changed to hot zero power temperatures.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

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.

UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features (ESF) Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

VENTING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

WASTE GAS HOLDUP SYSTEM

A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

TABLE 16.0-1 FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

TABLE 16.0-2 OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER[a]	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	> 5%	NA
2. STARTUP	≥ 0.99	≤ 5%	NA
3. HOT STANDBY	< 0.99	NA	≥ 350°F
4. HOT SHUTDOWN[b]	< 0.99	NA	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN[b]	< 0.99	NA	≤ 200°F
6. REFUELING[c]	NA	NA	NA

(a) Excluding decay heat.

(b) At least 53 of 54 reactor vessel head closure bolts fully tensioned.

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

16.1 REACTIVITY CONTROL SYSTEMS

16.1.1 INTENTIONALLY BLANK

16.1.2 BORATION SYSTEMS

16.1.2.1 FLOW PATH - SHUTDOWN LIMITING CONDITION FOR OPERATION

As a minimum, one of the following boron injection flow paths shall be FUNCTIONAL and capable of being powered from a FUNCTIONAL emergency power source:

a. For MODES 4, 5 and 6 with the reactor vessel head installed:

   1) A flow path from the Boric Acid Storage System via a boric acid transfer pump and an ECCS centrifugal charging pump to the Reactor Coolant System; or

   2) A flow path from the Refueling Water Storage Tank via an ECCS centrifugal charging pump to the Reactor Coolant System; or

b. For MODE 6 with the reactor vessel head removed and water level less than 23-feet above the reactor vessel flange:

1) Either flow path per Part a above; or

2) A flow path from the refueling water storage tank via a safety injection pump to the Reactor Coolant System if the refueling water storage tank is FUNCTIONAL as specified in Section 16.1.2.5b for Mode 6 (when the vessel head is off there is no restriction on SI pump operability/ inoperability); or

c. For MODE 6 with the reactor vessel head removed and water level greater than 23-feet above the reactor vessel flange:

   1) Any flow path per Part a or Part b above; or

   2) A flow path from the refueling water storage tank via the residual heat removal pump not credited for decay heat removal per Technical Specification 3.9.5 to the Reactor Coolant System or via the safety injection pump to the Reactor Coolant System if the refueling water storage tank is FUNCTIONAL as specified in Section 16.1.2.5b for Mode 6.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With none of the above flow paths FUNCTIONAL or capable of being powered from a FUNCTIONAL emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity additions that could result in loss of required SDM or boron concentration.

16.1.2.1.1 SURVEILLANCE REQUIREMENTS

At least one of the above required flow paths shall be demonstrated FUNCTIONAL at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is or can be readily placed in its correct position for the required flow path.

16.1.2.1.2 BASES

The Boration Systems ensure that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) ECCS centrifugal charging pumps (Modes 4-6), RHR pumps (Mode 6 with water level greater than 23-feet above the reactor vessel flange and with reactor vessel head removed), or SI pumps (Mode 6 with reactor vessel head removed), (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from FUNCTIONAL diesel generators (for Modes 4, 5, and 6 only).

With the RCS average temperature equal to or greater than 350°F, a minimum of two boron injection flow paths are required to ensure functional capability in the event an assumed single failure renders one of the flow paths non-functional. The Boration capability of either flow path is sufficient to provide the SHUTDOWN MARGIN specified in the COLR. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires either 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,745 gallons of 2350 ppm borated water from the RWST. With the RCS average temperature less than 350°F, only one boron injection flow path is required.

With the RCS temperature below 200°F, one Boration System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor. Additional restrictions apply in MODES 4, 5, and 6 prohibiting CORE ALTERATIONS and positive reactivity additions that could result in loss of required SDM (MODES 4 and 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must be evaluated to ensure they do not result in a loss of required SDM.

The boron capability required below 200°F is sufficient to provide the SHUTDOWN MARGIN specified in the COLR. This condition requires either 2968 gallons of 7000 ppm borated water from the boric acid storage tanks or 14,076 gallons of 2350 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a minimum equilibrium sump pH of 7.1 for the solution recirculated within Containment after a LOCA. This pH level minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

Either of the Centrifugal Charging Pumps (CCPs), PBG05A or PBG05B, may still be considered FUNCTIONAL with its associated alternate Reactor Coolant Pump Seal injection throttle valve, BGHV8357A or BGHV8357B non-functional. Each CCP is protected from degradation by its recirculation line. A non-functional alternate Reactor Coolant Pump seal injection throttle valve does render the associated alternate Reactor Coolant Pump seal injection and/or boron injection flowpath non-functional (RFR 16214B).

For Modes 4, 5, and 6, a FUNCTIONAL emergency diesel generator (D/G) must be aligned to provide emergency power to the FUNCTIONAL CCP and FUNCTIONAL Boric Acid Transfer Pump (BATP).

The flow paths required to satisfy the requirements of Sections 16.1.2.1 and 16.1.2.2 are in accordance with appropriate engineering evaluations.

If both Refueling Water Storage Tank (RWST) boration flow paths are non-functional and 'B' D/G is non-functional (rendering the immediate boration flow path through valve BGHV8104 non-functional) then the alternate immediate boration flow path through valve BGV0177 may be used provided: 1) Valve BGV0177 is locked open and 2) reactor makeup water is isolated by locking closed and tagging BGV0178 and BGV0601. Valve BGV0177 should only be used in this manner if all other flowpaths are non-functional.

In order for a boration flow path to be valid, surveillance testing may be required in accordance with the Inservice Testing (IST) Program.

The FUNCTIONALITY of one Boration System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. The borated water source should be a highly concentrated solution, such as that normally found in the boric acid storage tanks or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

The surveillance in Section 16.1.2.1.1 requires verification that each valve in the flow path is or can be readily placed in its correct position for the required flow path. This requires that a dedicated individual is available and authorized to manipulate valve position(s). It is not expected that an open flow path be maintained between the boron source and reactor coolant system (RCS) for the credited flow path, as this would likely create an operational challenge. It is required that a flow path be identified, with all of the critical valves identified, such that valves required to be open are, in fact, open or are capable of being opened by plant operators either remote-manually from the control room or locally at the valve(s).

The 18-month frequency for the surveillance in Section 16.1.2.2.1.b is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for unplanned plant transients if the surveillance were performed with the reactor at power.

16.1.2.1.3 REFERENCES

1. RFR 015703A
2. RFR 015703B
3. RFR 015703C
4. RFR 019286A
5. RFR 019286B
6. RFR 019289A
7. RFR 019356A
8. RFR 018471A
9. RFR 018471B
10. RFR 016214A
11. RFR 201109140

16.1.2.2 FLOW PATHS - OPERATING LIMITING CONDITION FOR OPERATION

At least two of the following three boron injection flow paths shall be FUNCTIONAL:

a. The flow path from the Boric Acid Storage System via a boric acid transfer pump and an ECCS centrifugal charging pump to the Reactor Coolant System; and

b. Two flow paths from the refueling water storage tank via ECCS centrifugal charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3. (The provisions of Sections 16.0.1.4 and 16.0.2.4 are not applicable for entry into MODE 3 for the ECCS centrifugal charging pump rendered incapable of injection pursuant to Technical Specification 3.4.12 provided that the ECCS centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of all RCS cold legs exceeding 375°F, whichever comes first.)

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System FUNCTIONAL, restore at least two boron injection flow paths to the Reactor Coolant System to FUNCTIONAL status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the COLR for MODE 5 within the next 6 hours; restore at least two flow paths to FUNCTIONAL status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

16.1.2.2.1 SURVEILLANCE REQUIREMENTS

At least two of the above required flow paths shall be demonstrated FUNCTIONAL:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 18 months by verifying that each automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to its correct position on an actual or simulated Safety Injection signal; and

c. At least once per 18 months by verifying that the flow path required by Section 16.1.2.2a delivers at least 30 gpm to the Reactor Coolant System.

16.1.2.2.2 BASES

See Section 16.1.2.1.2.

16.1.2.2.3 REFERENCES

SEE Section 16.1.2.1.3.

16.1.2.3 ECCS PUMPS - SHUTDOWN LIMITING CONDITION FOR OPERATION

One safety injection pump, ECCS centrifugal charging pump, or residual heat removal pump in the boron injection flow path required by Section 16.1.2.1 shall be FUNCTIONAL and capable of being powered from a FUNCTIONAL emergency power source.

APPLICABILITY:

MODES 4 and 5 (for ECCS centrifugal charging pump), and MODE 6 (for safety injection, ECCS centrifugal charging, or residual heat removal pump as defined in Section 16.1.2.1).

ACTION:

With no ECCS centrifugal charging pump, safety injection pump, or residual heat removal pump in the boron injection flow path FUNCTIONAL or capable of being powered from a FUNCTIONAL emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity additions that could result in loss of required SDM or boron concentration.

16.1.2.3.1 SURVEILLANCE REQUIREMENTS

The above required safety injection, ECCS centrifugal charging, or residual heat removal pump in the boron injection flow path shall be demonstrated FUNCTIONAL by verifying each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head in accordance with the Inservice Testing Program.

The provisions of Section 16.0.2.2 are not applicable.

16.1.2.3.2 BASES

See Section 16.1.2.1.2.

The provisions of Section 16.0.2.2 do not apply, as the Inservice Testing Program includes its own provisions for extending test frequencies.

16.1.2.4 CHARGING PUMPS - OPERATING LIMITING CONDITION FOR OPERATION

At least two ECCS centrifugal charging pumps shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, and 3. (The provisions of Sections 16.0.1.4 and 16.0.2.4 are not applicable for entry into MODE 3 for the ECCS centrifugal charging pump rendered incapable of injection pursuant to Technical Specification 3.4.12 provided that the ECCS centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of all RCS cold legs exceeding 375°F, whichever comes first.)

ACTION:

With only one ECCS centrifugal charging pump FUNCTIONAL, restore at least two ECCS centrifugal charging pumps to FUNCTIONAL status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the COLR for MODE 5 within the next 6 hours; restore at least two ECCS centrifugal charging pumps to FUNCTIONAL status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

16.1.2.4.1 SURVEILLANCE REQUIREMENTS

At least two ECCS centrifugal charging pumps shall be demonstrated FUNCTIONAL by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to the Inservice Testing Program.

The provisions of Section 16.0.2.2 are not applicable.

16.1.2.4.2 BASES

See Section 16.1.2.1.2.

The provisions of Section 16.0.2.2 do not apply, as the Inservice Testing Program includes its own provisions for extending test frequencies.

16.1.2.5 BORATED WATER SOURCE - SHUTDOWN LIMITING CONDITION FOR OPERATION

As a minimum, one of the following borated water sources, if required by Section 16.1.2.1 for MODES 5 and 6, shall be FUNCTIONAL:

a. A Boric Acid Storage System with:

   1) A minimum contained borated water volume of 2968 gallons,

   2) Between 7000 and 7700 ppm of boron, and

   3) A minimum solution temperature of 65°F; or

b. The refueling water storage tank (RWST) with:

   1) A minimum contained borated water volume of 55,420 gallons,

   2) A minimum boron concentration of 2350 ppm, and

   3) A minimum solution temperature of 37°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source FUNCTIONAL, suspend all operations involving CORE ALTERATIONS or positive reactivity additions that could result in loss of required SDM or boron concentration.

16.1.2.5.1 SURVEILLANCE REQUIREMENTS

The above required borated water source shall be demonstrated FUNCTIONAL:

a. At least once per 7 days by:

   1) Verifying the boron concentration of the water,

   2) Verifying the contained borated water volume, and

   3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 37°F.

16.1.2.5.2 BASES

See Section 16.1.2.1.2.

16.1.2.6 BORATED WATER SOURCES - OPERATING LIMITING CONDITION FOR OPERATION

The Boric Acid Storage System shall be FUNCTIONAL if required by Section 16.1.2.2 for MODES 1, 2, and 3 and shall be FUNCTIONAL if required by Section 16.1.2.1 for MODE 4 with:

1) A minimum contained borated water volume of 17,658 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

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

ACTION:

With the Boric Acid Storage System non-functional and being used as one of the required borated water sources, restore the storage system to FUNCTIONAL status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as specified in the COLR for MODE 5 and be in COLD SHUTDOWN within the next 198 hours.

16.1.2.6.1 SURVEILLANCE REQUIREMENTS

The Boric Acid Storage System shall be demonstrated FUNCTIONAL at least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

16.1.2.6.2 BASES

See Section 16.1.2.1.2.

16.1.3 MOVABLE CONTROL ASSEMBLIES

16.1.3.1 POSITION INDICATION SYSTEM-SHUTDOWN LIMITING CONDITION FOR OPERATION

One digital rod position indicator (excluding demand position indication) shall be FUNCTIONAL and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3, 4, and 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted. (See Special Test Exception in Section 16.10.1.)

ACTION:

With less than the above required position indicator(s) FUNCTIONAL, immediately insert all rods fully and place the Rod Control System in a condition incapable of rod withdrawal.

16.1.3.1.1 SURVEILLANCE REQUIREMENTS

Each of the above required digital rod position indicator(s) shall be determined to be FUNCTIONAL by verifying that the digital rod position indicator agrees with the demand position indicator within 12 steps when exercised over the full-range of rod travel prior to criticality after each removal of the reactor vessel head.

16.1.3.1.2 BASES

See Technical Specification Bases 3.1.7.

16.1.3.2 INTENTIONALLY BLANK

16.1.3.3 ROD POSITION DEVIATION MONITOR LIMITING CONDITION FOR OPERATION

The Rod Position Deviation Monitor shall be FUNCTIONAL.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the Rod Position Deviation Monitor non-functional, verify individual rod positions to be within the alignment limits and determine each digital rod position indicator to be FUNCTIONAL by verifying that the Demand Position Indication System and the Digital Position Indication System agree within 12 steps at least once per 4 hours.

16.1.3.3.1 SURVEILLANCE REQUIREMENTS

Not applicable.

16.1.3.3.2 BASES

Control rod alignment and OPERABILITY of the digital rod position indicators are required to be verified on a nominal basis per Technical Specification SR 3.1.4.1, with more frequent verifications required by this Technical Requirement if the automatic Rod Position Deviation Monitor is non-functional. The more frequent verifications are sufficient to ensure control rod alignment to be within assumed limits.

16.1.3.4 ROD INSERTION LIMIT MONITOR LIMITING CONDITION FOR OPERATION

The Rod Insertion Limit Monitor shall be FUNCTIONAL.

APPLICABILITY: MODES 1 AND 2 WITH k_{eff} greater than or equal to 1.

ACTION:

With the Rod Insertion Limit Monitor non-functional, verify individual control rod positions to be within the insertion limits specified in the COLR at least once per 4 hours.

16.1.3.4.1 SURVEILLANCE REQUIREMENTS

Not applicable.

16.1.3.4.2 BASES

Control bank insertion is required to be verified within limits on a nominal basis per Technical Specification SR 3.1.6.2, with more frequent verifications required for each control rod if the automatic Rod Insertion Limit Monitor is non-functional. The more frequent verifications are sufficient to ensure control rod insertion to be within assumed limits.

16.2 POWER DISTRIBUTION LIMITS

16.2.1 AXIAL FLUX DIFFERENCE MONITOR ALARM

16.2.1.1 LIMITING CONDITION FOR OPERATION

The AXIAL FLUX DIFFERENCE (AFD) Monitor Alarm shall be FUNCTIONAL.

APPLICABIITY: MODE 1 with THERMAL POWER > 50% RTP

ACTION:

With the AFD Monitor Alarm non-functional monitor and log the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter.

16.2.1.1.1 SURVEILLANCE REQUIREMENTS

Not Applicable.

16.2.1.1.2 BASES

AFD is required to be verified within limits on a nominal basis per Technical Specification SR 3.2.3.1, with more frequent verifications required if the AFD Monitor Alarm is non-functional. The more frequent verifications are sufficient to ensure AFD to be within assumed limits.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the one-minute average AFD for at least two OPERABLE excore detector channels are outside the allowed delta-I vs. power operating space. When the alarm is non-functional, the AFD is monitored every hour to detect operation outside its limit. The completion time of once per hour is based on operating experience regarding the amount of time required to vary the AFD and the fact that AFD is closely monitored.

The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

16.2 POWER DISTRIBUTION LIMITS

16.2.2 QUADRANT POWER TILT RATIO ALARM

16.2.2.1 LIMITING CONDITION FOR OPERATION

The QUADRANT POWER TILT RATIO (QPTR) Alarm shall be FUNCTIONAL.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP

NOTE: With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR.

ACTION:

With the QPTR Alarm non-functional, verify QPTR is within limit by calculation or by using core power distribution measurement information once per 12 hours during steady-state operation.

16.2.2.1.1 SURVEILLANCE REQUIREMENTS

Not Applicable

16.2.2.1.2 BASES

QPTR is required to be verified within limits on a nominal basis per Technical Specification SR 3.2.4.1, with more frequent verifications required if the QPTR Alarm is non-functional. The more frequent verifications are sufficient to ensure QPTR to be within assumed limits.

The completion time for calculating QPTR when the QPTR Alarm is non-functional is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

16.3 INSTRUCTIONS

16.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

16.3.1.1 LIMITING CONDITION FOR OPERATION

See Technical Specification 3.3.1.

16.3.1.1.1 SURVEILLANCE REQUIREMENTS

See Technical Specification SR 3.3.1.16. The REACTOR TRIP SYSTEM RESPONSE TIME limits are given in Table 16.3-1.

16.3.1.1.2 BASES

See Technical Specification Bases for SR 3.3.1.16. Only the numerical limits are listed here. Technical Specification LCO and SR applicability requirements apply. Sections 16.0.1 and 16.0.2 do not apply. No response time testing requirements apply where N.A. is listed since no direct credit is taken in any safety analysis for the applicable trip function.

16.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

16.3.2.1 LIMITING CONDITION FOR OPERATION

See Technical Specification 3.3.2.

16.3.2.1.1 SURVEILLANCE REQUIREMENTS

See Technical Specification SR 3.3.2.10. The ENGINEEERED SAFETY FEATURES RESPONSE TIME limits are given in Table 16.3-2.

16.3.2.1.2 BASES

See Technical Specification Bases for SR 3.3.2.10. Only the numerical limits are listed here. Technical Specification LCO and SR applicability requirements apply. Sections 16.0.1 and 16.0.2 do not apply. No response time testing requirements apply where N.A. is listed since no direct credit is taken in any safety analysis for the applicable trip function.

Engineered Safety Features response times specified in Table 16.3-2 which include sequential operation of the RWST and VCT valves (Notes 3 and 4) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 7), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response time specified in Table 16.3-2 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

Because the Callaway configuration of the auxiliary feedwater (AFW) system does not permit the filling of the piping volume between the main feedwater isolation valve and the feedline check valve until after the main feedwater isolation valves are fully closed, an assumed AFW actuation delay implies feedwater isolation occurs. For the feedline break, loss of normal feedwater, and loss of non-emergency AC power accidents, the response time specified in Table 16.3-2 for initiation of AFW flow following the reactor trip on a low-low steam generator water level signal is based on an assumed delay for 90 seconds.

16.3.3 MONITORING INSTRUMENTATION

16.3.3.1 MOVABLE INCORE DETECTORS LIMITING CONDITION FOR OPERATION

The Movable Incore Detection System shall be FUNCTIONAL with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ .and $F_Q(Z)$

d. Calibrating the power distribution monitoring system (PDMS).

ACTION:

a. With the Movable Incore Detection System non-functional, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.3.3.1.1 SURVEILLANCE REQUIREMENTS

The Movable Incore Detection System shall be demonstrated FUNCTIONAL at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ .and $F_Q(Z)$.

d. Calibrating the PDMS.

16.3.3.1.2 BASES

The FUNCTIONALITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The FUNCTIONALITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one or more Power Range Neutron Flux Channels are inoperable. Power distribution measurement is generally performed, however, via the power distribution monitoring system (PDMS) when thermal power is greater than 25 percent rated thermal power (RTP). At thermal power levels less than 25 percent RTP, or when the PDMS is non-functional, the movable incore detector system is used. Since the movable incore detector system is employed by the PDMS, it shall be FUNCTIONAL before it is used to calibrate the PDMS.

16.3.3.2 SEISMIC INSTRUMENTATION LIMITING CONDITION FOR OPERATION

The seismic monitoring instrumentation shown in Table 16.3-3 shall be FUNCTIONAL.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments non-functional for more than 30 days, prepare and submit a Special Report to the Commission within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to FUNCTIONAL status.

b. The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.3.3.2.1 SURVEILLANCE REQUIREMENTS

16.3.3.2.1.a

Each of the above required seismic monitoring instruments shall be demonstrated FUNCTIONAL by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL OPERATIONAL TEST at the frequencies shown in Table 16.3-4.

16.3.3.2.1.b

Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.02 g shall be restored to FUNCTIONAL status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission within 14 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

16.3.3.2.2 BASES

The FUNCTIONALITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

16.3.3.3 METEOROLOGICAL INSTRUMENTATION LIMITING CONDITION FOR OPERATION

The meteorological monitoring instrumentation for each Function shown in Table 16.3-5 shall be FUNCTIONAL.

APPLICABILITY: At all times.

ACTION:

Note: Separate entry is allowed for each Function.

a. One or more Functions with one required meteorological channel non-functional, restore channel to FUNCTIONAL within 7 days or initiate an adverse condition type document in the corrective action program.

b. The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.3.3.3.1 SURVEILLANCE REQUIREMENTS

Each of the above meteorological monitoring instrumentation channels in Table 16.3-6 shall be demonstrated FUNCTIONAL by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown.

16.3.3.3.2 BASES

The FUNCTIONALITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," March 2007.

16.3.3.4 ACCIDENT MONITORING INSTRUMENTATION LIMITING CONDITION FOR OPERATION

The accident monitoring instrumentation channels shown in Table 16.3-7 shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of FUNCTIONAL accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 16.3-7, restore the non-functional channel(s) to FUNCTIONAL status within 30 days or prepare and submit a Special Report to the Commission within the following 14 days outlining the preplanned alternate method of monitoring, the cause of the non-functionality, and the plans and schedule for restoring the channels to FUNCTIONAL status.

b. With the number of FUNCTIONAL accident monitoring instrumentation channels, except the unit vent-high range noble gas monitor, less than the Minimum Channels FUNCTIONAL requirements of Table 16.3-7, restore one channel to FUNCTIONAL status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With the number of FUNCTIONAL channels for the unit vent-high range noble gas monitor less than the Minimum Channels FUNCTIONAL requirement of Table 16.3-7, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours and either restore the non-functional channel to FUNCTIONAL status within 7 days, or prepare and submit a Special Report to the Commission within the following 14 days outlining the preplanned alternate method of monitoring, the cause of the non-functionality, and the plans and schedule for restoring the channel to FUNCTIONAL status.

d. The provisions of Section 16.0.1.4 are not applicable.

16.3.3.4.1 SURVEILLANCE REQUIREMENTS

Each accident monitoring instrumentation channel shall be demonstrated FUNCTIONAL by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 16.3-8.

16.3.3.4.2 BASES

Certain recorders of the accident monitoring instrumentation channel are described in Appendix 7A as meeting requirements of Regulatory Guide 1.97 and are considered license commitments. The recorders are in place and should be operational, however,

they are not covered in the Technical Specifications. Therefore, Technical Specification 3.3.3 and Sections 16.3.3.4 and 16.3.3.4.1 are to be interpreted such that the instrumentation channels shall not be declared inoperable/non-functional if their associated recorder fails to meet surveillance test requirements. However, every reasonable effort should be expended to maintain these recorders in service. This is based on the definitions for CHANNEL CHECK and CHANNEL CALIBRATION which do not include recorders if other indication is provided and OPERABLE/FUNCTIONAL.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

16.3.3.4.c requires that an alternate method of monitoring be initiated within 72 hours of the Unit Vent - High Range Noble Gas Monitor (GTRE0021B) becoming non-functional. This monitor consists of three separate monitoring channels; the low range channel (#214), the mid range channel (#215), and the high range channel (#216). These three channels are designed with overlapping ranges and collectively meet the regulatory requirements for this monitor. There are two preplanned alternate methods: grab samples of the Unit Vent activity and data from field monitoring. Channel #214 can be used to monitor the Unit Vent if one or both #215 and #216 become non-functional when the activity is in the low range. If #214 becomes non-functional, manual grab samples from the Unit Vent or field monitoring can be used as the alternate method.

See also Technical Specification Bases 3.3.3.

16.3.3.5 LOOSE-PART DETECTION SYSTEM LIMITING CONDITION FOR OPERATION

The Loose-Part Detection System shall be FUNCTIONAL.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Loose-Part Detection System channels non-functional for more than 30 days, prepare and submit a Special Report to the Commission within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to FUNCTIONAL status.

b. The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.3.3.5.1 SURVEILLANCE REQUIREMENTS

Each channel of the Loose-Part Detection System shall be demonstrated FUNCTIONAL by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An CHANNEL OPERATIONAL TEST except for verification of Setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

16.3.3.5.2 BASES

The FUNCTIONALITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. Data acquisition equipment including a recording device capable of recording four channels simultaneously is required to maintain FUNCTIONALITY of the Loose-Part Detection System. The allowable out-of-service times and Surveillance Requirements were developed to be consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981. Callaway has adopted a modified methodology for performing the 18-month channel calibration recommended in Regulatory Position C.3.a.(3). Channel components are tested initially with a mechanical input as described in C.3.a.(3). Subsequent testing entails on-line performance monitoring and trending of channel components located in high dose areas in lieu of testing with a mechanical input. Injecting a test signal downstream of these components, and verifying setpoint and alarm circuit functions, tests the remainder of the channel. Review and evaluation of historical

surveillance data has determined that this is an acceptable means of verifying continued FUNCTIONALITY.

16.3.3.6 SPENT FUEL POOL CRITICALITY MONITOR LIMITING CONDITION FOR OPERATION

One Spent Fuel Pool Criticality - High Radiation Level channel (SDRE0037 or SDRE0038) shall be FUNCTIONAL with an alarm setpoint ≤15 mR/hr.

APPLICABILITY: Whenever fuel is in the spent fuel pool.

ACTION:

With no channel FUNCTIONAL, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same alarm setpoint is provided in the spent fuel pool area. Restore one channel to FUNCTIONAL status within 30 days or suspend all operations involving fuel movement in the fuel building.

The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.3.3.6.1 SURVEILLANCE REQUIREMENTS

Each Spent Fuel Pool Criticality - High Radiation Level channel shall be demonstrated FUNCTIONAL by performance of:

a. A CHANNEL CHECK at least once per 12 hours;

b. A CHANNEL OPERATIONAL TEST at least once per 92 days; and

c. A CHANNEL CALIBRATION at least once per 18 months.

16.3.3.6.2 BASES

The FUNCTIONALITY of the Spent Fuel Pool Criticality Monitors assures that radiation levels indicative of abnormal conditions is sensed and appropriate actions will be initiated when the radiation level reaches the alarm setpoint. The monitors meet the requirements of 10CFR70.24, as discussed in Section 12.3.4.1.2.8.

16.3.3.7 NEW FUEL STORAGE AREA CRITICALITY MONITOR LIMITING CONDITION FOR OPERATION

One New Fuel Storage Area Criticality - High Radiation Level channel (SDRE0035 or SDRE0036) shall be FUNCTIONAL with an alarm setpoint ≤15 mR/hr.

APPLICABILITY: Whenever fuel is in the new fuel storage area.

ACTION:

a. With no channel FUNCTIONAL, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same alarm setpoint is provided in the new fuel storage area. Restore one channel to FUNCTIONAL status within 30 days or suspend all operations involving fuel movement in the fuel building.

b. The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.3.3.7.1 SURVEILLANCE REQUIREMENTS

Each New Fuel Storage Area Criticality - High Radiation Level channel shall be demonstrated FUNCTIONAL by performance of:

a. A CHANNEL CHECK at least once per 12 hours;

b. A CHANNEL OPERATIONAL TEST at least once per 92 days; and

c. A CHANNEL CALIBRATION at least once per 18 months.

16.3.3.7.2 BASES

The FUNCTIONALITY of the New Fuel Storage Area Criticality Monitors assures that radiation levels indicative of abnormal conditions are sensed and appropriate actions will be initiated when the radiation level reaches the alarm setpoint. The monitors meet the requirements of 10CFR70.24, as discussed in Section 12.3.4.1.2.8.

16.3.3.8 POWER DISTRIBUTION MONITORING SYSTEM LIMITING CONDITION FOR OPERATION

The power distribution monitoring system (PDMS) and associated instrumentation shall be FUNCTIONAL with the minimum required channels from the plant computer as specified in Table 16.3-9, whenever the PDMS is used to perform core power distribution monitoring and related surveillances required by the Technical Specifications.

APPLICABILITY: When the PDMS is used to perform core power distribution monitoring and related surveillances required by the Technical Specifications during MODE 1 with reactor THERMAL POWER greater than or equal to 25% of RATED THERMAL POWER (RTP).

ACTION:

a. With one or more Functions with one or more required channels non-functional, declare the PDMS non-functional.

b. The provisions of Sections 16.0.1.4 are not applicable.

16.3.3.8.1 Surveillance Requirements

The PDMS shall be verified FUNCTIONAL:

a. By performance of a CHANNEL CHECK of the PDMS input instrumentation specified in Table 16.3-9, prior to use of the PDMS for core power distribution measurement purposes;

b. By performance of a calibration of the PDMS following each refueling outage, and thereafter:

   1) Every 31 EFPD with only minimum thermocouple coverage, or

   2) Every 180 EFPD with optimum thermocouple coverage;

c. By verifying that a CHANNEL CALIBRATION has been performed for each required instrumentation channel listed in Table 16.3-9 in accordance with the instrument's applicable Technical Specification Surveillance Requirement(s), prior to initial use of the PDMS for core power distribution measurement purposes following each refueling outage.

16.3.3.8.2 Bases

The power distribution monitoring system (PDMS) is used for periodic measurement of the core power distribution to confirm operation within design limits, and for periodic

calibration of the excore detectors. The system does not initiate any automatic protection action.

Specifically, the PDMS generates a continuous measurement of the core power distribution and requires information on current plant and core conditions in order to determine the core power distribution using the core peaking factor measurement and measurement uncertainty methodology described in WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994. The core and plant condition information, which includes reactor power level, average reactor vessel inlet temperature, control bank positions, the Power Range Detector calibrated voltage values, and input from the Core Exit Thermocouples, is used as input to the continuous core power distribution measurement software (i.e., BEACON) that continuously and automatically determines the current core peaking factor values, including the most limiting factors, $F^N_{\Delta H}$ and FQ(Z). The measured peaking factor values are provided at nominal one-minute intervals to allow operators to confirm that the core peaking factors are within design limits. The peaking factor limit margins include measurement uncertainty which bounds the actual measurement uncertainty of a FUNCTIONAL PDMS.

Functionality of the PDMS is dependent on the specified number of channels from the plant computer for each Function listed in Table 16.3-9. The PDMS is FUNCTIONAL when the required channels are available, when it has been calibrated with an incore flux map within the required frequency, and when the reactor power level is at least 25% of RTP. The PDMS must be calibrated above 25% RTP to assure the accuracy of the calibration data set which can be generated from an incore flux map, core exit thermocouples, and the other instrumentation channels. Below 25% RTP, the PDMS is non-functional since the calculated power distribution is of reduced accuracy and may not be bounded by the uncertainties documented in WCAP-12472.

With one or more required channels from the plant computer non-functional or unavailable as input to the PDMS (or if the PDMS is non-functional for any other reason), the PDMS must be declared non-functional and shall not be used to perform a core power distribution measurement.

Surveillance Requirements are specified for the PDMS to support or provide assurance of its FUNCTIONALITY, as follows:

a. Performance of a CHANNEL CHECK for the PDMS input instrumentation prior to using the PDMS to obtain a core power distribution measurement ensures that no gross instrumentation failure has occurred, thus providing added assurance that the required inputs to the PDMS are available. A feature of the PDMS is its capability to automatically check its required inputs and provide a status indication that confirms all required inputs are available and working. The CHANNEL CHECK requirement can thus be satisfied by checking the BEACON status

indicator on the BEACON PDMS monitor screen prior to use of the PDMS for core power distribution measurement purposes.

b. Upon initial plant startup following refueling, the PDMS uses a calibration data set calculated by the core designer for the new core. An accurate incore flux map for PDMS calibration may be obtained upon exceeding 25% RTP. The initial calibration data set generated in each operating cycle must utilize incore flux measurements from at least 75% of the incore thimbles, with at least two incore thimbles in each core quadrant. The incore flux measurements in combination with inputs from the Table 16.3-9 channels are used to generate the updated calibration data set, including the nodal calibration factors and the thermocouple mixing factors.

   After completion of the first PDMS calibration following refueling, the PDMS must continue to be periodically calibrated at least once per 31 EFPD when only the minimum core exit thermocouple coverage is available for the PDMS calibration (which requires at least 13 thermocouples available with a minimum of two per core quadrant), or at least once per 180 EFPD when the optimum core exit thermocouple coverage is available for the PDMS calibration. Optimum thermocouple coverage also requires at least 13 thermocouples available with a minimum of two per core quadrant. In addition, optimum thermocouple coverage provides coverage of all interior fuel assemblies. (Coverage of fuel assemblies with a face along the baffle is not required.) Fuel assemblies that are within a chess knight's move of an OPERABLE thermocouple are also covered.

   With regard to thermocouple coverage, the PDMS software automatically analyzes the available thermocouple coverage, consistent with the above criteria, and determines the next surveillance interval for calibration (i.e., 31 or 180 EFPD). Typically, the required interval is reported in the surveillance report that is generated upon completion of the periodic core power distribution measurement.

c. Verification that a CHANNEL CALIBRATION has been performed for each of the instrument channels that provide input to the PDMS provides assurance of channel FUNCTIONALITY and accurate input to the PDMS prior to use of the PDMS for core power distribution measurement purposes following each refueling outage. The intent is that each instrument channel has been subject to a CHANNEL CALIBRATION at the frequency or interval required by the instrument's associated Technical Specification Surveillance Requirement(s). This can be verified by confirming that the instruments' surveillances are "current" with respect to their specified test intervals, based on their last performance and satisfactory completion (as scheduled per the plant surveillance program). A CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP.

16.3.4 TURBINE OVERSPEED PROTECTION

16.3.4.1 LIMITING CONDITION FOR OPERATION

The Turbine Overspeed Protection System shall be FUNCTIONAL.

APPLICABILITY:MODES 1, 2, and 3. (Not applicable in MODE 2 or 3 with all main steam line isolation valves and associated bypass valves in the closed position and all other steam flow paths to the turbine isolated.)

ACTION:

a. With the required Turbine Overspeed Protection System non-functional, evaluate the condition per the corrective action program.

16.3.4.1.1 SURVEILLANCE REQUIREMENTS

16.3.4.1.1.a

The provisions of Section 16.0.2.4 are not applicable.

16.3.4.1.1.b

The required Turbine Overspeed Protection System shall be maintained, calibrated, tested, and inspected in accordance with the Callaway Plant's Turbine Overspeed Protection Reliability Program. Adherence to this program shall demonstrate FUNCTIONALITY of this system.

16.3.4.1.2 BASES

This Requirement is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are FUNCTIONAL and will protect the turbine from excessive overspeed. Although the orientation of the turbine is such that the number of potentially damaging missiles which could impact and damage safety-related components, equipment, or structures is minimal, protection from excessive turbine overspeed is required.

The Turbine Overspeed Protection System consists of two major independent systems: The Primary Overspeed Trip System and the Emergency Overspeed Trip System. The systems are designed with multiple redundancies to prevent turbine rotor speed from exceeding 120% of rated speed. Failure of any single component will not result in rotor speed exceeding this design rated speed. The Turbine Overspeed Protection Reliability Program (TOPRP) contains the requirements for inspection, maintenance, calibration and testing of the Turbine Overspeed Protection System. The TOPRP provides assurance that components in the Turbine Overspeed Protection System will be tested

periodically to ensure their ability to function properly if needed to prevent turbine overspeed.

TABLE 16.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES(2)

FUNCTIONAL UNIT			RESPONSE TIME
1.	Manual Reactor Trip		N.A.
2.	Power Range, Neutron Flux		≤ 0.5 second*
3.	Power Range, Neutron Flux, High Positive Rate		≤ 0.5 second*
4.	Deleted		
5.	Intermediate Range, Neutron Flux		N.A.
6.	Source Range, Neutron Flux		N.A.
7.	Overtemperature ΔT		*
8.	Overpower ΔT		*
9.	Pressurizer Pressure-Low		≤ 2.0 seconds
10.	Pressurizer Pressure-High		≤ 1.0 seconds
11.	Pressurizer Water Level-High		N.A.
12.	Reactor Coolant Flow-Low		
	a.	Single Loop (Above P-8)	≤ 1.0 second
	b.	Two Loops (Above P-7 and below P-8)	≤ 1.0 second
13.	Steam Generator Water Level Low-Low		
	a.	Steam Generator Water Level Low-Low (Adverse Containment Environment)	≤ 2.0 seconds(1)
	b.	Steam Generator Water Level Low-Low (Normal Containment Environment)	≤ 2.0 seconds(1)
	c.	Not used	
	d.	Containment Pressure - Environmental Allowance Modifier	≤ 2.0 seconds(1)
14.	Undervoltage-Reactor Coolant Pumps		≤ 1.5 seconds
15.	Underfrequency-Reactor Coolant Pumps		≤ 0.6 seconds

TABLE 16.3-1 (Sheet 2)

FUNCTIONAL UNIT	RESPONSE TIME
16. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input from ESF	(3)
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	(4)
20. Automatic Trip and Interlock Logic	(5)

TABLE 16.3-1 (Sheet 3)

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* Functional Units 7 and 8 instrument response times include the following:

1. A lag of 4 seconds for the RTD/thermowell time response.

2. A delay of 2 seconds for the channel electronics delay and the trip logic circuitry delay, plus the time for the reactor trip breakers to open and the time for the control rod drive mechanism stationary grippers to disengage (gripper release time).

3. Dynamic Tavg/DT signal compensation (filter, lead-lag, and rate-lag compensators) with the time constants specified in the core operating limits report (COLR).

Safety analyses credit an additional 2.0 second lag for thermal lag and transport delays. See Table 15.0-4.

Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be verified from detector output or input to the first electronic component in channel.

(1) Response times noted above include the transmitters, 7300 process protection cabinets, solid state protection cabinets, and actuation devices only.

(2) NRC approved the use of allocated response times for some components in their letter from Jack Donohew to Garry L. Randolph, "Application of WCAP-14036-P-A for Response Time Testing Elimination at Callaway Plant, Unit 1 (TAC NO. MA7283)," dated March 3, 2000.

(3) Covered in the response times listed in Table 16.3-2.

(4) The response time for the reactor trip breakers to open and the gripper release time are satisfied by measurement and included in the response time for each required reactor trip function (i.e., all the above except where N.A. is listed).

(5) Logic is covered by an allocated response time per note (2) above.

TABLE 16.3-2 ENGINEERED SAFETY FEATURES RESPONSE TIMES[11]

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Purge Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Service Water	N.A.
j. Containment Cooling	N.A.
k. Control Room Isolation	N.A.
l. Reactor Trip	N.A.
m. Emergency Diesel Generators	N.A.
n. Component Cooling Water	N.A.
o. Turbine Trip	N.A.
2. Containment Pressure-High-1	
a. Safety Injection (ECCS)	≤ 29[7][15]/27[4]/39[3]
1) Reactor Trip	≤ 2[18]
2) Feedwater Isolation	≤ 2[5]
3) Phase "A" Isolation	≤ 1.5[5]
4) Motor-Driven Auxiliary Feedwater	≤ 60[1]
5) Essential Service Water	≤ 85[1]
6) Containment Cooling	≤ 85[1]
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14[6]
9) Turbine Trip	N.A.
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 29[7][15]/27[4]/39[3]
1) Reactor Trip	≤ 2[18]
2) Feedwater Isolation	≤ 2[5]
3) Phase "A" Isolation	≤ 2[5]

TABLE 16.3-2 (Sheet 2)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4) Motor-Driven Auxiliary Feedwater	$\leq 60^{(1)}$
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 39^{(3)}/27^{(4)}/29^{(7)(15)}$
1) Reactor Trip	$\leq 2^{(18)}$
2) Feedwater Isolation	$\leq 2^{(5)}$
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Motor-Driven Auxiliary Feedwater	$\leq 60^{(1)}$
5) Essential Service Water	$\leq 85^{(1)}$
6) Containment Cooling	$\leq 85^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
b. Steam Line Isolation	$\leq 2^{(5)}$
5. Containment Pressure-High-3	
a. Containment Spray	$\leq 32^{(1)}/20^{(2)}$
b. Phase "B" Isolation	$\leq 31.5^{(12)}$
6. Containment Pressure-High-2	
Steam Line Isolation	$\leq 2^{(5)}$
7. Steam Line Pressure-Negative-Rate-High	
Steam Line Isolation	$\leq 2^{(5)}$
8. Steam Generator Water Level-High-High	
a. Feedwater Isolation	$\leq 2^{(5)}$
b. Turbine Trip	N.A.
9. Steam Generator Water Level-Low-Low	
a. Start Motor-Driven Auxiliary Feedwater Pumps	$\leq 60^{(8)(16)(17)}$

TABLE 16.3-2 (Sheet 3)

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
	b.	Start Turbine-Driven Auxiliary Feedwater Pump	$\leq 60^{(8)(17)}$
	c.	Feedwater Isolation	$\leq 2^{(5),(8)}$
10.	Loss-of-Offsite Power		
		Start Turbine-Driven Auxiliary Feedwater Pump	$\leq 60^{(9)}$
11.	Trip of All Main Feedwater Pumps		
		Start Motor-Driven Auxiliary Feedwater Pumps	N.A.
12.	Auxiliary Feedwater Pump Suction Pressure-Low		
		Transfer to Essential Service Water	$\leq 60^{(1)}$
13.	RWST Level-Low-Low Coincident with Safety Injection		
		Automatic Switchover to Containment Sump	$\leq 40^{(10)}$
14.	Loss of Power		
	a.	4 kV Bus Undervoltage-Loss of Voltage	$\leq 14^{(6)}$
	b.	4 kV Bus Undervoltage-Grid Degraded Voltage	$\leq 144^{(13)}$
15.	Phase "A" Isolation		
	a.	Control Room Isolation	N.A.
	b.	Containment Purge Isolation	$\leq 2^{(5)}$
16.	Control Room High Gaseous Activity		
		Control Room Isolation	$\leq 60^{(14)}$
17.	Fuel Building Ventilation High Gaseous Activity		
		Emergency Exhaust System in the FBVIS Mode	$\leq 90^{(19)}$

TABLE NOTATIONS

(1) Signal actuation, diesel generator starting, and sequencer loading delays included. Valve stroke times and spin-up times for pumps and fans included, as applicable.

TABLE 16.3-2 (Sheet 4)

(2) Diesel generator starting delay <u>not</u> included. Offsite power available. Signal actuation, sequencer loading, and pump spin-up delays included.

(3) Signal actuation, diesel generator starting and sequencer loading delays included. RHR pumps <u>not</u> included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(4) Diesel generator starting and sequencer loading delays not included. Offsite power available. RHR pumps <u>not</u> included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) and signal actuation delays are included.

(5) Does not include valve closure time.

(6) Includes signal actuation delay and time for diesel to reach full speed.

(7) Signal actuation, diesel generator starting and sequencer loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is <u>not</u> included. Response time assumes only opening of RWST valves.

(8) Response times noted above include the transmitters, 7300 process protection cabinets, solid state protection cabinets, and actuation devices only.

(9) Includes signal actuation, valve stroke, and pump spin-up times.

(10) Includes signal actuation and valve stroke times (sequential suction transfer as containment recirculation sump valves must fully open before RWST valves go fully closed).

(11) NRC approved the use of allocated response times for some components in their letter from Jack Donohew to Garry L. Randolph, "Application of WCAP-14036-P-A for Response Time Testing Elimination at Callaway Plant, Unit 1 (TAC NO. MA7283)," dated March 3, 2000.

(12) Signal actuation and valve stroke time delays included.

(13) Signal actuation, diesel generator starting, loss of voltage time delay relay, LSELS degraded voltage bistable delay timers and degraded voltage feeder breaker time delay relays included.

(14) The radiation monitor detector is excluded from response time testing. The stated response time accounts for the elapsed time between introduction of a count rate from the detector corresponding to the actuation setpoint and repositioning of the components necessary to achieve Control Room isolation.

TABLE 16.3-2 (Sheet 5)

(15) The maximum allowed stroke time for RHR-mini-flow valves EJFCV0610/0611 to close is 15 seconds. Therefore, full RHR flow is not delivered to the core for 44 seconds (29 seconds plus 15 seconds) following the initiation of an SI signal.

(16) Response times noted above include the transmitters, 7300 process protection cabinets, solid state protection cabinets, and actuation devices only. For the loss of non-emergency AC and the loss of normal feedwater accident analyses, initiation of AFW flow is assumed delayed for 90 seconds following reactor trip on a low-low steam generator water level signal.

(17) Response times noted above include the transmitters, 7300 process protection cabinets, solid state protection cabinets, and actuation devices only. For the feedline break accident analysis, initiation of AFW flow is assumed delayed for 90 seconds following reactor trip on low-low steam generator water level signal.

(18) The response time for the reactor trip breakers to open and the gripper release time are satisfied by measurement and included in the response time for each required reactor trip functon.

(19) The radiation monitor detector is excluded from response time testing. The stated response time accounts for the elapsed time between introduction of a count rate from the detector corresponding to the actuation setpoint and repositioning of the components necessary to place the Emergency Exhaust System in the FBVIS mode of operation.

TABLE 16.3-3 SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE			MINIMUM INSTRUMENTS FUNCTIONAL
1. Triaxial Peak Recording Accelerographs				
a. Radwaste Base Slab	± 1.0 g			1
b. Control Room	± 1.0 g			1
c. ESW Pump Facility	± 1.0 g			1
d. Ctmt. Structure	± 2.0 g			1
e. Auxiliary Bldg. SI Pump Suctions	± 1.0 g			1
f. SGB Piping	± 2.0 g			1
g. SGC Support	± 1.0 g			1
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)				
a. Ctmt. Base Slab	± 1.0 g			1
b. Ctmt. Oper. Floor	± 1.0 g			1
c. Reactor Support	± 1.0 g			1
d. Aux. Bldg. Base Slab	± 1.0 g			1
e. Aux. Bldg. Control Room Air Filters	± 1.0 g			1
f. Free Field	± 0.5 g			1
3. Triaxial Response-Spectrum Recorder (Passive)				
a. Ctmt. Base Slab	± 1.0 g			1
4. Triaxial Seismic Switches	ACCELERATION LEVEL			
	North	East	Vertical	
a. OBE Ctmt. Base Slab	0.09 g	0.09 g	0.13 g	1
b. SSE Ctmt. Base Slab	0.13 g	0.14 g	0.20 g	1
c. OBE Ctmt. Oper. Fl.	0.10 g	0.10 g	0.13 g	1
d. SSE Ctmt. Oper. Fl.	0.14 g	0.16 g	0.21 g	1
e. System Trigger	0.02 g	0.02 g	0.02 g	1

TABLE 16.3-4 SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Peak Recording Accelerographs			
a. Radwaste Base Slab	N.A.	R	N.A.
b. Control Room	N.A.	R	N.A.
c. ESW Pump Facility	N.A.	R	N.A.
d. Ctmt. Structure	N.A.	R	N.A.
e. Auxiliary Bldg. SI Pump Suction	N.A.	R	N.A.
f. SGB Piping	N.A.	R	N.A.
g. SGC Support	N.A.	R	N.A.
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)			
a. Ctmt. Base Slab	M	R	SA
b. Ctmt. Oper. Floor	M	R	SA
c. Reactor Support	M	R	SA*
d. Aux. Bldg. Base Slab	M	R	SA*
e. Aux. Bldg. Control Room Air Filters	M	R	SA*
f. Free Field	M	R	SA*
3. Triaxial Response-Spectrum Recorder (Passive)			
Ctmt. Base Slab	N.A.	R	N.A.**
4. Triaxial Seismic Switches			
a. OBE Ctmt. Base Slab	M	R	SA
b. SSE Ctmt. Base Slab	M	R	SA
c. OBE Ctmt. Oper. Fl.	M	R	SA
d. SSE Ctmt. Oper. Fl.	M	R	SA
e. System Trigger	M	R	SA

* The Bistable Trip Setpoint need not be determined during the performance of an ANALOG CHANNEL OPERATIONAL TEST.

** Checking at the Main Control Board Annunciators for contact closure output in the Control Room shall be performed at least once per 184 days.

TABLE 16.3-5 METEOROLOGICAL MONITORING INSTRUMENTATION

FUNCTION		LOCATION	REQUIRED CHANNELS
1.	Wind Speed	Nominal Elev. 10 m	1
2.	Wind Speed	Nominal Elev. 60 m	1
3.	Wind Direction	Nominal Elev. 10 m	1
4.	Wind Direction	Nominal Elev. 60 m	1
5.	Air Temperature - ΔT	Nominal Elev. 10 m - 60m	1

TABLE 16.3-6 METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 10 m	D	SA
b. Nominal Elev. 60 m	D	SA
2. Wind Direction		
a. Nominal Elev. 10 m	D	SA
b. Nominal Elev. 60 m	D	SA
3. Air Temperature - ΔT		
a. Nominal Elev. 10-60 m	D	SA

TABLE 16.3-7 ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS FUNCTIONAL
1.	Containment Pressure - Extended Range	2	1
2.	Safety Valve Position Indicator	1/Valve	1/Valve
3.	Unit Vent - High Range Noble Gas Monitor (GT-RE-21)*#	1	1
4.	PORV Position Indicator**	1/Valve	1/Valve
5.	PORV Block Valve Position Indicator***	1/Valve	1/Valve
6.	Containment Hydrogen Analyzers	2	1

* See also Section 16.11.2.4.

** Not applicable if the associated block valve is in the closed position.

*** Not applicable if the block valve is verified in the closed position and power is removed.

\# The RM 23 or RM 11 are equivalent indications to satisfy Section 16.3.3.4 requirements.

TABLE 16.3-8 ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Extended Range	M	R
2. Safety Valve Position Indicator	M	N.A.
3. Unit Vent - High Range Noble Gas Monitor*#	M	R
4. PORV Position Indicator**	M	N.A.
5. PORV Block Valve Position Indicator***	M	N.A.
6. Containment Hydrogen Analyzers	N.A.	R

* See also Section 16.11.2.4.

** Not applicable if the associated block valve is in the closed position.

*** Not applicable if the block valve is verified in the closed position and power is removed.

\# The RM23 or RM11 are equivalent indications to satisfy Section 16.3.3.4 requirements.

TABLE 16.3-9 POWER DISTRIBUTION MONITORING SYSTEM INSTRUMENTATION

FUNCTION		REQUIRED CHANNELS
1.	Control Bank Position	4[(a)]
2.	RCS Cold Leg Temperature, Tcold	2
3.	Reactor Power Level	1[(b)]
4.	NIS Power Range Excore Detector Section Signals	6[(c)]
5.	Core Exit Thermocouple Temperatures	13 with ≥ 2 per core quadrant

a. Control Bank position inputs may be bank positions from either valid Demand Position indications or the average of all valid individual RCCA positions in the bank determined from Digital Rod Position Indication (DRPI) System values for each Control Bank. A maximum of one rod position indicator per group may be inoperable when RCCA position indications are being used as input to the PDMS.

b. Reactor Power Level input may be reactor thermal power derived from either a valid secondary calorimetric measurement, the average Power Range Neutron Flux Power, or the average RCS Loop DT.

c. The total must consist of three pairs of corresponding upper and lower detector sections.

16.4 REACTOR COOLANT SYSTEM

16.4.1 SAFETY VALVES

16.4.1.1 SHUTDOWN LIMITING CONDITION FOR OPERATION

A minimum of one pressurizer Code safety valve shall be FUNCTIONAL with a lift setting of ≥ 2411 psig and ≤ 2509 psig. (The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.)

APPLICABILITY: MODE 4 with any RCS cold leg temperature ≤ 275 degrees F and MODE 5 when the RCS is not vented to atmosphere by a vent opening of ≥ 2.0 square inches.

ACTION:

With no pressurizer Code safety valve FUNCTIONAL, immediately suspend all operations involving positive reactivity additions that could result in loss of required SDM and place a FUNCTIONAL RHR loop into operation in the shutdown cooling mode.

16.4.1.1.1 SURVEILLANCE REQUIREMENTS

No additional requirements other than those required by the Inservice Testing Program. The lift setting shall be within ±1% of 2460 psig following testing.

The provisions of Section 16.0.2.2 are not applicable.

16.4.1.1.2 BASES

The pressurizer Code safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 with the reactor vessel head on. Overpressure protection is not required in MODE 6 with the reactor vessel head removed and the RCS vented with a vent path ≥ 2.0 square inches. With the RCS depressurized, analyses show a vent size of 2.0 square inches is capable of mitigating the limiting cold overpressurization transient.

In MODES 1, 2, 3 and portions of MODE 4 (when any RCS cold leg temperature is > 275 degrees F), FUNCTIONALITY of three safety valves is required because the combined capacity is needed to keep the RCS pressure below 110% of its design value during certain accidents. However, the relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are FUNCTIONAL during shutdown, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

In MODE 4 with any RCS cold leg temperature ≤ 275 degree F, in MODE 5, and in MODE 6, with the reactor vessel head on, overpressure protection is also provided by operating procedures and by meeting the requirements of LCO 3.4.12, “Cold Overpressure Mitigation System (COMS)”. COMS provides diverse methods of protection against RCS overpressurization at low temperatures. Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

With the required pressurizer safety valve non-functional, the plant must be placed in a condition that minimizes the risk of a pressure spike large enough to actuate a safety valve. This is done by suspending operations involving positive reactivity additions that could result in loss of required SDM of LCO 3.1.1, “Shutdown Margin (SDM).” Suspending positive reactivity additions that could result in failure to meet the minimum SDM limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

The pressurizer safety valves are tested under the Inservice Testing (IST) Program. Valves tested under the IST Program are governed by the testing requirements of the ASME OM Code. The upper and lower pressure limits are based on the tolerance requirements assumed in the safety analyses. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The provisions of Section 16.0.2.2 do not apply, as the Inservice Testing Program includes its own provisions for extending test frequencies.

Reference: TS 3.4.10 Bases; TS 3.4.12 Bases; and License Amendment 149.

16.4.2 OPERATIONAL LEAKAGE

16.4.2.1 LIMITING CONDITION FOR OPERATION

Reactor coolant pump (RCP) seal water return flow rate shall be limited to 8 gpm per pump at an RCS pressure of 2235 ± 20 psig.

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

ACTION:

With any RCP seal water return flow rate not within limit, reduce the flow rate to within limit within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

16.4.2.1.1 SURVEILLANCE REQUIREMENTS

Verify each RCP seal water return flow rate is within limit with the RCS pressure at 2235 ± 20 psig at least once per 31 days. The provisions of Section 16.0.2.4 are not applicable for entry into MODE 3 or 4.

Verify the RCP seal water return throttle valve (BGV0202) mechanical position stop is in the correct position at least once per 18 months.

16.4.2.1.2 BASES

The RCP seal water return flow rate limit restricts operation when the total flow from the RCP seals exceeds 8 gpm per pump at a nominal pressure of 2235 psig. This limitation ensures adequate performance of the RCP seals.

The RCP seal water return throttle valve (BGV0202) is set to ensure proper ECCS performance. Once set, this valve is secured with a locking device and a mechanical position stop. See the Technical Specification 3.5.2 Bases and Section 16.5.2 for further discussion of required ECCS performance.

16.4.3 CHEMISTRY

16.4.3.1 LIMITING CONDITION FOR OPERATION

The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 16.4-3.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

16.4.3.1.1 SURVEILLANCE REQUIREMENTS

The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 16.4-4.

16.4.3.1.2 BASES

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects

of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

16.4.4 PRESSURE/TEMPERATURE LIMITS

16.4.4.1 PRESSURIZER LIMITING CONDITION FOR OPERATION

The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period.

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 583°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

16.4.4.1.1 SURVEILLANCE REQUIREMENTS

The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

16.4.4.1.2 BASES

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. These rates do not apply to the pressurizer surge line. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 583°F.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the

ASME Code Requirements. The following references provide additional information on the application of these limits:

- WCAP-14717 Revision 1, "Westinghouse Owners Group Evaluation of the Effects of Insurge/Outsurge Out-of-Limit Transients on the Integrity of the Pressurizer," August 1988.

- Supplement 1 to WCAP-14717 Revision 1, "Basis and Application of Pressurizer Heatup and Cooldown Limits," November 1999.

- Westinghouse letter SCP-06-16 dated April 7, 2006.

16.4.5 STRUCTURAL INTEGRITY

16.4.5.1 LIMITING CONDITION FOR OPERATION

The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Section 16.4.5.1.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Section 16.0.1.4 are not applicable.

16.4.5.1.1 SURVEILLANCE REQUIREMENTS

Verify the structural integrity of ASME Code Class 1, 2, and 3 components in accordance with the Inservice Inspection and Testing Programs.

16.4.5.1.2 BASES

The Inservice Inspection and Testing Programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI and Section OM (respectively) of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(f) and 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(f)(6)(i) and 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

Visual tests (i.e., VT-2) inspections for Class 2 components may be performed with the RCS temperature greater than 200°F. When the loss of any ASME Class 2 component structural integrity is identified per Section 16.4.5.1.1 (i.e. ASME Section XI/ISI) and an RCS temperature greater than 200°F, the approved actions are:

(1) Isolate the affected component(s) from service and enter the appropriate action statement for the component/system isolated.

(2) If the component/system cannot be isolated, refer to Section 16.0.1.3 guidance.

16.4.6 REACTOR COOLANT SYSTEM VENTS

16.4.6.1 LIMITING CONDITION FOR OPERATION

At least one reactor vessel head vent path consisting of at least two valves in series powered from emergency busses shall be FUNCTIONAL and closed.

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

ACTION:

With the above reactor vessel head vent path non-functional, STARTUP and/or POWER OPERATION may continue provided the non-functional vent path is maintained closed with power removed from the valve actuator of all the valves in the non-functional vent path; restore the non-functional vent path to FUNCTIONAL status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

16.4.6.1.1 SURVEILLANCE REQUIREMENTS

Each reactor vessel head vent path shall be demonstrated FUNCTIONAL at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING, and

c. Verifying flow through the reactor vessel head vent paths during venting during COLD SHUTDOWN or REFUELING.

16.4.6.1.2 BASES

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The FUNCTIONALITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

16.4.7 PRESSURIZER PORVs (Deleted)

16.4.7.1 LIMITING CONDITION FOR OPERATION (Deleted)

16.4.7.1.1 SURVEILLANCE REQUIREMENTS (Deleted)

16.4.7.1.2 BASES (Deleted)

16.4.8 RCS PRESSURIZER/TEMPERATURE LIMITS

16.4.8.1 LIMITING CONDITION FOR OPERATION

The RCS heatup, RCS cooldown, and PORV COMS setpoint figures in the Pressure and Temperature Limits Reports (PTLR) shall be based upon results obtained from the reactor vessel material irradiation surveillance specimens.

APPLICABILITY: At all times.

ACTION:

With the LCO not met, enter Section 16.0.1.3.

16.4.8.1.1 SURVEILLANCE REQUIREMENTS

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties, as required by 10CFR50 Appendix H.

16.4.8.1.2 BASES

This Requirement complies with NRC regulations and provides the basis for updating RCS heatup, RCS cooldown, and PORV COMS setpoint figures in the PTLR. Failure to meet this Requirement may result in entry into Technical Specifications 3.4.3 and 3.4.12, as well as a violation of 10CFR50 Appendix H.

16.4.9 DNB PARAMETERS - FEEDWATER FLOW VENTURI

16.4.9.1 LIMITING CONDITION FOR OPERATION

Each feedwater flow venturi shall be FUNCTIONAL.

APPLICABILITY: MODE 1

ACTION:

With the LCO not met, enter Section 16.0.1.3.

16.4.9.1.1 SURVEILLANCE REQUIREMENTS

Each feedwater flow venturi shall be inspected and cleaned as necessary at least once per 18 months.

16.4.9.1.2 BASES

Potentially undetected fouling of a feedwater venturi(s) could bias the result from the precision heat balance in a non-conservative manner. Therefore, an inspection is performed on the feedwater venturis each refueling outage. This Frequency has been shown by experience to be acceptable for ensuring the continued performance of the feedwater venturis. Refer to Technical Specification 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits.”

TABLE 16.4-1

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TABLE 16.4-2

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TABLE 16.4-3 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

* Limit not applicable with T_{avg} less than or equal to 250°F.

TABLE 16.4-4 REACTOR COOLANT SYSTEM CHEMISTRY SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Hydrogen*	At least once per 72 hours
Dissolved Oxygen*	At least once per 72 hours whenever hydrogen is <15 cc/kg
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

* Not required with T_{avg} less than or equal to 250°F.

16.5 EMERGENCY CORE COOLING SYSTEMS

16.5.1 ACCUMULATORS

16.5.1.1 LIMITING CONDITION FOR OPERATION

One water level channel and one pressure channel shall be FUNCTIONAL for each ECCS accumulator.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With less than the required number of channels FUNCTIONAL, enter Section 16.0.1.3.

16.5.1.1.1 SURVEILLANCE REQUIREMENTS

Each accumulator water level and pressure channel shall be demonstrated FUNCTIONAL at least once per 18 months by the performance of a CHANNEL CALIBRATION.

16.5.1.1.2 BASES

See ULNRC-2960 dated February 17, 1994, OL Amendment No. 91 dated August 5, 1994, and FSAR CN 94-57.

* RCS pressure above 1000 psig.

16.5.2 ECCS SUBSYSTEMS - T_{avg} ≥200°F

16.5.2.1 LIMITING CONDITION FOR OPERATION

Two Emergency Core Cooling System (ECCS) trains shall be FUNCTIONAL in MODES 1, 2, and 3 and one train (one RHR subsystem and one centrifugal charging pump subsystem) shall be FUNCTIONAL in MODE 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the required number of ECCS trains FUNCTIONAL solely due to failure to meet the Surveillance Requirements of Section 16.5.2.1.1, enter Section 16.0.1.3.

16.5.2.1.1 SURVEILLANCE REQUIREMENTS

Each ECCS subsystem shall be demonstrated FUNCTIONAL:

a. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

   1) Once prior to entry into MODE 4 from MODE 5 for all accessible areas of the containment; and

   2) At the completion of each containment entry for all affected areas within containment.

b. By performing a flow balance test of the affected centrifugal charging pump portions of the ECCS subsystem, during shutdown, following completion of modifications to that centrifugal charging pump subsystem that alter the subsystem flow characteristics. The test shall be performed with a single pump running and the throttle valves set within setting tolerance to provide balanced branch line flow. Under these conditions there is zero mini-flow and 87 plus 2 or minus 4 gpm simulated reactor coolant pump seal injection line flow. This test shall verify:

   1) The total flow to the four branch lines is less than or equal to 461 gpm, and

   2) The total flow to the four branch lines is greater than or equal to 411 gpm (which corresponds to an analyzed flow rate of 305.25 gpm through the three lowest flow branch lines).

c. By performing a flow balance test of the affected safety injection pump portions of the ECCS subsystem, during shutdown, following completion of modifications to that safety injection pump subsystem that alter the subsystem flow characteristics. The test shall be performed with a single pump running and the throttle valves set within setting tolerance to provide balanced branch line flow. This test shall verify:

   1) The total pump flow rate is less than or equal to 675 gpm, and

   2) The total flow to the four branch lines is greater than or equal to 611.3 gpm (which corresponds to an analyzed flow rate of 455.6 gpm through the three lowest flow branch lines).

d. By performing a flow test, during shutdown, following completion of modifications to the RHR subsystems that alter the subsystem flow characteristics and verifying that for RHR pump lines, with a single pump running:

   1) The sum of the injection line flow rates is greater than or equal to 3800 gpm, and

   2) The total pump flow rate is less than or equal to 5500 gpm.

16.5.2.1.2 BASES

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem.

The SR related to maintaining cleanliness in the containment assures that the containment sump screens are not fouled by debris. Loose rags, clothing, or other debris may be entrained in the flow of ECCS and containment spray water during the recirculation mode of ECCS operation and be transported to the sump screens. The Frequency of performance for this SR ensures that no restrictions to recirculation flow would occur when the ECCS system is required to be FUNCTIONAL.

The SRs related to flow balance testing of the ECCS subsystems ensure that the assumptions used in the safety analyses are met and that subsystem FUNCTIONALITY is maintained. While not under the direct control of the operators in the control room, the flow balance of ECCS subsystems is assumed in the safety analyses. The safety analyses make assumptions with respect to: (1) both the maximum and minimum total system resistance; (2) both the maximum and minimum branch injection line resistance; and (3) the maximum and minimum ranges of potential pump performance. These resistances and ranges of pump performance are used to calculate the maximum and minimum ECCS flows assumed in the safety analyses.

The CCP minimum flow SR provides the absolute minimum injected flow assumed in the safety analyses. The maximum total system resistance defines the range of minimum

flows (including the minimum flow SR), with respect to pump head, that is assumed in the safety analyses. Therefore, the CCP total system resistance $([P_d + (Z_d - Z_{RCS})] / Q_d^2)$ must not be greater than 1.004E-2 ft/gpm2, where P_d is pump discharge pressure in feet, Z_d is the pump discharge elevation in feet, Z_{RCS} is RCS water level elevation in feet, and Q_d is the total pump flow rate in gpm.

The SI pump minimum flow SR provides the absolute minimum injected flow assumed in the safety analyses. The maximum total system resistance defines the range of minimum flows (including the minimum flow SR), with respect to pump head, that is assumed in the safety analyses. Therefore, the safety injection pump total system resistance $([P_d + (Z_d - Z_{RCS})] / Q_d^2)$ must not be greater than 0.414E-2 ft/gpm^2, where P_d is pump discharge pressure in feet, Z_d is the pump discharge elevation in feet, Z_{RCS} is RCS water level elevation in feet, and Q_d is the total pump flow rate in gpm.

The CCP maximum total pump flow SR ensures the maximum injection flow limit of 550 gpm is not exceeded. This value of flow is comprised of the total flow to the four branch lines of 461 gpm and a seal injection flow of 87 gpm plus 2 gpm for instrument uncertainties.

The SI pump maximum total pump flow SR ensures the maximum injection flow limit of 675 gpm is not exceeded. This value of flow includes a nominal 30 gpm of mini-flow.

The test procedure places requirements on instrument accuracy (20 inches of water column for the ECCS charging branch lines and 10 inches of water column for the SI branch lines) and setting tolerance (30 inches of water column for both the ECCS charging and SI branch lines) such that branch line flow imbalance remains within the assumptions of the safety analyses.

The maximum and minimum potential pump performance curves, in conjunction with the maximum and minimum flow SRs, the maximum total system resistance, and the test procedure requirements, ensure that the assumptions of the safety analyses remain valid.

The surveillance flow and differential pressure requirements are the Safety Analysis Limits and do not include instrument uncertainties. These instrument uncertainties will be accounted for in the surveillance test procedure to assure that the Safety Analysis Limits are met.

16.5.3 ECCS SUBSYSTEMS - MODE 4 ENTRY

16.5.3.1 LIMITING CONDITION FOR OPERATION

The reactor shall be subcritical for a minimum of 4 hours prior to entering MODE 4 from MODE 3.

APPLICABILITY: MODE 3 with RCS pressure $\leq$ 1000 psig and ECCS accumulators isolated, MODE 4.

ACTION:

With the reactor subcritical for less than the LCO time limit, do not enter MODE 4 from MODE 3. The unit shutdown exception and other provisions of Section 16.0.1.4 are not applicable for MODE 4 entry.

16.5.3.1.1 SURVEILLANCE REQUIREMENTS

The reactor shall be determined to have been subcritical per the LCO time limit prior to allowing MODE 4 entry from MODE 3. The unit shutdown exception and other provisions of Section 16.0.2.4 are not applicable for MODE 4 entry.

16.5.3.1.2 BASES

WCAP-12476, Revision 1, provides the results of a generic probabilistic risk assessment showing that a break with an equivalent diameter greater than 6-inches is not a credible MODE 4 scenario. That topical report also presents a generic bounding thermal-hydraulic analysis for the MODE 4 small break loss of coolant accident (SBLOCA) based on limiting representative plant parameters with the accumulators isolated. The assumed ECCS availability is based on one OPERABLE ECCS train consisting of a centrifugal charging subsystem and an RHR subsystem.

The generic thermal-hydraulic analysis for the limiting MODE 4 SBLOCA in WCAP-12476, Revision 1, is supplemented by a plant-specific evaluation (Westinghouse letter SCP-10-31 dated May 11, 2010) which demonstrates that the minimum safeguards ECCS flow from one centrifugal charging pump (CCP) and one RHR pump can satisfy the MODE 4 small break LOCA ECCS flow requirements given in Table 4-7 of that topical report provided that:

1) ECCS flow from one centrifugal charging subsystem can be established within 10 minutes of recognition of the event;

2) Flow from one RHR subsystem into two or more cold leg injection nozzles (i.e., RHR cross-tie valves EJHV8716A/B either open or closed) can be established within 30 minutes of recognition of the event; and

3) Decay heat loads are not exceeded at MODE 4 entry.

The last item, a four hour minimum subcritical decay heat time, is required to maintain complicance with WCAP-12476, Revision 1, and Westinghouse letter SCP-10-31 dated May 11, 2010, both of which assumed a 50°F/hour cooldown rate from 550°F prior to MODE 4 entry. It is possible to perform a cycle-specific post-shutdown analysis that demonstrates acceptable decay heat loads for cooldowns to MODE 4 that take less than 4 hours; however, the LCO statement in Section 16.5.3.1 provides a conservative time limit that is readily verifiable.

Technical Specifications 3.4.12 and 3.5.1 contain limitations on accumulator isolation. Additional documents incorporated by reference include CAR 201000601, CAR 201004572, and CAR 201004819.

16.6 CONTAINMENT SYSTEMS

16.6.1 PRIMARY CONTAINMENT

16.6.1.1 CONTAINMENT LEAKAGE LIMITING CONDITION FOR OPERATION

Containment leakage rates shall be limited in accordance with Technical Specification 3.6.1 and 5.5.16.

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

ACTION:

With the as left overall integrated containment leakage rate exceeding 0.75 L_a, restore the overall integrated leakage rate to less than 0.75 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

16.6.1.1.1 SURVEILLANCE REQUIREMENTS

The containment leakage rates shall be demonstrated per Technical Specification SR 3.6.1.1.

Leakage rate testing shall be in conformance with the criteria specified in Appendix J of 10 CFR Part 50.

a. Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted in accordance with 10CFR50, Appendix J, Option B during shutdown at a pressure not less than P_a, 48.1 psig, during each 10-year service period.

b. If any periodic as found Type A test fails to meet L_a, then a determination will be performed to identify the cause of unacceptable performance and determine appropriate corrective action. Once this is complete, acceptable performance is reestablished by performing a successful Type A test within 48 months following the failed Type A test. The as left overall integrated containment leakage rate shall be less than 0.75 L_a;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

    1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a.

    2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test, and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 48.1 psig, in accordance with 10CFR50, Appendix J, Option B except for tests involving:

1) deleted

2) Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Technical Specification SR 3.6.2.1 and SR 3.6.2.2.

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Technical Specification SR 3.6.3.6 and SR 3.6.3.7, as applicable; and

g. The provisions of Section 16.0.2.2 are not applicable.

16.6.1.1.2 BASES

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

The following exemptions have been granted to the requirements of Appendix J of 10 CFR Part 50:

1. Section III.A.1(a) - an exemption to the requirement to stop the Type A test if excessive leakage is determined. This exemption allows the satisfactory completion of the Type A test if the leakage can be isolated and appropriately factored into the results.

2. Section III.A.5(b) - an exemption for the acceptance criteria, in lieu of the present single criterion of the total measured containment leakage rate being less than

0.75 of the maximum allowable leakage rate, L_a, the "as found" allowable leakage rate will be L_a and the "as left" allowable leakage rate will be less than 0.75 L_a.

3. Section III.D.1(a) - an exemption that removes the requirement that the third test of each set of three Type A tests be conducted when the plant is shutdown for the 10-year plant inservice inspection.

Exemption 1 allows the continuance of a Type A test when excessive leakage is found provided that significant leaks are identified and isolated. After completion of the modified Type A test (i.e., a Type A test with the significant leakage paths isolated during the test), local leakage rates of those paths isolated during the modified Type A test will be measured before and after repairs to those paths. The adjusted "as found" leakage rate for the Type A test can be determined by adding the local leakage rates, measured before any repairs to those previously isolated leakage paths, to the containment integrated leakage determined in the modified Type A test, plus any leakage improvements (defined below) made prior to the test. This adjusted "as found" leakage rate is to be used in determining the scheduling of the periodic Type A test in accordance with Section III.A.6 of Appendix J.

The acceptability of the modified Type A test can be determined by calculating the adjusted "as left" containment overall integrated leakage rate and comparing it to the acceptance criteria of 0.75 L_a . The adjusted "as left" Type A leakage rate is determined by adding the local leakage rates, measured after any repairs and/or adjustments to those previously isolated leakage paths, to the leakage rate determined in the modified Type A test. It should be noted that additional adjustments for non-standard lineup and changes in containment volume are added to the measured leakage rate for both "as found" and "as left" determinations.

Leakage improvements are defined as the difference between the pre-repair LLRT and post-repair LLRT done on containment penetrations prior to the start of the Type A test.

The only differences between this approach and Appendix J requirements are that: (1) the potentially excessive leakage paths will be repaired and/or adjusted after the Type A test is completed; and (2) the Type A test leakage rate is partially determined by calculation rather than by direct measurement.

An OPERABLE containment requires the leakage rates to be determined per SR 3.6.1.1. The Section 16.6.1.1 action statement applies to testing prior to exceeding 200ºF. Failure to perform a surveillance within the specified time interval constitutes a failure to meet the FUNCTIONALITY requirements for the Section 16.6.1.1 LCO. This results in the OPERABILITY requirements for T/S 3.6.1 not being met. Therefore, the action statement for T/S 3.6.1 is applicable.

The provisions of Technical Specification SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Program. The provisions of Technical Specification SR 3.0.3 are applicable to the Containment Leakage Rate Program.

16.6.1.2 DELETED

16.6.3 CONTAINMENT ISOLATION VALVES

16.6.3.1 LIMITING CONDITION FOR OPERATION

See Technical Specification 3.6.3.

16.6.3.1.1 SURVEILLANCE REQUIREMENTS

See Technical Specification SR 3.6.3.1 through SR 3.6.3.8. The containment isolation valves are listed in Table 16.6-1.

16.6.3.1.2 BASES

See Technical Specification 3.6.3 Bases. For each containment isolation valve listed in Table 16.6-1, only the valve identification number, associated containment penetration, function, and type of App. J test are identified per the Inservice Testing program for Callaway.

Technical Specification LCO requirements and Surveillance Requirements apply to these valves. Sections 16.0.1 and 16.0.2 do not apply.

TABLE 16.6-1 CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
1. Phase "A" Isolation (active)			
P-62	BB HV-8026	PRT Nitrogen Iso Valve	C
P-62	BB HV-8027	PRT Nitrogen Iso Valve	C
P-24	BG HV-8100	RCP Seal Water Return CTMT Iso Valve	C
P-24	BG HV-8112	RCP Seal Water Return CTMT Iso Valve	C
P-23	BG HV-8152	Letdown System CTMT Iso Valve	C
P-23	BG HV-8160	Letdown System CTMT Iso Valve	C
P-25	BL HV-8047	Reactor Makeup Water CTMT Iso Valve	C
P-21	EJ HCV-8825	RHR to SI Test Line Iso Valve	A
P-82	EJ HCV-8890A	RHR A to SI Pumps Test Line Iso Valve	A
P-27	EJ HCV-8890B	RHR B to SI Pumps Test Line Iso Valve	A
P-49	EM HV-8823	SI/Accumulator Injection Test Line Iso Valve	A
P-48	EM HV-8824	Safety Injection Pump B Test Line Iso Valve	A
P-88	EM HV-8843	Boron injection Header Test Line Iso	A
P-92	EM HV-8871	SI Test Line to RWST Inside CTMT Iso	C
P-87	EM HV-8881	Safety Injection Pump Test Line Iso Valve	A
P-92	EM HV-8964	SI Test Line to RWST Outside CTMT Iso	C
P-99	GS HV-3*	Hydrogen Analyzer B Inlet Iso	A,C
P-99	GS HV-4*	Hydrogen Analyzer B Inlet Iso	A,C

TABLE 16.6-1 (Sheet 2)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
1. Phase "A" Isolation (active) - Continued			
P-99	GS HV-5*	Hydrogen Analyzer B Inlet Iso	A,C
P-56	GS HV-8*	Hydrogen Analyzer B Disch Iso	A,C
P-56	GS HV-9*	Hydrogen Analyzer B Disch Iso	A,C
P-101	GS HV-12*	Hydrogen Analyzer A Inlet Iso	A,C
P-101	GS HV-13*	Hydrogen Analyzer A Inlet Iso	A,C
P-101	GS HV-14*	Hydrogen Analyzer A Inlet Iso	A,C
P-97	GS HV-17*	Hydrogen Analyzer A Disch Iso	A,C
P-97	GS HV-18*	Hydrogen Analyzer A Disch Iso	A,C
P-101	GS HV-31	Sample Line to CTMT Atmos Monitor	A,C
P-101	GS HV-32	Sample Line to CTMT Atmos Monitor	A,C
P-97	GS HV-33	Sample Return From CTMT Atmos. Monitor	A,C
P-97	GS HV-34	Sample Return From CTMT Atmos. Monitor	A,C
P-99	GS HV-36	Sample Line to CTMT Atmos Monitor	A,C
P-99	GS HV-37	Sample Line to CTMT Atmos Monitor	A,C
P-56	GS HV-38	Sample Return from CTMT Atmos Monitor	A,C
P-56	GS HV-39	Sample Return from CTMT Atmos Monitor	A,C
P-44	HB HV-7126	RCDT Vent Inside CTMT Iso.	C
P-26	HB HV-7136	RCDT Pumps Disch Hdr Outside CTMT Iso	C
P-44	HB HV-7150	RCDT Vent Outside CTMT Iso.	C

TABLE 16.6-1 (Sheet 3)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
1. Phase "A" Isolation (active) - Continued			
P-26	HB HV-7176	RCDT Pumps Disch Hdr Inside CTMT Iso	C
P-30	KA FV-29	Reactor Bldg. Instr Air Supply Outside CTMT Iso	C
P-32	LF FV-95	CTMT Normal Sumps to Floor Drain Tank Inside CTMT Iso	C
P-32	LF FV-96	CTMT Normal Sumps to Floor Drain Tank Outside CTMT Iso	C
P-93	SJ HV-5	RCS Liquid Sample Inner CTMT Iso	C
P-93	SJ HV-6	RCS Liquid Sample Outer CTMT Iso	C
P-69	SJ HV-12	PZR Vapor Sample Inner CTMT Iso	C
P-69	SJ HV-13	PZR Vapor Sample Outer CTMT Iso	C
P-95	SJ HV-18	Accumulator Sample Inner CTMT Iso	C
P-95	SJ HV-19	Accumulator Sample Outer CTMT Iso	C
P-93	SJ HV-127	RCS Liquid Sample Outer CTMT ISO	C
P-64	SJ HV-128	PZR/RCS Liquid Sample Inner CTMT Iso	A,C
P-64	SJ HV-129	PZR/RCS Liquid Sample Outer CTMT Iso	A,C
P-64	SJ HV-130	PZR/RCS Liquid Sample Outer CTMT Iso Valve	A,C
P-57	Removed	Capped Penetration	A
P-57	Removed	Capped Penetration	A
2. Phase "A" Isolation (passive)			
P-58	EM HV-8888	Accumulator Tank Fill Line Iso Valve	C

TABLE 16.6-1 (Sheet 4)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
2. Phase "A" Isolation (passive) - Continued			
P-16	EN HV-01	CTMT Recirc Sump to CTMT Spray Pump A Iso	A
P-13	EN HV-07	CTMT Recirc Sump to CTMT Spray Pump B Iso	A
P-45	EP HV-8880	Accumulator Nitrogen Supply Iso Valve	C
P-65	GS HV-20	Hydrogen Purge Inner CTMT Iso	C
P-65	GS HV-21	Hydrogen Purge Outer CTMT Iso	C
P-67	KC HV-253	Fire Protection System Hdr Outer CTMT Iso	C
3. Phase "B" Isolation (active)			
P-74	EG HV-58	CCW Supply to RCPs Iso	C
P-75	EG HV-59	CCW Return From RCPs Iso	C
P-75	EG HV-60	CCW Return From RCPs Iso	C
P-76	EG HV-61	CCW Return From RCPs Thermal Barrier Iso.	C
P-76	EG HV-62	CCW Return From RCPs Thermal barrier Iso.	C
4. Containment Purge Isolation (active)			
V-161	GT HZ-4	CTMT Mini-Purge Supply Outside CTMT Iso	C
V-161	GT HZ-5	CTMT Mini-Purge Supply Inside CTMT Iso	C
V-160	GT HZ-11	CTMT Mini-Purge Exh Inside CTMT Iso	C
V-160	GT HZ-12	CTMT Mini-Purge Exh Outside CTMT Iso	C

TABLE 16.6-1 (Sheet 5)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
5. Containment Purge Isolation (passive)			
V-161	GT HZ-6	CTMT S/D Purge Supply Outside CTMT Iso	C
V-161	GT HZ-7	CTMT S/D Purge Supply Inside CTMT Iso	C
V-160	GT HZ-8	CTMT S/D Purge Exh Inside CTMT Iso	C
V-160	GT HZ-9	CTMT S/D Purge Exh Outside CTMT Iso	C
6. Remote Manual			
P-41	BB HV-8351A	RCP A Seal Water Supply	C
P-22	BB HV-8351B	RCP B Seal Water Supply	C
P-39	BB HV-8351C	RCP C Seal Water Supply	C
P-40	BB HV-8351D	RCP D Seal Water Supply	C
P-15	Removed	Capped Penetration	N/A
P-15	Removed	Capped Penetration	N/A
P-14	Removed	Capped Penetration	N/A
P-14	Removed	Capped Penetration	N/A
P-71	EF HV-31	ESW Supply to Containment Coolers	C
P-28	EF HV-32	ESW Supply to Containment Coolers	C
P-71	EF HV-33	ESW Supply to Containment Coolers	C
P-28	EF HV-34	ESW Supply to Containment Coolers	C

TABLE 16.6-1 (Sheet 6)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
6. Remote Manual - Continued			
P-73	EF HV-45	ESW Return From Containment Coolers	C
P-29	EF HV-46	ESW Return From Containment Coolers	C
P-73	EF HV-47	ESW Return From Containment Coolers	C
P-29	EF HV-48	ESW Return From Containment Coolers	C
P-73	EF HV-49	ESW Return From Containment Coolers	C
P-29	EF HV-50	ESW Return From Containment Coolers	C
P-74	EG HV-127	CCW Supply to RCPs	C
P-75	EG HV-130	CCW Return from RCPs	C
P-75	EG HV-131	CCW Return from RCPs	C
P-76	EG HV-132	CCW Return from RCP Thermal Barriers	C
P-76	EG HV-133	CCW Return from RCP Thermal Barriers	C
P-79	EJ HV-8701A	RCS Hot Leg 1 to RHR Pump A Suction	A
P-52	EJ HV-8701B	RCS Hot Leg 4 to RHR Pump B Suction	A
P-82	EJ HV-8809A	RHR Pump A Cold Leg Injection Iso Valve	A
P-27	EJ HV-8809B	RHR Pump B Cold Leg Injection Iso Valve	A
P-15	EJ HV-8811A	CTMT Recirc Sump to RHR Pump A Suction	A

TABLE 16.6-1 (Sheet 7)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
6. Remote Manual - Continued			
P-14	EJ HV-8811B	CTMT Recirc Sump to RHR Pump B Suction	A
P-21	EJ HV-8840	RCS Hot Leg Recirc Iso Valve	A
P-87	EM HV-8802A	SI Pump A Disch Hot Leg Iso Valve	A
P-48	EM HV-8802B	SI Pump B Disch Hot Leg Iso Valve	A
P-49	EM HV-8835	SI Pumps Disch to Cold Legs Iso Valve	A
P-89	EN HV-6	CTMT Spray Pump A Discharge Iso Valve	A
P-66	EN HV-12	CTMT Spray Pump B Discharge Iso Valve	A
7. Active for SIS			
P-80	BG HV-8105	CVCS Charging Line	C
P-88	EM HV-8801A	Boron Injection Header to RCS Cold Legs	A
P-88	EM HV-8801B	Boron Injection Header to RCS Cold Legs	A
8. Hand-Operated and Check Valves			
P-41	BB V-118	RCP A Seal Water Supply	C
P-22	BB V-148	RCP B Seal Water Supply	C
P-39	BB V-178	RCP C Seal Water Supply	C
P-40	BB V-208	RCP D Seal Water Supply	C
P-24	BG V-135	RCP Seal Water Return	C
P-80	BG 8381	CVCS Charging Line	C
P-25	BL 8046	Reactor Makeup Water Supply	C
P-78	BM V-045	Steam Generator Drain Line Iso Valve	C

TABLE 16.6-1 (Sheet 8)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
8. Hand-Operated and Check Valves - Continued			
P-78	BM V-046	Steam Generator Drain Line Iso Valve	C
P-53	EC V-083	Refueling Pool Supply From Fuel Pool Cleanup	C
P-53	EC V-084	Refueling Pool Supply From Fuel Pool Cleanup	C
P-54	EC V-087	Refueling Pool Return to Fuel Pool Cooling	C
P-54	EC V-088	Refueling Pool Return to Fuel Pool Cooling	C
P-55	EC V-095	Refueling Pool Skimmers to Fuel Pool Cooling Loop	C
P-55	EC V-096	Refueling Pool Skimmers to Fuel Pool Cooling Loop	C
P-74	EG V-204	CCW Supply to RCPs	C
P-82	EP 8818A	RHR Pump to Cold Leg 1 Injection	A
P-82	EP 8818B	RHR Pump to Cold Leg 2 Injection	A
P-27	EP 8818C	RHR Pump to Cold Leg 3 Injection	A
P-27	EP 8818D	RHR Pump to Cold Leg 4 Injection	A
P-21	EJ 8841A	RHR Pump Disch to RCS hot Leg 2	A
P-21	EJ 8841B	RHR Pump Disch to RCS Hot Leg 3	A
P-87	EM V-001	SI Pump Hot Leg 2 Injection	A
P-87	EM V-002	SI Pump Hot Leg 3 Injection	A
P-48	EM V-003	SI Pump Hot Leg 1 Injection	A
P-48	EM V-004	SI Pump Hot Leg 4 Injection	A
P-58	EM V-006	Accumulator Fill Line From SI Pumps	C
P-49	EP V-0010	SI Pumps Disch to Cold Leg 1	A

TABLE 16.6-1 (Sheet 9)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED
8. Hand-Operated and Check Valves - Continued			
P-49	EP V-0020	SI Pump Disch to Cold Leg 2	A
P-49	EP V-0030	SI Pump Disch to Cold Leg 3	A
P-49	EP V-0040	SI Pump Disch to Cold Leg 4	A
P-88	EM 8815	Boron Injection Header to RCS Cold Leg Injection	A
P-89	EN V-013	CTMT Spray Pump A to CTMT Spray Nozzles	A
P-66	EN V-017	CTMT Spray Pump B to CTMT Spray Nozzles	A
P-45	EP V-046	Accumulator Nitrogen Supply Line	C
P-43	HD V-016	Auxiliary Steam to Decon System	C
P-43	HD V-017	Auxiliary Steam to Decon System	C
P-63	KA V-039	Rx Bldg Service Air Supply	C
P-63	KA V-118	Rx Bldg Service Air Supply	C
P-30	KA V-204	Rx Bldg Instrument Air Supply	C
P-98	KB V-001	Breathing Air Supply to Rx Bldg.	C
P-98	KB V-002	Breathing Air Supply to Rx Bldg.	C
P-67	KC V-478	Fire Protection Supply to RX Bldg.	C
P-57	Removed	Capped Penetration	A
9. RHR Suction Relief Valves			
P-52	EJ 8708B	RHR Pump B Suction Relief	A
P-79	EJ 8708A	RHR Pump A Suction Relief	A

* Although identified as automatic (active) containment isolation valves, these normally closed valves are maintained closed and secured in the closed position (i.e., de-activated) during normal plant operation, except that the valves may be opened under administrative control in accordance with the plant Technical Specifications.

16.7 PLANT SYSTEMS

16.7.1 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

16.7.1.1 LIMITING CONDITION FOR OPERATION

The temperatures of both the reactor and secondary coolants in the steam generator shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the above requirements not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

16.7.1.1.1 SURVEILLANCE REQUIREMENTS

The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

16.7.1.1.2 BASES

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

16.7.2 DELETED

16.7.3 SEALED SOURCE CONTAMINATION

16.7.3.1 LIMITING CONDITION FOR OPERATION

Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma-emitting material or 5 microCuries of alpha-emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

    1) Decontaminate and repair the sealed source, or

    2) Dispose of the sealed source in accordance with Commission Regulations.

16.7.3.1.1 SURVEILLANCE REQUIREMENTS

16.7.3.1.1.a

Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

16.7.3.1.1.b

Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

    1) With a half-life greater than 30 days (excluding Hydrogen 3), and

    2) In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

16.7.3.1.1.c

Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

16.7.3.1.2 BASES

The limitation on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

16.7.4 AREA TEMPERATURE MONITORING

16.7.4.1 LIMITING CONDITION FOR OPERATION

The temperature limit of each area given in Table 16.7-2 shall not be exceeded.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 16.7-2, enter the Corrective Action Program to document the condition and determine whether or not a degraded or non-conforming condition exists. If equipment in the affected area(s) is degraded or non-conforming, identify and apply compensatory measures as necessary. This includes determining the FUNCTIONALITY/ OPERABILITY of equipment in the affected area(s), as applicable, by evaluating the effects of the out-of-limit temperature(s) on the equipment.

16.7.4.1.1 SURVEILLANCE REQUIREMENTS

The temperature in each of the areas shown in Table 16.7-2 shall be determined to be within its limit at least once per 12 hours.

16.7.4.1.2 BASES

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of the environmental qualification temperatures for the equipment. Exposure to excessive temperatures may degrade equipment and can cause a loss of its FUNCTIONALITY/OPERABILITY. The temperature limits include an allowance for instrument error of ±3°F, except for Electrical Penetration Rooms A and B. These rooms have an alarm at ≤ 103°F with a maximum room temperature of 106°F.

16.7.5 COMPONENT COOLING WATER (CCW) SYSTEM

16.7.5.1 LIMITING CONDITION FOR OPERATION

The CCW surge tank level instrumentation shall be FUNCTIONAL for each CCW train.

APPLICABILITY: MODES 1, 2, 3, and 4 except when the supply and return lines for the radwaste service loads are isolated by EGHV0069A/B and EGHV0070A/B.

ACTIONS:

With the CCW surge tank level instrumentation non-functional for one CCW train, restore CCW surge tank level instrumentation to FUNCTIONAL status with 72 hours; otherwise, enter Section 16.0.1.3.

16.7.5.1.1 SURVEILLANCE REQUIREMENTS

a. A CHANNEL OPERATIONAL TEST of the surge tank level instrumentation which provides automatic isolation of the non-nuclear safety-related portion of the system shall be performed at least once per 31 days.

b. A CHANNEL CALIBRATION of the surge tank level instrumentation which provides automatic isolation of the non-nuclear safety-related portion of the system shall be performed at least once per 18 months.

16.7.5.1.2 BASES

The CCW surge tank level instrumentation provides for automatic isolation of the non-safety related portion of the CCW system from the safety-related portion upon detection of a low water level in the surge tank. Specifically, in the event of a hazard that could compromise the integrity of the non-safety, non-seismic piping to the radwaste service loads, such as a seismically induced pipe break or crack or a non-mechanistic malfunction in the moderate energy piping, no safety injection signal would be generated for the automatic isolation of the non-safety related radwaste service loads (via closure of valves EGHV0069A/B and EGHV0070A/B). For such an event, the low-low level signal from the CCW surge tank level channels (initiated by transmitters EGLT0001 and EGLT0002) will isolate the radwaste service loads.

The CCW surge tank level channels driven by transmitters EGLT0001 and EGLT0002 shall be FUNCTIONAL except when the radwaste service loads are isolated. If the non-safety related radwaste service loads are isolated, at a minimum, by closing and de-activating one isolation valve in the supply line (either EGHV0069A or EGHV0070A) and by closing and de-activating one isolation valve in the return line (either EGHV0069B or EGHV0070B), FUNCTIONALITY of the CCW surge tank level channels is not required

per the Applicability. Closing and de-activating an isolation valve may be accomplished by closing the isolation valve and removing power from its associated solenoid valve.

Hazard mitigation functions do not satisfy any of the 10 CFR 50.36(c)(2)(ii) criteria for inclusion within the scope of the Technical Specifications. Therefore, FUNCTIONALITY of the CCW surge tank level channels is addressed in Chapter 16, and non-functionality of the CCW surge tank level channels does not affect the OPERABILITY of the CCW system with respect to the requirements of Technical Specification 3.7.7, “Component Cooling Water (CCW) System.”

A CHANNEL OPERATIONAL TEST of surge tank level instrumentation circuits that provide automatic isolation of the non-safety-related portion of the system assures that the channel components are capable of performing their functions upon the insertion of a simulated or actual actuation signal.

A CHANNEL CALIBRATION of surge tank level instrumentation circuits that provide automatic isolation of the non-safety-related portion of the system assures that the channels will respond within the required range and accuracy to known values of input.

16.7.6 ESSENTIAL SERVICE WATER (ESW) SYSTEM

16.7.6.1 LIMITING CONDITION FOR OPERATION

The ESW differential pressure instrumentation shall be FUNCTIONAL for each ESW train.

APPLICABILITY: MODES 1, 2, 3, and 4 for each ESW train except when that train's air compressor and aftercooler load is isolated by EFHV0043 ('A' train) or EFHV0044 ('B' train).

ACTIONS:

With the ESW differential pressure instrumentation non-functional for one ESW train, restore ESW differential pressure instrumentation to FUNCTIONAL status within 72 hours; otherwise, enter Section 16.0.1.3.

16.7.6.1.1 SURVEILLANCE REQUIREMENTS

a. A CHANNEL OPERATIONAL TEST of differential pressure instrumentation for automatic isolation of ESW to the air compressors shall be performed at least once per 31 days.

b. A CHANNEL CALIBRATION of differential pressure instrumentation for automatic isolation of ESW to the air compressors shall be performed at least once per 18 months.

16.7.6.1.2 BASES

Each ESW train services a non-safety related air compressor and associated aftercooler via non-safety related, non-seismic lines downstream of safety-related, air-operated isolation valves (EFHV0043 in 'A' train and EFHV0044 in 'B' train) as shown on M-22EF01 and M-22EF02. The ESW differential pressure channels provide for automatic isolation of the non-safety related piping from the safety-related portion of the system (for each train) upon detection of a high differential pressure (high flow) at the safety-related to non-safety related interface.

A potential hazard can be postulated to occur that could compromise the integrity of the non-safety, non-siesmic piping to the air compressors and aftercoolers, such as a seismically induced pipe break or crack or a non-mechanistic malfunction in the moderate energy piping. As discussed further below, depending on whether the hazard occurs with or without offsie power available, the non-safety piping would either be automatically isolated by signals from the ESW differential pressure channels (EFPDT0043 and EFPDSH0043 in 'A' train; EFPDT0044 and EFPSH0044 in 'B' train), or

by operator action for smaller leaks prior to impacting the available UHS inventory, or by a loss of air supply to the isolation valve solenoids.

Following a hazard that results in a load shed of the air compressors, isolation valves EFHV0043 and EFHV0044 will fail closed and the loss of ESW inventory via the pipe break or crack will be terminated. After the air compressor loads are shed from the safety-related 4.16 kV NB system buses after a safety injection signal or a loss of NB bus voltage, the isolation valve actuators will bleed off their air supply via the porting arrangement of their associated solenoid valves, and the isolation valves will close, thereby isolating flow.

Following a hazard that occurs with offsite power available and the air compressor electrical loads being supplied, the non-seismic air compressor aftercooler lines would continue to leak. If the magnitude of the nominal flow to the aftercooler plus the postulated hazard-induced flow were to exceed 100 gpm, the isolation valves would be automatically isolated by signals from the ESW differential pressure channels (EFPDT0043 and EFPDSH0043 in ‘A’ train; EFPDT0044 and EFPDSH0044 in ‘B’ train). Smaller leaks would be isolated by operator action prior to impacting the available UHS inventory.

The ESW differential pressure channel in each ESW train (driven by transmitters EFPDT0043 and EFPDT0044, respectively) shall be FUNCTIONAL except when the applicable train’s air compressor and aftercooler load is isolated. If the non-safety related air compressor and aftercooler load is isolated by closing and isolating EFHV0043 (in the ‘A’ train) or EFHV0044 (in the ‘B’ train), FUNCTIONALITY of the ESW differential pressure channel in that ESW train is not required per the Applicability. Closing and isolating the isolation valve may be accomplished by closing the isolation valve and removing power from its associated solenoid valve or by closing the isolation valve and closing a manual valve.

Hazard event mitigation functions do not satisfy any of the 10 CFR 50.36(c)(2)(ii) criteria for inclusion within the scope of the Technical Specifications. Therefore, FUNCTIONALITY of the ESW differential pressure channels is addressed in Chapter 16, and non-functionality of the ESW differential pressure channels does not affect the OPERABILITY of the ESW system with respect to the requirements of Technical Specification 3.7.8, “Essential Service Water (ESW) System.”

A CHANNEL OPERATIONAL TEST of differential pressure instrumentation channels for automatic isolation of ESW to air compressors assures that the channel components are capable of performing their functions upon the insertion of a simulated or actual actuation signal.

A CHANNEL CALIBRATION of differential pressure instrumentation channel for automatic isolation of ESW to the air compressors assures that the channels will respond within the required range and accuracy to known values of input.

16.7.7 ULTIMATE HEAT SINK (UHS)

16.7.7.1 LIMITING CONDITION FOR OPERATION

The UHS cooling tower fill materials for each train and the UHS riprap shall be FUNCTIONAL.

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

ACTIONS:

a. With the UHS riprap non-functional, immediately remove any abnormal degradation that might lead to blockage of the ESW pump suction.

b. With the UHS cooling tower fill materials non-functional for one train, restore the fill material to FUNCTIONAL status within 72 hours; otherwise, enter Section 16.0.1.3.

16.7.7.1.1 SURVEILLANCE REQUIREMENTS

a. At least once per 18 months, visually inspect and verify that there is no abnormal breakage or degradation of the UHS cooling tower fill materials.

b. At least once per 31 days, visually inspect the UHS riprap for an abnormal degradation which might lead to blockage of the ESW pump suction.

16.7.7.1.2 BASES

Verification of the integrity of the UHS cooling tower fill materials ensures the UHS cooling tower trains are available to perform their function. Inspection of UHS riprap ensures that no abnormal degradation has occurred which might lead to blockage of the ESW pump suction.

16.7.8 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS)

16.7.8.1 LIMITING CONDITION FOR OPERATION

Pressurization System flow rate shall be within limits during system operation CRVIS mode for each CREVS train.

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

ACTIONS:

a. With Pressurization System flow rate not within limits for one CREVS train in MODES 1, 2, 3 or 4, restore Pressurization System flow rate to within limits within 7 days.

b. With Pressurization System flow rate not within limits for one CREVS train during movement of irradiated fuel assemblies, restore Pressurization System flow rate to within limits within 7 days; otherwise, immediately place the other CREVS train in CRVIS mode, or immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

16.7.8.1.1 SURVEILLANCE REQUIREMENTS

At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, verify the system flow rate is 2200 cfm (+800/-400) for the Pressurization System.

16.7.8.1.2 BASES

Verification of the flow rate in the Pressurization System assures that the ventilation system flows are properly balanced within the system.

16.7.9 WATER LEVEL - FUEL STORAGE POOL

16.7.9.1 LIMITING CONDITION FOR OPERATION

The fuel storage pool water level shall be ≥ 23 ft over the top of the storage racks.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the fuel storage pool.

ACTION:

With the requirements of the LCO not satisfied, immediately suspend all crane operations with loads in the fuel storage areas.

16.7.9.1.1 SURVEILLANCE REQUIREMENTS

At least once per 7 days, the water level in the fuel storage pool shall be verified to be greater than or equal to 23 feet above the storage racks.

16.7.9.1.2 BASES

Restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

Restrictions on crane operation are not part of the assumptions used for the FHA. However, crane operations should be controlled consistent with analyzed heavy load movements. If crane operations are inconsistent with the analyzed heavy load movements, the impact could be adverse to fuel stored in the fuel storage pool.

16.7.10 EMERGENCY EXHAUST SYSTEM (EES) for CRANE OPERATION FUEL BUILDING

16.7.10.1 LIMITING CONDITION FOR OPERATION

Two EES trains shall be FUNCTIONAL.

NOTE: The fuel building boundary may be opened intermittently under administrative control.

APPLICABILITY:

During crane or hoist operation involving load over the spent fuel storage areas, with irradiated fuel assemblies in the fuel storage pool.

ACTIONS:

a. With one EES train non-functional, crane operation with loads over the fuel storage areas may proceed provided the FUNCTIONAL EES train is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no EES train FUNCTIONAL, immediately suspend all crane operation with loads over the fuel storage areas until at least one EES train is restored to FUNCTIONAL status.

16.7.10.1.1 SURVEILLANCE REQUIREMENTS

The above required EES train shall be demonstrated FUNCTIONAL:

a. At the frequency described in Technical Specification Surveillance Requirement 3.7.13.1 by operating each EES train for ≥ 15 continuous minutes with the heaters operating;

b. By performing required EES filter testing in accordance with the Ventilation Filter Testing Program (VFTP);

c. At the frequency described in Technical Specification Surveillance Requirement 3.7.13.3 by verifying each EES train actuates on an actual or simulated actuation signal;

d. At the frequency described in Technical Specification Surveillance Requirement 3.7.13.5 on a STAGGERED TEST BASIS, by verifying one EES train can maintain a negative pressure ≥ 0.25 inches water gauge with respect to atmospheric pressure in the fuel building during the FBVIS mode of operation.

16.7.10.1.2 BASES

The required actions provide additional assurance that damage to irradiated fuel in the fuel storage pool would not occur due to crane operations when the EES was degraded.

16.7.11 AREA 5 MISSILE SHIELDS

16.7.11.1 LIMITING CONDITION FOR OPERATION

Area 5 missile shields shall be installed and FUNCTIONAL.

APPLICABILITY:

Anytime MSIVs and MFIVs are required to be OPERABLE per TS 3.7.2 and TS 3.7.3, respectively.

ACTION:

a. Weather monitoring shall be demonstrated available prior to removing the missile shield(s).

b. With the concrete missile shields between the Turbine Building and Area 5 of the Auxiliary Building removed:

   1) Equipment and tools required to close the missile shield(s) shall be staged for immediate use prior to removing the missile shield(s).

   2) If thunderstorm watches or warnings, tornado watches or warnings, or high winds are within 105 miles of the plant moving toward the plant, the missile shield must be closed immediately.

   3) Sufficient personnel shall be available to facilitate immediate replacement of the shield(s) if required.

   4) Otherwise, immediately replace the missile shield or declare affected equipment in Area 5 inoperable, in accordance with TS 3.7.2 and TS 3.7.3.

16.7.11.1.1 SURVEILLANCE REQUIREMENTS

Not Applicable.

16.7.11.1.2 BASES

Weather monitoring and prompt closure of the missile shield(s) ensures OPERABILITY of the components located in Area 5 (Main Steam Lines and associated isolation valves, Main Feedwater Lines and associated isolation valves, and Auxiliary Feedwater Lines) should thunderstorm, tornado, or other severe weather conditions arise, creating the potential need for missile protection. Reference: RFR 19618, Rev. E.

16.7.12 <u>MAIN STEAM LINE VALVES ACTUATOR TRAINS</u>

16.7.12.1 “NOT USED”

16.7.12 MAIN STEAM LINE VALVES ACTUATOR TRAINS

16.7.12.2 DELETED

16.7.13 CLASS 1E ELECTRICAL EQUIPMENT AIR CONDITIONING (A/C)

16.7.13.1 SUPPLEMENTAL COOLING TRAINS LIMITING CONDITION FOR OPERATION

Two Class 1E Electrical Equipment A/C supplemental cooling trains shall be FUNCTIONAL.

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

ACTIONS:

With the above requirements not satisfied:

a. With one Class 1E Electrical Equipment A/C supplemental cooling train nonfunctional, restore the Class 1E A/C supplemental cooling train to FUNCTIONAL status within 30 days.

b. With two Class 1E Electrical Equipment A/C supplemental cooling trains nonfunctional, restore one Class 1E Electrical Equipment A/C supplemental cooling train to FUNCTIONAL status within 7 days.

c. With Action a or b not met, enter Section 16.0.1.3.

16.7.13.1.1 SURVEILLANCE REQUIREMENTS

a. Verify each Class 1E Electrical Equipment A/C supplemental cooling train is available at least once per 30 days.

b. Verify each Class 1E Electrical Equipment A/C supplemental cooling train actuates and provides recirculation air flow at least once per 18 months.

16.7.13.1.2 BASES

Technical Specification 3.7.20, “Class 1E Electrical Equipment Air Conditioning (A/C) System,” specifies requirements for the Class 1E electrical equipment air conditioning (A/C) system. The Class 1E electrical equipment A/C system provide a suitable environment for the Class 1E electrical equipment within the Control Building. The system consists of two independent trains such that each train is normally aligned to cool only the equipment associated with its emergency load group.

The Class 1E electrical equipment A/C trains are designed to provide temperature control for the Engineered Safety Features (ESF) switchgear room components, DC switchboard room components, and NK battery room components during normal and emergency conditions.

The specific rooms supplied by the Class 1E electrical equipment A/C trains are as follows:

SGK05A		**SGK05B**	
SWBD RM No. 1	(3408)	SWBD RM No. 4	(3404)
SWBD RM No. 3	(3414)	SWBD RM No. 2	(3410)
Battery RM No. 1	(3407)	Battery RM No. 4	(3405)
Battery RM No. 3	(3413)	Battery RM No. 2	(3411)
ESF SWGR RM No. 1	(3301)	ESF SWGR RM No. 2	(3302)

With one Class 1E electrical equipment A/C train inoperable, a Class 1E Electrical Equipment A/C supplemental cooling train can be employed to enable the remaining Class 1E electrical equipment A/C train to provide adequate area cooling for both trains of electrical equipment during normal and accident conditions. The Class 1E Electrical Equipment A/C supplemental cooling trains serve to promote the circulation of cool air from one Class 1E electrical equipment A/C train to the rooms/areas of both Class 1E electrical equipment trains. To initiate operation of a supplemental cooling train, manual operator action is required to open the motorized control dampers and start the recirculation fans using hand switches located at a panel on the 2000’ level of the Control Building.

Specifically, with one of the Class 1E electrical equipment A/C trains required by TS 3.7.20 inoperable, Required Action A.1 of TS 3.7.20 requires immediately initiating action to implement mitigating actions. The mitigating actions to be taken are to start the appropriate supplemental cooling train, consistent with the intended use of the train.

Surveillance 16.7.13.1.1a requires verifying the availability of each Class 1E electrical equipment A/C supplemental cooling train. This verification includes a check of breaker position and indication lights to ensure the equipment is available. The Surveillance interval of 30 days is reasonable based on the supplemental cooling train typically being in a standby condition.

Surveillance 16.7.13.1.1b requires verifying the proper actuation of each Class 1E electrical equipment A/C supplemental cooling train fan and associated dampers, including verifying that each train provides recirculation air flow. This involves starting each supplemental cooling train and verifying fan-running indication as well as correct damper positions and indication. The Surveillance interval of 18 months is consistent with similar surveillances based on the typical industry refueling cycle.

References: FSAR Section 9.4, License Amendment 219 to the Callaway Operating License.

16.7.13.2 DELETED

16.7.13.2.1 DELETED

16.7.13.2.2 DELETED

TABLE 16.7-1 DELETED

TABLE 16.7-2 AREA TEMPERATURE MONITORING

	AREA	MAXIMUM TEMPERATURE LIMIT (°F)
1.	ESW Pump Room A	119
2.	ESW Pump Room B	119
3.	Auxiliary Feedwater Pump Room A	119
4.	Auxiliary Feedwater Pump Room B	119
5.	Turbine-Driven Auxiliary Feedwater Pump Room(1, 2)	121
6.	ESF Switchgear Room I	87
7.	ESF Switchgear Room II	87
8.	Switchboard Room No. 1	87
9.	Switchboard Room No. 2	87
10.	Switchboard Room No. 3	87
11.	Switchboard Room No. 4	87
12.	RHR Pump Room A	119
13.	RHR Pump Room B	119
14.	CTMT Spray Pump Room A	119
15.	CTMT Spray Pump Room B	119
16.	Safety Injection Pump Room A	119
17.	Safety Injection Pump Room B	119
18.	ECCS Centrifugal Charging Pump Room A	119
19.	ECCS Centrifugal Charging Pump Room B	119
20.	Electrical Penetration Room A	106
21.	Electrical Penetration Room B	106
22.	Component Cooling Water Room A	119
23.	Component Cooling Water Room B	119

TABLE 16.7-2 (Sheet 2)

	AREA	MAXIMUM TEMPERATURE LIMIT (°F)
24.	Diesel Generator Room A	119
25.	Diesel Generator Room B	119
26.	Control Room(1)	84

NOTES:

1. Current temperature limits are maximum temperatures under normal operating conditions. Temperatures in the TDAFP Room may rise to 144.5°F during a Station Blackout (SBO) event and to 148.6°F during a Design Basis Accident (DBA) when the pump is operating coincident with a Loss of Offsite Power (analyzed in FSAR Section 15.2.6). The DBA temperature exceeds the SBO temperature. See Table 8.3A-1 for a discussion of the qualification of equipment located in room 1331. Control Room temperatures may rise to 104°F under Station Blackout conditions.

2. TDAFP Room temperature may rise to 123.8°F while the pump is running and normal HVAC is available to that room.

16.8 ELECTRICAL POWER SYSTEMS

16.8.1 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

16.8.1.1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES LIMITING CONDITION FOR OPERATION

All containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating shall be FUNCTIONAL.

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

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) non-functional:

a. Restore the protective device(s) to FUNCTIONAL status or deenergize the circuit(s) by tripping the associated backup circuit breaker, or racking out or removing the non-functional circuit breaker within 72 hours, declare the affected system or component inoperable/non-functional, and verify the backup circuit breaker to be tripped or the non-functional circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Section 16.0.1.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their non-functional circuit breakers racked out, or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

16.8.1.1.1 SURVEILLANCE REQUIREMENTS

All containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating shall be demonstrated FUNCTIONAL:

a. At least once per 18 months:

   1) By verifying that the 13.8 kV circuit breakers are FUNCTIONAL by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following:

      a) A CHANNEL CALIBRATION of the associated protective relays,

      b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and

associated circuit breakers and control circuits function as designed, and

c) For each circuit breaker found non-functional during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the non-functional type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal Setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found non-functional during functional testing shall be restored to FUNCTIONAL status prior to resuming operation. For each circuit breaker found non-functional during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the non-functional type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested within the initial surveillance frequency plus 25% extension.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

16.8.1.1.2 BASES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the FUNCTIONALITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

A list of containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating is maintained for the plant site. The

addition or deletion of any containment penetration conductor overcurrent protective device shall be made in accordance with plant procedures.

16.8.2 A.C. SOURCES-OPERATING

16.8.2.1 LIMITING CONDITION FOR OPERATION

Two diesel generators shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With the LCO not met, enter Section 16.0.1.3.

b. The diesel generators are not considered non-functional when the fuel oil storage tank missile shield is removed provided the following administrative controls are in place:

   1) Weather monitoring is in place prior to and during shield removal, and no thunderstorms are within 70 miles.

   2) Equipment, tools, and personnel required to close the missile shield shall be on site and located such that the shield can be closed in one hour or less.

   3) Only one Emergency Diesel Fuel Oil storage tank missile shield may be open at a time.

   4) If thunderstorm watches or warnings, tornado watches or warnings, or high winds are within 70 miles of the plant moving toward the plant, the missile shield must be closed immediately.

   5) Installation of hold down bolt nuts and washers are required for tornado missile protection.

   6) Loads are not hoisted over a given train when the missile shield is removed.

   7) Upon completion or stoppage of the activity requiring the shield open, immediately replace the missile shield.

16.8.2.1.1 SURVEILLANCE REQUIREMENTS

Each diesel generator shall be demonstrated FUNCTIONAL:

a. By performance of the following at the frequencies specified:

   1) By verifying that the auto-connected loads to each diesel generator do not exceed 6201 kW, at least once per 36 months.

   2) By verifying that the fuel transfer pump transfers fuel oil from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines, at least once per 18 months.

b. At least once per 10 years by:

   1) Draining each fuel oil storage tank,

   2) Removing the accumulated sediment,

   3) Cleaning the tank to remove microbiological growth.

16.8.2.1.2 BASES

See Technical Specification Bases 3.8.1 and 3.8.3 and FSAR Section 9.5.4.2.1 and Table 9.5.4-3, Item 2f.

In order for the diesel generator to remain FUNCTIONAL when the missile shield is removed, appropriate administrative controls are followed to ensure adequate missile protection. Reference: RFR 19618, Rev. G.

16.8.3 A.C. SOURCES-SHUTDOWN

16.8.3.1 LIMITING CONDITION FOR OPERATION

The following AC electrical power sources shall be FUNCTIONAL:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem; and

b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem; and

c. The Shutdown portion of one Load Shedder and Emergency Load Sequencer (LSELS) associated with the required DG and AC electrical power distribution train.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTION:

a. With less than the above minimum required A.C. electrical power sources FUNCTIONAL, immediately suspend all operations involving crane operation with loads over the spent fuel pool.

b. The diesel generators are not considered non-functional when the fuel oil storage tank missile shield is removed provided the following Administrative Controls are in place:

   1) Weather monitoring is in place prior to and during shield removal, and no thunderstorms are within 70 miles.

   2) Equipment, tools, and personnel required to close the missile shield shall be on site and located such that the shield can be closed in one hour or less.

   3) Only one Emergency Diesel Fuel Oil storage tank missile shield may be open at a time.

   4) If thunderstorm watches or warnings, tornado watches or warnings, or high winds are within 70 miles of the plant moving toward the plant, the missile shield must be closed immediately.

   5) Installation of hold down bolt nuts and washers are required for tornado missile protection.

6) Loads are not hoisted over a given train when the missile shield is removed.

7) Upon completion or stoppage of the activity requiring the shield open, immediately replace the missile shield.

c. In Mode 6, with the emergency diesel generator required per TS 3.8.2 inoperable, enter Condition B of TS 3.8.2 as well as Required Action A.4 of TS 3.9.5 and Required Action B.3 of TS 3.9.6 (for closing all direct access containment penetrations within 4 hours).

16.8.3.1.1 SURVEILLANCE REQUIREMENTS

See Technical Specification SR 3.8.2.1.

16.8.3.1.2 BASES

See FSAR Section 9.1.4

In order for the diesel generator to remain FUNCTIONAL when the missile cover is removed, appropriate administrative controls are followed to ensure adequate missile protection. Reference: RFR 19618, Rev. G.

16.9 REFUELING OPERATIONS

16.9.1 COMMUNICATIONS

16.9.1.1 LIMITING CONDITION FOR OPERATION

Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

16.9.1.1.1 SURVEILLANCE REQUIREMENTS

Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

16.9.1.1.2 BASES

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

16.9.2 REFUELING MACHINE

16.9.2.1 LIMITING CONDITION FOR OPERATION

The refueling machine shall be used for movement of drive rods or fuel assemblies and shall be FUNCTIONAL with:

a. The refueling machine used for movement of fuel assemblies having:

   1) A minimum capacity of 4000 pounds,

   2) Automatic overload trip setpoint less than or equal to 200 pounds above the indicated suspended weight for the heaviest fuel assembly/insert combination being moved.

   3) An automatic load reduction trip with a Setpoint of less than or equal to 200 pounds below the suspended weight for the lightest fuel assembly/insert combination being moved.

b. The auxiliary hoist used for latching and unlatching drive rods and thimble plug handling operations having:

   1) A minimum capacity of 3000 pounds, and

   2) A 1000-pound minimum capacity load indicator which shall be used to monitor lifting loads for these operations.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for refueling machine and/or auxiliary hoist FUNCTIONALITY not satisfied, suspend non-functional refueling machine crane and/or auxiliary hoist operations involving the movement of drive rods and fuel assemblies within the reactor vessel. This required action does not preclude moving a suspended fuel assembly or drive rod to a safe location, with administrative controls in place to monitor and ensure hoist load setpoints are not exceeded.

16.9.2.1.1 SURVEILLANCE REQUIREMENTS

16.9.2.1.1.a

The refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated FUNCTIONAL within a window of time that begins 100 hours prior

to removing the reactor vessel head and ends when the refueling machine is first used to move fuel assemblies by performing a load test of at least 125% of the automatic overload cutoff and demonstrating an automatic load cutoff when the refueling machine load exceeds the Setpoint of Section 16.9.2.1.a.2) and by demonstrating an automatic load reduction trip when the load reduction exceeds the Setpoint of Section 16.9.2.1.a.3).

16.9.2.1.1.b

Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated FUNCTIONALwithin a window of time that begins 100 hours prior to removing the reactor vessel head and ends when the auxiliary hoist is first used to manipulate drive rods by performing a load test of at least 1250 pounds.

16.9.2.1.2 BASES

The FUNCTIONALITY requirements for the refueling machine and auxiliary hoist ensure that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

16.9.3 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

16.9.3.1 LIMITING CONDITION FOR OPERATION

Loads in excess of 2250 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage facility, except for the spent fuel pool transfer gates which may be moved over fuel assemblies in the spent fuel pool for refueling activities, fuel handling system maintenance, and transfer gate seal replacement.

APPLICABILITY: With fuel assemblies in the spent fuel storage facility.

ACTION:

With the above requirements not satisfied, place the crane load in a safe condition.

16.9.3.1.1 SURVEILLANCE REQUIREMENTS

Crane interlocks and physical stops which prevent crane travel with loads in excess of 2250 pounds over fuel assemblies shall be demonstrated FUNCTIONAL within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

16.9.3.1.2 BASES

The spent fuel storage facility is located within the fuel building and is also referred to as the fuel storage pool. The fuel storage pool consists of the spent fuel pool and cask loading pool (with racks installed).

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool areas ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

The spent fuel pool transfer gates are excluded from this restriction because with a limited gate lift height, the spent fuel pool racks will absorb the impact of a dropped gate without damage to fuel assemblies. In addition, redundant trolleys and supports are used when moving the gates to preclude dropping a gate on the spent fuel racks, the time and distance the gates are moved over fuel is minimized as much as practical, and gate travel over fuel assemblies containing rod cluster control assemblies (RCCAs) is prohibited. The spent fuel pool transfer gates are only moved for refueling activities, fuel handling system maintenance, and to change gate seals.

16.9.4 WATER LEVEL - REACTOR VESSEL

16.9.4.1 CONTROL RODS LIMITING CONDITION FOR OPERATION

At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

APPLICABILITY:

During movement of control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the above requirements not satisfied, suspend all operations involving movement of control rods within the pressure vessel.

16.9.4.1.1 SURVEILLANCE REQUIREMENTS

The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of control rods within the reactor vessel.

16.9.4.1.2 BASES

See Technical Specification Bases 3.9.7.

16.9.5 DECAY TIME

16.9.5.1 LIMITING CONDITION FOR OPERATION

The reactor shall be subcritical for at least 72 hours.

APPLICABIITY:

During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

16.9.5.1.1 SURVEILLANCE REQUIREMENTS

The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

16.9.5.1.2 BASES

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the fuel handling accident radiological consequence and spent fuel thermal-hydraulic analyses.

16.10 SPECIAL TEST EXCEPTIONS

16.10.1 POSITION INDICATION SYSTEM - SHUTDOWN

16.10.1.1 LIMITING CONDITION FOR OPERATION

The limitations of Section 16.1.3.1 may be suspended during the performance of shutdown and control rod drop time measurements provided :

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, or

b. RCS Boron Concentration is equal to or greater than boron concentration for K_{eff} = 0.99 plus 50 ppm with all shutdown and control banks fully withdrawn.

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements and during surveillance of digital rod position indicators for OPERABILITY.

ACTION:

With the Position Indication System non-functional and with boron concentration less than K_{eff} = 0.99 plus 50 ppm and with more than one bank of rods withdrawn, immediately insert all rods fully and place the Rod Control System in a condition incapable of rod withdrawal.

16.10.1.1.1 SURVEILLANCE REQUIREMENTS

16.10.1.1.1.a

The above required Position Indication System shall be determined to be FUNCTIONAL within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

16.10.1.1.1.b

RCS boron concentration will be determined to be greater than concentration required for K_{eff} = 0.99 plus 50 ppm prior to the start of rod drop time measurements.

16.10.1.1.2 BASES

This special test exception permits the Position Indication Systems to be non-functional during rod drop time measurements. If more than one bank is withdrawn, a boron concentration greater than the boron concentration for K_{eff} = 0.99 plus 50 ppm with all banks fully withdrawn will ensure that the reactor remains in Mode 3 and that shutdown margin is adequate.

16.11 OFFSITE DOSE CALCULATION MANUAL
(ODCM 9.0) RADIOACTIVE EFFLUENT CONTROLS

16.11.1 LIQUID EFFLUENT

16.11.1.1 LIQUID EFFLUENTS CONCENTRATION LIMITING CONDITION FOR OPERATION

(ODCM 9.3.1)

The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 16.11-1) shall be limited to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

16.11.1.1.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.3.2)

16.11.1.1.1.a

Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 16.11-1.

16.11.1.1.1.b

The results of the radioactivity analysis shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Section 16.11.1.1.

16.11.1.1.2 BASES

This section is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than 10 times the concentration in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and

(2) the limits of 10 CFR Part 20.1301 to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLD's).

16.11.1.2 DOSE FROM LIQUID EFFLUENTS LIMITING CONDITION FOR OPERATION

(ODCM 9.4.1)

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 16.11-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include: (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Clean Drinking Water Act.*

16.11.1.2.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.4.2)

Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

16.11.1.2.2 BASES

This section is provided to implement the requirements of Sections II.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required

* The requirements of ACTION a.(1) and (2) are applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river-sited plants this is 3 miles downstream only.

operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable".

Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I which specify that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculations of Annual Doses to Man from Routine Releases of Reactor Effluents with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic and Dispersion of Effluents from accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", April 1977.

The reporting requirements of Action(a) implement the requirements of 10CFR20.2203.

16.11.1.3 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION LIMITING CONDITION FOR OPERATION

(ODCM 9.1.1)

The radioactive liquid effluent monitoring instrumentation channels shown in Table 16.11-2 shall be FUNCTIONAL with their Alarm/Trip Setpoints set to ensure that the limits of Section 16.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel non-functional.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels FUNCTIONAL, take the ACTION shown in Table 16.11-2. Restore the non-functional instrumentation to FUNCTIONAL status within 30 days and, if unsuccessful, explain in the next Radioactive Effluent Release Report, pursuant to Technical Specification 5.6.3, why this non-functionality was not corrected within the time specified.

16.11.1.3.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.1.2)

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated FUNCTIONAL by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL OPERATIONAL TEST at the frequencies shown in Table 16.11-3.

16.11.1.3.2 BASES

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The FUNCTIONALITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

16.11.1.4 LIQUID RADWASTE TREATMENT SYSTEM LIMITING CONDITION FOR OPERATION

(ODCM 9.5.1)

The Liquid Radwaste Treatment System shall be FUNCTIONAL and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 16.11-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

With radioactive liquid waste being discharged in excess of the above limits and the Liquid Radwaste Treatment Systems are not being fully utilized, prepare and submit to the Commission within 30 days a Special Report that includes the following information:

1) Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability.

2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and

3) Summary description of action(s) taken to prevent a recurrence.

16.11.1.4.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.5.2)

16.11.1.4.1.a

Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

16.11.1.4.1.b

The installed Liquid Radwaste Treatment System shall be considered FUNCTIONAL by meeting Sections 16.11.1.1 and 16.11.1.2.

16.11.1.4.2 BASES

The FUNCTIONALITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to

release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This section implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

16.11.1.5 LIQUID HOLDUP TANKS
LIMITING CONDITION FOR OPERATION

The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 150 Curies, excluding tritium and dissolved or entrained noble gases:

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.

APPLICABILITY: At all times.

ACTION:

With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Radioactive Effluent Release Report, pursuant to Technical Specification 5.6.3.

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16.11.1.5.1 SURVEILLANCE REQUIREMENTS

(4.11.1.4)

The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank. The provisions of Sections 16.0.2.2 and 16.0.2.3 are applicable, however the allowed surveillance interval extension beyond 25% shall not be exceeded. These tanks are also covered by Administrative Controls Section 5.5.12 of the plant Technical Specifications.

16.11.1.5.2 BASES

The tanks listed above include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20.1-20.602, Appendix B, Table II, Column 2, (redesignated at 56FR23391, May 21, 1991) at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

16.11.2 GASEOUS EFFLUENTS

16.11.2.1 GASEOUS EFFLUENTS DOSE RATE LIMITING CONDITION OF OPERATION

(ODCM 9.6.1)

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 16.11-2) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131 and 133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

16.11.2.1.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.6.2)

16.11.2.1.1.a

The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

16.11.2.1.1.b

The dose rate due to Iodine-131 and 133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 16.11-4.

16.11.2.1.2 BASES

This section is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The dose rate limits are the

doses associated with the concentrations of 10 CFR Part 20.1-20.601, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the dose limits specified in 10 CFR Part 20 10 CFR 20.1301. For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLD's).

The requirement for additional sampling of the Unit Vent following a reactor power transient is provided to ensure that the licensee is aware of and properly accounts for any increases in the release of gaseous effluents due to spiking which may occur as a result of the power transient. Monitoring the Unit Vent for increased noble gas activity is appropriate because it is the release point for any increased activity which may result from the power transient.

Since the escape rate coefficients for the noble gas nuclides is equal to or greater than the escape rate coefficient for iodine and the particulate nuclides[*,**] , it is reasonable to assume that the RCS spiking behavior of the noble gas nuclides is similar to that of the particulate and iodine nuclides. Considering the effects of iodine and particulate partitioning, plateout on plant and ventilation system surfaces, and the 99% efficiency of the Unit Vent HEPA filters and charcoal absorbers, it is reasonable to assume that the relative concentrations of the noble gas nuclides will be much greater than those of the iodine and particulate nuclides. Therefore, an increase in the iodine and particulate RCS activity is not an appropriate indicator of an increase in the Unit Vent activity, and it is appropriate to monitor the Unit Vent effluent activity as opposed to the RCS activity as an indicator of the need to perform post-transient sampling. In addition, it is appropriate to monitor the noble gas activity due to its relatively greater concentration in the Unit Vent.

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* Cohen, Paul, Water Coolant Technology of Power Reactors, Table 5.19, page 198. American Nuclear Society. 1980.

** NUREG-0772, “Technical Bases for Estimating Fission Product Behavior During LWR Accidents”, Silberberg, M., editor, USNRC; Figure 4.3, page 4.22. June, 1981.

16.11.2.2 DOSE - NOBLE GASES LIMITING CONDITION OF OPERATION

(ODCM 9.7.1)

The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 16.11-2) shall be limited to the following:

During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

16.11.2.2.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.7.2)

Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

16.11.2.2.2 BASES

This section is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statement provides the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable".

The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for

calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases on Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors", Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

The reporting requirements of Action(a) implement the requirements of 10CFR20.2203.

16.11.2.3 DOSE - IODINE-131 AND 133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM LIMITING CONDITION OF OPERATION

(ODCM 9.8.1)

The dose to a MEMBER OF THE PUBLIC from Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 16.11-2) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ, and

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limits and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.11.2.3.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.8.2)

Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

16.11.2.3.2 BASES

This section is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the release of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as reasonably achievable". The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix

I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors", Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate controls for Iodine-131, and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition of radionuclides onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

The reporting requirements of Action(a) implement the requirements of 10CFR20.2203.

16.11.2.4 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION LIMITING CONDITION FOR OPERATION

(ODCM 9.2.1)

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 16.11-5 shall be FUNCTIONAL with their Alarm/Trip Setpoints set to ensure that the limits of Section 16.11.2.1 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Section 16.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 16.11-5.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above, immediately declare the channel non-functional.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels FUNCTIONAL, take the ACTION shown in Table 16.11-5. Restore the non-functional instrumentation to FUNCTIONAL status within the time specified in the ACTION, or explain in the next Radioactive Effluent Release Report, pursuant to Technical Specification 5.6.3, why this non-functionality was not corrected within the time specified.

16.11.2.4.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.2.2)

Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated FUNCTIONAL by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL OPERATIONAL TEST at the frequencies shown in Table 16.11-6.

16.11.2.4.2 BASES

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The FUNCTIONALITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Section 16.11.2.1 shall be such that concentrations as low as 1 x 10^{-6} μCi/cc are measurable.

The monitors GT-RE-22 and GT-RE-33 are only required for automatic containment purge isolation in MODES 1 through 4. For plant conditions during CORE ALTERATIONS and during movement of irradiated fuel within containment, the function of the monitors is to alarm only and the trip signals for automatic actuation of CPIS may be bypassed. Based on the guidance provided in Regulatory Guide 1.97 concerning monitoring requirements for containment or purge effluent, the monitors GT-RE-22 and GT-RE-33 do not need to meet the single failure criterion for an Alarm function only during CORE ALTERATIONS or during movement of irradiated fuel in containment. One instrumentation channel at a minimum is required for the alarm only function during refueling activities.

In the event that the containment mini-puge supply and exhaust valves have been closed to satisfy Action 41 of Table 16.11-5 due to non-functionality of GTRE0022 and/or GTRE0033, an allowance is provided in Action 41 to open the containment mini-purge supply and exhaust valves under administrative controls for the purpose of equalizing containment pressure. The administrative controls consist of designating a control room operator to rapidly close the valves when a need for system isolation is indicated.

16.11.2.5 GASEOUS RADWASTE TREATMENT SYSTEM LIMITING CONDITION OF OPERATION

(ODCM 9.9.1)

The VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEM shall be FUNCTIONAL and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 16.11-2) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times

ACTION:

With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days a Special Report that includes the following information:

1) Identification of any non-functional equipment or subsystems, and the reason for the non-functionality,

2) Action(s) taken to restore the non-functional equipment to FUNCTIONAL status, and

3) Summary description of action(s) taken to prevent a recurrence.

16.11.2.5.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.9.2)

16.11.2.5.1.a

Doses due to gaseous releases to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

16.11.2.5.1.b

The installed VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEMS shall be considered FUNCTIONAL by meeting Sections 16.11.2.1 and 16.11.2.2 or 16.11.2.3.

16.11.2.5.2 BASES

The FUNCTIONALITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This control implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

16.11.2.6 EXPLOSIVE GAS MIXTURE LIMITING CONDITION FOR OPERATION

The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 3% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 3% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limit within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration on oxygen to less than or equal to 4% by volume, then take ACTION a. above.

16.11.2.6.1 SURVEILLANCE REQUIREMENTS

The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required FUNCTIONAL by Section 16.11.2.7. This system is covered by Technical Specification 5.5.12 which governs surveillance test frequencies and missed surveillances.

16.11.2.6.2 BASES

This Requirement is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

16.11.2.7 WASTE GAS HOLDUP SYSTEM RECOMBINER EXPLOSIVE GAS MONITORING INSTRUMENTATION LIMITING CONDITION FOR OPERATION

At least one hydrogen and both the inlet and outlet oxygen explosive gas monitoring instrument channels for each WASTE GAS HOLDUP SYSTEM recombiner shall be FUNCTIONAL with their Alarm/Trip Setpoints (with the exception of the "FEED H2 4%/ FEED O2 3%" and "FEED H2 4%/FEED O2 4%" alarms) set to ensure that the limits of Section 16.11.2.6 are not exceeded.

APPLICABILITY: During WASTE GAS HOLDUP SYSTEM operation.

ACTION:

a. With an outlet oxygen monitor channel non-functional, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours.

b. With both oxygen or both hydrogen channels or both the inlet oxygen and inlet hydrogen monitor channels for one recombiner non-functional, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least: 1) once per 4 hours during mechanical or chemical degassing in preparation for plant shutdown, and 2) once per 24 hours during other operations.

c. With the inlet oxygen analyzer non-functional, operation of the system may continue provided the inlet hydrogen is maintained less than 4%. If inlet hydrogen is greater than 4%, suspend oxygen to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least: 1) once per 4 hours during mechanical or chemical degassing operations in preparation for plant shutdown, and 2) once per 24 hours during other operations.

16.11.2.7.1 SURVEILLANCE REQUIREMENTS

This system is covered by Technical Specification 5.5.12 which governs surveillance test frequencies and missed surveillances.

Each waste gas holdup system recombiner explosive gas monitoring instrumentation channel shall be demonstrated FUNCTIONAL by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. Not used

c. A CHANNEL CALIBRATION at least once per 92 days with the use of standard gas samples containing a nominal:

1) One volume percent hydrogen, balance nitrogen and four volume percent hydrogen, balance nitrogen for the hydrogen monitor, and

2) One volume percent oxygen, balance nitrogen, and four volume percent oxygen, balance nitrogen for the inlet oxygen monitor, and

3) 10 ppm by volume oxygen, balance nitrogen and 80 ppm by volume oxygen, balance nitrogen for the outlet oxygen monitor.

16.11.2.7.2 BASES

Mechanical 'degassing operation' is defined as the transfer of gas from the Volume Control Tank (VCT) to the Waste Gas Holdup System when establishing a nitrogen blanket on the VCT in preparation for a plant shutdown. Chemical 'degassing operation' is the process of adding hydrogen peroxide to the RCS after the VCT hydrogen blanket has been replaced with nitrogen per the mechanical degassification process and the RCS has been reduced to less than 180°F. Both mechanical and chemical degassification may lead to an explosive gas mixture in the Waste Gas Holdup System, thus requiring the more restrictive 4-hour sampling. Other operations require 24-hour sampling.

The "FEED H2 4%/FEED O2 3%" AND "FEED H2 4%/FEED O2 4%" alarms are not required to be FUNCTIONAL. These alarms result from the combination of inlet hydrogen and inlet oxygen analyzer outputs while the FSAR only addresses FUNCTIONALITY of each separate analyzer. Only the individual alarms and control functions associated with each analyzer are to be used to determine its functionality. These alarms and control functions are sufficient to ensure that the limits of Section 16.11.2.6 are not exceeded.

The CHANNEL CALIBRATION includes triggering the following alarms at the analyzer and verifying that the required control board annunciators and control functions actuate:

1) Feed Gas High H_2

2) HARC-1104 OAIC-1112 Hi Hi H_2/O_2 O_2 Shutdown

3) H_2 Reactor High Oxygen O_2 Limit

4) Product Gas High H_2

5) Product Gas High Oxygen

6) Product Gas Hi Hi O_2 Shutdown

This surveillance verifies the FUNCTIONALITY of the analyzers' output relays, all interposing relays, and the annunciators. Setpoint verification consists of verifying that the correct setpoint values are entered in the analyzers' database.

16.11.2.8 GAS STORAGE TANKS LIMITING CONDITION FOR OPERATION

The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 2.5 x 10^5 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and, within 48 hours, reduce the tank contents to within the limit, and describe the events leading to this condition in the next Radioactive Effluent Release Report, pursuant to Technical Specification 5.6.3.

16.11.2.8.1 SURVEILLANCE REQUIREMENTS

The provisions of Sections 16.0.2.2 and 16.0.2.3 are applicable, however the allowed surveillance interval extension beyond 25% shall not be exceeded. This system is also covered by Administrative Controls Section 5.5.12 of the plant Technical Specifications.

The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 18 months.

16.11.2.8.2 BASES

The tanks included in this Requirement are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Requirement. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981. The determination of Xe-133 equivalent uses the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, EPA-402-R-93-081, “External Exposure to Radionuclides in Air, Water, and Soil,” 1993.

16.11.3 TOTAL DOSE

16.11.3.1 TOTAL DOSE LIMITING CONDITION FOR OPERATION

(ODCM 9.10.1)

The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

APPLICABILITY: At all times.

ACTION:

With the calculated doses from the release of radioactive materials in gaseous effluents exceeding twice the limits of Section 16.11.2.2a, 16.11.2.2b, 16.11.2.3a, or 16.11.2.3b, calculations should be made including direct radiation contributions from the units and from outside storage tanks to determine whether the above limits of Section 16.11.3.1 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days a Special Report that defines the corrective action to be taken to reduce subsequent release to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.2203, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

16.11.3.1.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.10.2)

16.11.3.1.1.a

Cumulative dose contributions from gaseous effluents shall be determined in accordance with Sections 16.11.2.2.1, and 16.11.2.3.1, and in accordance with the methodology and parameters in the ODCM.

16.11.3.1.1.b

Cumulative dose contributions from direct radiation from the units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirements is applicable only under conditions set forth in ACTION a. of Section 16.11.3.1.

16.11.3.1.2 BASES

This Requirement is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20.1301. The control requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and the radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and from outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits.

For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR 20.2203, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Sections 16.11.1.1 and 16.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

There are three defined effluent release categories: 1.) Releases directly to the hydrosphere; 2.) noble gas releases to the atmosphere; 3.) radioiodine and particulate releases to the atmosphere. For each effluent release category, it is assumed in the dose calculations that an individual with the highest dose potential is the receptor. In general, the adult is considered to be the critical age group for liquid effluents, and the child age group is the most limiting for radioiodine and particulates in gaseous effluents. Thus, it is highly unlikely or impossible for the same individual to simultaneously receive the highest dose via all three effluent categories. For most reactor sites, it is also unlikely that all different potential dose pathways would contribute to the dose to a single real individual. Since it is difficult or impossible to continually determine actual food use patterns and critical age group, for calculational purposes, assumptions are made which tend to maximize doses. Any refinement in the assumptions would have the effect of

reducing the estimated dose. For radionuclides released to the hydrosphere, the degree of overestimation in most situations is such that no individual will receive a significant dose. These conservative assumptions generally result in an overestimation of dose by one or two orders of magnitude. Since these assumptions are reflected in the Radiological Effluent Controls limiting radionuclide releases to design objective individual doses, no offsite individual is likely to actually receive a significant dose. Since the doses from liquid releases are very conservatively evaluated, there is reasonable assurance that no real individual will receive a significant dose from radioactive liquid release pathway. Therefore, only doses to individuals via airborne pathways and dose resulting from direct radiation need to be considered in determining potential compliance to 40 CFR 190*.

The reporting requirements of Action(a) implement the requirements of 10CFR20.2203.

* NUREG-0543, “Methods for Demonstrating LWR compliance with the EPA Uranium Fuel Cycle Standard (40 CFR 190)”, Congel, F. J., Office of Nuclear Reactor Regulation, USNRC. January, 1980. pp. 5-8.

16.11.4 RADIOLOGICAL ENVIRONMENTAL MONITORING

16.11.4.1 MONITORING PROGRAM LIMITING CONDITION OF OPERATION (ODCM 9.11.1)

The Radiological Environmental Monitoring Program shall be conducted as specified in Table 16.11-7.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 16.11-7, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Technical Specification 5.6.2, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 16.11-8 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Sections 16.11.1.2, 16.11.2.2, or 16.11.2.3. When more than one of the radionuclides in Table 16.11-8 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 16.11-8 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Sections 16.11.1.2, 16.11.2.2 or 16.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report, required by Technical Specification 5.6.2.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 16.11-7, identify specific locations for

* The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program*. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In the next Annual Radiological Environmental Operating Report include the revised figure(s) and tables reflecting the new sample location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of new location(s) for obtaining samples.

d. When LLDs specified in Table 16.11-9 are unachievable due to uncontrollable circumstances such as background fluctuations, unavailable small sample sizes, the presence of interfering nuclides, etc., the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

16.11.4.1.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.11.2)

The radiological environmental monitoring samples shall be collected pursuant to Table 16.11-7 and shall be analyzed pursuant to the requirements of Table 16.11-7 and the detection capabilities required by Table 16.11-9.

16.11.4.1.2 BASES

The Radiological Environmental Monitoring Program provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for the initial monitoring program was provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLD's). The LLD's required by Table 16.11-9 are considered optimum for routine environmental measurements in industrial laboratories.

* Excluding short term or temporary unavailability.

16.11.4.2 LAND USE CENSUS LIMITING CONDITION OF OPERATION

(ODCM 9.12.1)

A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. The Land Use Census shall identify water intakes constructed within 10 river miles downstream of the plant discharge point.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated by Section 16.11.2.3.1, identify the new location(s) in the next Radioactive Effluent Release Report, pursuant to Technical Specification 5.6.3.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Section 16.11.4.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program except for vegetation samples which shall be added to the program before the next growing season. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this Land Use Census was conducted. In the next Annual Radiological Environmental Operating Report include the revised figure(s) and tables reflecting the new sample location(s) with information supporting the change in sample location.

c. With a Land Use Census identifying a water intake within 10 river miles downstream of the plant discharge point, implement the appropriate waterborne or ingestion sampling required by Table 16.11-7.

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* Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each to two different direction sectors with the highest predicted D/Q's in lieu of the garden census. Requirements for broad leaf vegetation sampling in Table 16.11-7, Part 4.c shall be followed, including analysis of control samples.

16.11.4.2.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.12.2)

The Land Use Census shall be conducted during the growing season at least once per 12 months using that information which will provide the best results, such as, but not limited to, door-to-door survey, aerial survey, or by consulting local agriculture authorities and/or residents. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

16.11.4.2.2 BASES

This Requirement is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. Information that will provide the best results, such as door-to-door survey, aerial survey, or consulting with local agricultural authorities, shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

16.11.4.3 INTERLABORATORY COMPARISON PROGRAM LIMITING CONDITION OF OPERATION

(ODCM 9.13.1)

Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the USNRC.

APPLICABILITY: At all times.

ACTION:

With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

16.11.4.3.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.13.2)

The Interlaboratory Comparison Program shall be described in the plant procedures. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

16.11.4.3.2 BASES

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purpose of Section IV.B.2 of Appendix I to 10 CFR Part 50.

16.11.5 ADMINISTRATIVE CONTROLS

16.11.5.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (ODCM 7.1)

Routine Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall include the results of Land Use Census required by Section 16.11.4.2. It shall also include a listing of new locations for environmental monitoring identified by the Land Use Census pursuant to Section 16.11.4.2.

The Annual Radiological Environmental Operating Report shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to Section 16.11.4.1, as well as summarized tabulated results of these analyses and measurements in the format of 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 as soon as possible in a supplementary report. The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the midpoint between the two reactors; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action being taken if the specified program is not being performed as required by Section 16.11.4.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by Section 16.11.4.1 and discussion of all deviations from the sampling schedule of Table 16.11-7, discussion of environmental sample measurements that exceed the reporting levels of Table 16.11-8, but are not the result of the plant effluents, pursuant to Section 16.11.4.1; and discussion of all analyses in which the LLD required by Table 16.11-9 was not achievable.

16.11.5.1.1 BASES

The reporting requirement for the Annual Radiological Environmental Operating Report is provided to ensure compliance with Technical Specification 5.6.2. This requirement was relocated from the Offsite Dose Calculation Manual to FSAR Chapter 16.

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* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

16.11.5.2 RADIOACTIVE EFFLUENT RELEASE REPORT

(ODCM 7.2)

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, "Revision 1, June 1974, with data summarized on a quarterly basis in a format acceptable to the NRC.

The Radioactive Effluent Release Report shall include an annual summary of hourly meteorological data collected over the previous calender year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability* .

This report shall also include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. This report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figures 16.11-1 and 16.11-2) during the report period using historical average atmospheric conditions. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report shall include an assessment of radiation doses to the most likely exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Doses to the MEMBER OF THE PUBLIC shall be calculated using the methodology and parameters of the ODCM.

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* In lieu of submission with the Annual Radioactive Effluent Release Report, Union Electric has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

As required by 10 CFR 72.44(d)(3), an annual report shall be submitted to the Commission in accordance with 10 CFR 72.4, specifying the quantity of each of the principal radionuclides released to the environment in liquid and in gaseous effluents during the previous 12 months of operation. The report must be submitted within 60 days after the end of the 12-month monitoring period.

The Radioactive Effluent Release Report shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Report shall include a summary description of any major changes made during the year to any Liquid or Gaseous Treatment Systems, pursuant to Offsite Dose Calculation Manual. It shall also include a listing of new locations for dose calculations identified by the Land Use Census pursuant to Section 16.11.4.2.

Reporting requirements for changes to Solid Waste Treatment Systems are addressed in APA-ZZ-01011, PROCESS CONTROL PROGRAM (PCP).

The Radioactive Effluent Release Report shall also include the following information: An explanation as to why the liquid or gaseous effluent monitoring instrumentation was not restored to service within the time specified, and a description of the events leading the liquid holdup tanks or gas storage tanks exceeding the limits of Section 16.11.1.5 or 16.11.2.8.

The Radioactive Effluent Release Report shall include as part of or submitted concurrent with, a complete and legible copy of all revisions of the ODCM that occurred during the year pursuant to Technical Specification 5.5.1.

Solid Waste reporting is addressed in APA-ZZ-01011, PROCESS CONTROL PROGRAM (PCP).

16.11.5.2.1 BASES

The reporting requirement for the Radioactive Effluent Release Report is provided to ensure compliance with Technical Specification 5.6.3. This requirement was relocated from the Offsite Dose Calculation Manual implementing procedure to FSAR Chapter 16.

In addition to the above reporting requirement, an annual report shall also be submitted in compliance with the HI-STORM UMAX Certificate of Compliance (CoC), Appendix A, Section 5.1

TABLE 16.11-1 RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

1. Discharge Monitor Tanks (Batch Release) (2)			
SAMPLING FREQUENCY(7)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Prior to Each Batch	Prior to Each Batch	Principal Gamma Emitters (3)	5E-7
		I-131	1E-6
		Dissolved and Entrained Gases (Gamma Emitters)	1E-5
		H-3	1E-5
	Monthly Composite (4)	Gross Alpha	1E-7
	Quarterly Composite (4)	Sr-89, Sr-90	5E-8
		Fe-55	1E-6
		Ni-63	5E-8
		Np-237	5E-9
		Pu-238	5E-9
		Pu-239/240	5E-9
		Pu-241	5E-7
		Am-241	5E-9
		Cm-242	5E-9
		Cm-243/244	5E-9

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these Requirements, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ S_b}{E \times V \times 2.22E6 \times Y \times \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

S_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22E6= the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec-1), and

Δt = the elapsed time between the midpoint of the sample collection period, and the time of counting (sec). For batch releases, Δt=0.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLD's will be achieved under routine conditions.

(2) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

(3) The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report pursuant to Technical Specification 5.6.3, in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

TABLE 16.11-1 (Sheet 2)

(4) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released. Prior to analysis, all samples taken for the composite shall be thoroughly mixed in order for the composite samples to be representative of the effluent release.

(5) Deleted

(6) Deleted

(7) Samples shall be representative of the effluent release.

TABLE 16.11-2 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS FUNCTIONAL	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Discharge Monitor (HB-RE-18)	1	31
b. DELETED		
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Blowdown Discharge Line (HB-FE-2017)	1	34
b. Steam Generator Blowdown Discharge Line (BM-FE-0054)	1	34
c. Cooling Tower Blowdown and Bypass Flow Totalizer (FYDB1017A)	1	34
3. Discharge Monitoring Tanks (DMT's) Level		
a. DMT A (HB-LI-2004)	1	33
b. DMT B (HB-LI-2005)	1	33

ACTION STATEMENTS

ACTION 31 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Section 16.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

TABLE 16.11-2 (Sheet 2)

ACTION STATEMENTS

ACTION 32 - DELETED

ACTION 33 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided the volume discharged is determined by alternate means.

ACTION 34 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

TABLE 16.11-3 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Discharge Monitor (HB-RE-18)	D	P	R(2)	Q(1)
b. DELETED				
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Blowdown Discharge Line (HB-FE-2017)	D(3)	N.A.	R	N.A.
b. DELETED				
c. Cooling Tower Blowdown and Bypass Flow Totalizer (FYDB1017A)	D(3)	N.A.	R	N.A.
3. Discharge Monitoring Tanks (DMT's) Level				
a. DMT A(HB-LI-2004)	Prior to release (4)	N.A.	R	N.A.
b. DMT B(HB-LI-2005)	Prior to release (4)	N.A.	R	N.A.

TABLE 16.11-3 (Sheet 2)

TABLE NOTATIONS

1. The CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur as appropriate if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or

   b. Circuit failure (alarm only), or

   c. Instrument indicates a downscale failure (alarm only), or

   d. Instrument controls not set in operate mode (alarm only).

2. The initial CHANNEL CALIBRATION shall be performed using one or more of the reference (gas or liquid and solid) standards obtained from the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy, measurement range, and establish monitor response to a solid calibration source. For subsequent CHANNEL CALIBRATION, NIST traceable standard (gas, liquid, or solid) may be used; or a gas, liquid, or solid source that has been calibrated by relating it to equipment that was previously (within 30 days) calibrated by the same geometry and type of source standard traceable to NIST.

3. CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

4. CHANNEL CHECK shall consist of verifying indication of tank level during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made from the DMT.

TABLE 16.11-4 RADIOACTIVE GASEOUS EFFLUENTS SAMPLING AND ANALYSIS PROGRAM

1. Waste Gas Decay Tank			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Prior to each release- grab sample	Prior to each tank	Principal Gamma Emitters- particulate, iodine, noble gas (2)	1E-4
Continuous	See footnote 8		

2. Containment Purge or Vent			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Prior to each release- grab sample	Prior to each release	Principal Gamma Emitters- particulate, iodine, noble gas (2) H-3(oxide)	1E-4 1E-6
Continuous	See footnote 8		

3. Unit Vent (3)			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Monthly- grab sample (3)(4)	Monthly (3)(4)	Principal Gamma Emitters- noble gas (2) H-3(oxide)	1E-4 1E-6
Continuous (6)	Weekly (7)	I-131 I-133 Principal Gamma Emitters- particulate nuclides only (2)	1E-12 1E-10 1E-11
	MonthlyComposite	Gross Alpha	1E-11
	Quarterly Composite	Sr-89, Sr-90, Ni-63, Fe-55	1E-11

4. Radwaste Building Vent			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Monthly- grab sample	Monthly	Principal Gamma Emitters- noble gas (2)	1E-4
Continuous (6)	Weekly (7)	I-131 I-133 Principal Gamma Emitters- particulate nuclides only (2)	1E-12 1E-10 1E-11
	Monthly Composite	Gross Alpha	1E-11
	Quarterly Composite	Sr-89, Sr-90, Ni-63, Fe-55	1E-11

5. Laundry Decontamination Facility Dryer Exhaust			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Continuous (6)	Weekly (7)	Principal Gamma Emitters- particulate nuclides only (2)	1E-11
	Monthly (10) Composite	Gross Alpha	1E-11
	Quarterly (10) Composite	Sr-89, Sr-90, Ni-63, Fe-55	1E-11

TABLE 16.11-4 (Sheet 2)

6. Containment ILRT Depressurization (Post-test Vent)			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Prior to each release- grab sample	Prior to each release	Principal Gamma Emitters- particulate, iodine, noble gas (2) H-3(oxide)	1E-4 1E-6

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these Requirements, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ S_b}{E \times V \times 2.22E6 \times Y \times \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

S_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22E6= the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of the sample collection period, and the time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLD's will be achieved under routine conditions.

(2) The principal gamma emitters for which the LLD Requirement applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Any nuclide which is identified in the sample and which is also listed in the ODCM gaseous effluents dose factor tables, shall be analyzed and reported in the Radioactive Effluent Release Report.

(3) If the Unit Vent noble gas monitor (GT-RE-21B) shows that the effluent activity has increased (relative to the pre-transient activity) by more than a factor of 3 following a reactor shutdown, startup, or a thermal power change which exceeds 15% of the rated thermal power within a 1 hour period, samples shall be obtained and analyzed for noble gas, particulates and iodines. This sampling shall continue to be performed at least once per 24 hours for a period of 7 days or until the Unit Vent noble gas monitor no longer indicates a factor of 3 increase in Unit Vent noble gas activity, whichever comes first.

(4) Tritium grab samples shall be taken and analyzed at least once per 24 hours when the refueling canal is flooded.

(5) Deleted.

(6) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Sections 16.11.2.1, 16.11.2.2, and 16.11.2.3.

(7) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or removal from the sampler. When sampling is performed in accordance with footnote 3 (above), then the LLD may be increased by a factor of 10.

(8) Continuous sampling of this batch release pathway is included in the continuous sampling performed for the corresponding continuous release pathway.

(9) Samples shall be representative of the effluent release.

(10) Required only if Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, Ce-141, or Ce-144 are detected in principle gamma emitter analyses.

TABLE 16.11-5 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS FUNCTIONAL	APPLICABILITY	ACTION
1. Unit Vent System			
a. Noble Gas Activity Monitor - Providing Alarm (GT-RE-21)	1	At all times	40,46
b. Iodine Sampler	1	At all times	43
c. Particulate Sampler	1	At all times	43
d. Unit Vent Flow Rate	1	At all times	45
e. Particulate and Radioiodine Sampler Flow Rate Monitor	1	At all times	43
2. Containment Purge System			
a. Noble Gas Activity Monitor			
- Providing Alarm and Automatic Termination of Release (GT-RE-22, GT-RE-33)	2	MODES 1,2,3, and 4.	41
- Providing Alarm function only	1	During CORE ALTERATIONS or movement of irradiated fuel within the containment	42
b. Iodine Sampler	1	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment	43
c. Particulate Sampler	1	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment	43
d. Containment Purge Ventilation Flow Rate	N/A	N/A	N/A

TABLE 16.11-5 (Sheet 2)

e. Particulate and Radioiodine Sampler Flow Rate Monitor	1	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment	43
3. Radwaste Building Vent System			
a. Noble Gas Activity Monitor-Providing Alarm and Automatic Termination of Release (GH-RE-10)	1	At all times	38,40
b. Iodine Sampler	1	At all times	43
c. Particulate Sampler	1	At all times	43
d. Radwaste Building Vent Flow Rate	N/A	N/A	N/A
e. Particulate and Radioiodine Sampler Flow Rate Monitor	1	At all times	43
4. Laundry Decontamination Faciliy Dryer Exhaust			
a. Particulate Monitor	1	When the dryers are operating	47
b. Particulate Monitor Flow Rate Meter	1	When the dryers are operating	47
c. Dryer Exhaust Ventilation Flow Rate	NA	NA	NA

ACTION STATEMENTS

ACTION 38 - With the number of low range channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

TABLE 16.11-5 (Sheet 3)

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 39 - Deleted.

ACTION 40 - With the number of low range channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 41 - With the number of channels FUNCTIONAL one less than required by the Minimum Channels FUNCTIONAL requirement, restore the affected channel to FUNCTIONAL status within 4 hours. If the non-functional channel is not restored within 4 hours or with no channels FUNCTIONAL, immediately suspend the release of radioactive effluents via this pathway.

Containment mini-purge supply and exhaust valves that have been closed to satisfy this Action may be opened under administrative controls provided either:

a. one channel is FUNCTIONAL, or

b. the requirements for Table 16.11-5 Function 1.a are met and the requirements for minimum channels FUNCTIONAL for the Unit Vent Noble Gas Monitor (GT-RE-21) specified in Table 16.3-7 Function 3 are met.

The administrative controls consist of designating a control room operator to rapidly close the valves when a need for system isolation is indicated.

ACTION 42 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, and if the containment equipment hatch is open, then immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. If the containment equipment hatch is not open, then suspend the release of radioactive effluents via this pathway or immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment.

ACTION 43 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 16.11-4.

TABLE 16.11-5 (Sheet 4)

ACTION 44 - Deleted.

ACTION 45 - Flow rate for this system shall be based on fan status and operating curves or actual measurements.

ACTION 46 - For midrange and high range channels only - with the number of FUNCTIONAL channels less than required by the Minimum Channels FUNCTIONAL requirement, take the action specified in Section 16.3.3.4, ACTION C.

ACTION 47 - With the number of channels FUNCTIONAL less than required by the Minumum Channels FUNCTIONAL requirement, immediately suspend the release of radioactivite effluents via this pathway.

TABLE 16.11-6 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Unit Vent System					
a. Noble Gas Activity Monitor - Providing Alarm (GT-RE-21)	D	M	R(3)	Q(2)	At all times
b. Iodine Sampler	W	N.A.	N.A.	N.A.	At all times
c. Particulate Sampler	W	N.A.	N.A.	N.A.	At all times
d. Unit Vent Flow Rate	N.A.	N.A.	R(4)	Q	At all times
e. Particulate and Radioiodine Sampler Flow Rate Monitor	D	N.A.	R	Q	At all times
2. Containment Purge System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (GT-RE-22, GT-RE-33)	N.A.	P	N.A.	N.A.	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment
b. Iodine Sampler	W	N.A.	N.A.	N.A.	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment
c. Particulate Sampler	W	N.A.	N.A.	N.A.	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment
d. Containment Purge Ventilation Flow Rate	N.A.	N.A.	R(4)	N.A.	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment

TABLE 16.11-6 (Sheet 2)

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
e. Particulate and Radioiodine Sampler Flow Rate Monitor	D	N.A.	R	N.A.	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment
3. Radwaste Building Vent System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (GH-RE-10)	D,P	M,P	R(3)	Q(1)	At all times
b. Iodine Sampler	W	N.A.	N.A.	N.A.	At all times
c. Particulate Sampler	W	N.A.	N.A.	N.A.	At all times
d. Radwaste Building Vent Flow Rate	N.A.	N.A.	R(4)	N.A.	At all times
e. Particulate and Radioiodine Sampler Flow Rate Monitor	D	N.A.	R	N.A.	At all times
4. Laundry Decontamination Facility Dryer Exhaust					
a. Particulate Monitor	NA	D	A	Q(5)	When the dryers are operating
b. Particulate Monitor Flow Rate Meter	D	NA	A	NA	When the dryers are operating
c. Dryer Exhaust Ventilation Flow Rate	NA	NA	R(4)	NA	When the dryers are operating

TABLE 16.11-6 (Sheet 3)

1. The CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur as appropriate if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or

   b. Circuit failure (alarm only), or

   c. Instrument indicates a downscale failure (alarm only), or

   d. Instrument controls not set in operate mode (alarm only).

2. The CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm Setpoint, or

   b. Circuit failure, or

   c. Instrument indicates a downscale failure, or

   d. Instrument controls not set in operate mode.

3. The initial CHANNEL CALIBRATION shall be performed using one or more of the reference (gas or liquid and solid) standards certified by the National Institute of Standards & Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy, measurement range, and establish monitor response to a solid calibration source. For subsequent CHANNEL CALIBRATION, NIST traceable standard (gas, liquid, or solid) may be used; or a gas, liquid, or solid source that has been calibrated by relating it to equipment that was previously (within 30 days) calibrated by the same geometry and type of source standard traceable to NIST.

4. If flow rate is determined by exhaust fan status and fan performance curves, the following surveillance operations shall be performed at least once per 18 months:

   a. The specific vent flows by direct measurement, or

   b. The differential pressure across the exhaust fan and vent flow established by the fan's "flow-ΔP" curve, or

   c. The fan motor horsepower measured and vent flow established by the fan's "flow-horsepower" curve.

5. The CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and the shutdown of the dryers occur as appropriate if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

   b. Monitor failure.

TABLE 16.11-7 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM'

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS [(1)]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation[(2)]	Forty routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of sixteen stations, one in each meteorological sector in the general area of the SITE BOUNDARY; Four of the stations shall be placed to monitor for gamma and neutron dose from the ISFSI; An outer ring of stations, one in each meteorological sector in the 6- to 8-km (3 to 5 mile) range from the site; and Eight stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly	Gamma dose for each sample. Neutron dose for the four samples monitoring ISFSI direct radiation.
2. Airborne Radioiodine and Particulates	Samples from five locations; Three samples from close to the SITE BOUNDARY locations, in different sectors, with high calculated annual average ground level D/Qs. One sample from the vicinity of a community located near the plant with a high calculated annual average ground level D/Q. One sample from a location in the vicinity of Fulton, MO.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Canister: I-131 analysis for each sample. Gamma isotopic analysis(5) for each sample.

TABLE 16.11-7 (Sheet 2)

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS (1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne			
a. Surface (6) (river)	One sample upstream One sample downstream	Composite sample over 1-month period (7).	Gamma Isotopic(5) and tritium analysis for each sample.
b. DELETED			
c. Groundwater (non-drinking water)	Groundwater samples from non-drinking water shallow and deep(12) monitoring wells located as follows: Samples from one deep well located upgradient of the plant power block and one deep well located downgradient of the sludge lagoons. Samples from six shallow wells or groundwater sumps in locations suitable to monitor for subsurface leakage from power block structures and components. Samples from five shallow wells located along the discharge pipeline corridor. Samples from three shallow wells near the property boundary located to monitor for migration of contaminated groundwater from the discharge pipeline to the nearest potable water well.	Quarterly	Gamma isotopic(5) and tritium analyses for each sample. If contaminated with gamma emitting nuclides of plant origin, analyze for HTD nuclides(11).

TABLE 16.11-7 (Sheet 3)

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS [1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
d. Drinking (river water)	One sample of each of one to three of the nearest water supplies within 10 miles downstream that could be affected by its discharge.	Composite sample over 2-week period[7] when I-131 analysis is performed, monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year[9]. Composite for gross beta and gamma isotopic analyses[5] monthly. Composite for tritium analysis quarterly.
	One sample from a control location.		

As there are no drinking water intakes within 10 miles downstream of the discharge point, the drinking water pathway is currently not included as part of the Callaway Plant Radiological Environmental Monitoring Program. Should the annual Land Use Census identify water intakes within 10 river miles downstream of the discharge point, the program will be revised to include this pathway.

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS [1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
e. Drinking (potable well water)	Samples of potable well water appropriate for monitoring for radioactivity in drinking water supplies in areas most likely to be affected by a spill or leak.	Quarterly	Gamma isotopic[5] and tritium analyses for each sample. If contaminated with nuclides of plant origin, analyze for HTD nuclides[11].
	Two samples of potable well water from the community of Portland, MO.		
	One sample of Callaway Plant potable water.		
	One sample of potable well water from each resident bordering plant property along Mud Creek and Logan Creek.		
f. Sediment from river shoreline	One sample from downstream area with existing or potential recreational value	Semiannually	Gamma isotopic analysis[5] for each sample
	One sample from upstream control location.		

TABLE 16.11-7 (Sheet 4)

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS (1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
g. Shoreline sediment from sludge ponds	Shoreline sediment from each on site sludge pond most likely to be affected. One sample from each in-service sludge pond. One sample from each wetlands pond.	Annually	Gamma isotopic(5) analysis for each sample.
4. Ingestion			
a. Milk	Samples from milking animals in three different meteorological sectors within 5 km (3 mile) distance having the highest dose potential. If there are none, then one sample from milking animals in each of three different meteorological sectors between 5 to 8 km (3 to 5 mile) distance where doses are calculated to be greater than 1 mrem per yr.(9) One sample from milking animals at a control location, 15 to 30 km (10 to 20 mile) distance and in the least prevalent wind direction.	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic(5) and I-131 analyses for each sample

Due to the lack of milking animals which satisfy these requirements, the milk pathway is currently not included as part of the Callaway Plant Radiological Environmental Monitoring Program. Should the Annual Land Use Census identify the existence of milking animals in locations which satisfy these requirements, then the program will be revised to include this pathway.

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS (1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
b. Fish	One sample of each commercially and recreationally important species in vicinity of plant discharge area. One sample of same species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis(5) on edible portions for each sample

TABLE 16.11-7 (Sheet 5)

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS (1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest(10)	Gamma isotopic analysis(5) on edible portion for each sample
As there are no areas irrigated by water in which liquid plant wastes have been discharged within 50 miles downstream of the discharge point, this sample type is not currently included as part of the Callaway Plant Radiological Environmental Monitoring Program. Should the annual Land Use Census identify irrigation water intakes within 10 river miles downstream of the discharge point, the program will be revised to include this sample type.			
	Samples of three different kinds of broad leaf vegetation if available grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed	Monthly when available	Gamma isotopic (5) and I-131 analyses
	One sample of each of the similar broad leaf vegetation grown 15 to 30 km (10 to 20 mile) distant in the least prevalent wind direction if milk sampling is not performed	Monthly when available	Gamma isotopic (5) and I-131 analyses
5. Soil	Surface soil samples suitable for monitoring for ground deposition if radioactivity in gaseous effluents as follows: Four ecology plots located in four quadrants surrounding the plant. One control location from an area not likely to be influenced by plant gaseous effluents.	Annually	Gamma isotopic(5) analysis for each sample.

TABLE 16.11-7 (Sheet 6)

TABLE NOTATIONS

1. Specific parameters of distance and direction sector from the centerline of one unit, and additional description where pertinent, shall be provided for each and every sample location in Table 16.11-7 in a table and figure(s) in the appropriate plant procedures. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 5.6.2.

   It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable specific alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program. Submit in the next Annual Radiological Environmental Operating Report documentation for a change including the revised figure(s) and table reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.

   The selection of sample locations should consider accessibility of sample site, availability of power, wind direction frequency, sector population, equipment security, and the presence of potentially adverse environmental conditions (such as unusually dusty conditions, etc.).

2. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) and/or an optically stimulated luminescent dosimeter (OSLD), are considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

3. Deleted.

4. Deleted.

5. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

6. The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area near the downstream edge of the mixing zone.

7. In this program, composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

8. Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

9. The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

10. If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 16.11-7 (Sheet 7)

11. In this program, HTD nuclides are defined as ^{89}Sr, ^{90}Sr, ^{55}Fe, ^{63}Ni, ^{237}Np, ^{238}Pu, ^{241}Am, ^{242}Cm, and $^{243/244}Cm$.

12. In this program, a shallow well is defined as a well which extracts groundwater from the vadose zone. A deep well is defined as a well which extracts groundwater from the saturated zone.

TABLE 16.11-8 REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/ℓ)[a]	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)[b]	MILK (pCi/ℓ)[a]	FOOD PRODUCTS pCi/kg, wet)[b]
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95**	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140**	200			300	

(a) Multiply the values in this table by 1E-9 to convert to units of µCi/ml.
(b) Multiply the values in this table by 1E-9 to convert to units of µCi/g.

* For drinking water samples. This is 40 CFR Part 141 value. For surface water samples, a value of 30,000 pCi/ℓ may be used.

** Total activity, parent plus daughter activity.

TABLE 16.11-9 DETECTION CAPABILITITES FOR ENVIRONMENTAL SAMPLE ANALYSIS

LOWER LIMIT OF DETECTION (LLD) (1), (2), (3)

ANALYSIS	SURFACE WATER (pCi/ℓ)[a]	DRINKING WATER (pCi/ℓ)[a]	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)[b]	MILK (pCi/ℓ)[a]	FOOD PRODUCTS (pCi/kg, wet)[b]	SEDIMENT (pCi/kg, dry)[b]
Gross Beta	4	4	0.01				
H-3	3000	2000					
Mn-54	15	15		130			
Fe-59	30	30		260			
Co-58,60	15	15		130			
Zn-65	30	30		260			
Zr-Nb-95*	15	15					
I-131	**	1	0.07		1	60	
Cs-134	15	15	0.05	130	15	60	150
Cs-137	18	18	0.06	150	18	80	180
Ba-La-140*	15	15			15		

(a) Multiply the values in this table by 1E-9 to convert to units of µCi/ml.
(b) Multiply the values in this table by 1E-9 to convert to units of µCi/g.

* Total activity, parent plus daughter activity.

** For surface water samples, the LLD of gamma isotopic analysis may be used.

TABLE 16.11-9 (Sheet 2)

TABLE NOTATIONS

1. This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the listed nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report.

2. Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13, Revision 1, July 1977.

3. The LLD is defined, for purposes of these Requirements, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

   For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 S_b}{E \times V \times 2.22E6 \times Y \times \exp(-\lambda \Delta t)}$$

   Where:

   LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

   S_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

   E = the counting efficiency (counts per disintegration),

   V = the sample size (units of mass or volume),

   2.22E6 = the number of disintegrations per minute per microCurie,

   Y = the fractional radiochemical yield, when applicable,

   λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

   Δt = the elapsed time between the end of the sample collection period, and the time of counting (sec).

   Typical values of E, V, Y, and Δt should be used in the calculation.

   It should be recognized that the LLD is defined as a "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLD's will be achieved under routine conditions.

16.12 ADMINISTRATIVE CONTROLS

16.12.1 ORGANIZATION - UNIT STAFF

The Unit organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 16.12-1;

b. A site Fire Brigade of at least five members (may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.) shall be maintained onsite at all times. The Fire Brigade shall not include the Shift Manager, and the other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

16.12 ADMINISTRATIVE CONTROLS

16.12.2 BURNUP ANALYSIS RECORDS

A complete record of analyses performed in support of Technical Specification 3.7.17 for the duration that the spent fuel assembly remains in Region 2 or 3 of the fuel storage pool shall be maintained.

16.12 ADMINISTRATIVE CONTROLS

16.12.3 PROCEDURES AND PROGRAMS

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

a. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

b. Emergency Diesel Generator Reliability Program

An emergency diesel generator reliability program that establishes the requirements and guidelines for emergency diesel generator reliability, availability, and monitoring. The program shall include the following:

1) Emergency diesel generator reliability performance goals (target reliability) based upon the station blackout coping assessment. Target reliability goal monitoring is accomplished through monitoring methods that are based upon those described in Appendix D of NUMARC 87-00,

2) Measures to ensure detailed root cause analysis of emergency diesel generator failures is performed and effective corrective actions are taken in response to failures,

3) Implementation of an emergency diesel generator preventive maintenance program that is consistent with the Maintenance Rule, and

4) Monitoring of emergency diesel generator availability and performance parameters to ensure the target reliability is met or exceeded.

16.12 ADMINISTRATIVE CONTROLS

16.12.4 REPORTING REQUIREMENTS

The following reporting requirements shall be maintained:

a. Monthly Operating Report Data

Routine reports of operating statistics and shutdown experience shall be input into an industry database on a monthly basis. The data (for each calendar month) that is described in Generic Letter 97-02, “Revised Contents of the Monthly Operating Report,” shall be provided to the NRC via the industry database by the last day of the month following the end of each calendar quarter.

TABLE 16.12-1 MINIMUM SHIFT CREW COMPOSITION#

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3 or 4	MODE 5 or 6
SM	1	1
SRO	1	None
FBL	1	1
RO	2	1
OT/AOT	5*	4*
STA	1**	None

SM - Shift Manager with a Senior Operator license on Unit 1
SRO - Individual with a Senior Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
OT/AOT- Operations Technician or respective watchstation qualified Assistant Operations Technician
STA - Shift Technical Advisor
FBL - Operating Supervisor or Operations Technician who meets the requirements of FSAR Section 9.5.1.5.1 for Fire Brigade Leader

The Shift Crew Composition may be one less than the minimum requirements of Table 16.12-1 for a period of time not to 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 of Table 16.12-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

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* In Modes 1, 2, 3, or 4, at least 2 of the 5 shall be fully qualified OTs. In Modes 5 or 6, at least 1 of the 4 shall be a fully qualified OT. The totals may be reduced by up to 2, if a respective number of Fire Brigade qualified personnel are on site to respond for Fire Brigade duty. At least 2 OTs/AOTs shall meet the requirements of FSAR Section 9.5.1.5.2, and 1 of these shall be designated as the Assistant Fire Brigade Leader.

** The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Manager or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

\# Additional staffing may be required by the Radiological Emergency Response Plan (RERP).

16.15 FIRE PROTECTION

Fire Protection FUNCTIONALITY requirements are detailed in APA-ZZ-00703.

16.24 ASME INSERVICE INSPECTION PROGRAM

16.24.1 The Callaway Inservice Inspection Program is administered under procedure EDP-ZZ-01003, “Inservice Inspection Program.”

16.24.2 The Callaway Containment Exterior Concrete and Tendon Inspection Program is administered under procedure ESP-ZZ-01012, "Containment Post-Tensioning System Inspection."

16.24.3 The Callaway Containment Pressure Boundary Inspection Program is administered under procedure ESP-ZZ-01016, "ASME Section XI, IWE Containment Pressure Boundary Inspection."

16.24.4 Sections 16.0.1 and 16.0.2 do not apply where 10 CFR 50.55a(g) takes precedence.

16.25 PROCESS CONTROL PROGRAM (PCP)

16.25.1 PROGRAMS DEFINITION

The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way to assure compliance with 10 CFR 20, 61, and 71, and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

16.25.2 PROGRAMS CHANGES

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by OQAM Section 17.11. This documentation shall contain:

    1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

    2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the ORC and the approval of the Plant Manager.