CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM shall be that program which contains the current formula, sampling, analyses, test, 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 Part 20, 10 CFR Part 71 and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the 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

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3338 MWt.

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

(4) - Manual SSPS functional input check every 18 months.

(5) - Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - If not performed in the previous 24 hours, conduct a functional test of the Manual Reactor Trip Switches (using voltmeters).

(10) - If not performed in the previous 24 hours, conduct a functional test of:

- Reactor Trip Breaker UV Trip (via SSPS)
- Reactor Trip Breaker Shunt Trip (via manual pushbutton controls)

(11) - Perform a functional test of:

- Reactor Trip Breaker UV Trip (via SSPS) and conduct response time testing of UV/Breakers (event recorders)
- Reactor Trip Breaker Shunt Trip (via manual pushbutton controls)

(12) - Perform periodic maintenance on Reactor Trip Breakers and Reactor Trip Bypass Breakers semiannually as follows:

a. response time testing, (3 times) (visicorder) trend data
b. trip bar lift force measurements
c. UV output force measurement
d. dropout voltage check
e. servicing/lubrication/adjustments (See Table 3.3-1 Notation ###)
f. repeat testing steps (a-d) following any necessary actions at step (e)

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line (1-R18)	1	26
b. Steam Generator Blowdown Line (1-R19 A, B, C, and D)	4	27
2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Containment Fan Coolers - Service Water Line (1-R13 A, B, C, D, E) Discharge	5	28
3. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	29
b. Steam Generator Blowdown Line	4	29
4. TANK LEVEL INDICATING DEVICES		
a. Temporary Outside Storage Tanks as Required	1	30

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line (1-R18)	D	P#	R(3)	Q(1)
b. Steam Generator Blowdown Line (1-R19 A, B, C, and D)	D	M	R(3)	Q(1)
2. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Containment Fan Coolers - Service Water Line (1-R13 A, B, C, D, E) Discharge	D	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	N.A.
b. Steam Generator Blowdown Line	D(4)	N.A.	R	N.A.
4. TANK LEVEL INDICATING DEVICES**				
a. Temporary Outside Storage Tanks as Required	D*	N.A.	R	Q

TABLE 4.3-12 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power)

3. Instrument indicates a downscale failure. (Indication on instrument drawer in Control Equipment Room only)

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power)

3. Instrument indicates a downscale failure. (Indication on instrument drawer in Control Equipment Room only)

4. Instrument controls not set in operate mode. (On instruments equipped with operate mode switches only)

(3) The initial CHANNEL CALIBRATION was performed using appropriate liquid or gaseous calibration sources obtained from reputable suppliers. The activity of the calibration sources were reconfirmed using a multi-channel analyzer which was calibrated using one or more NBS standards.

(4) 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.

* During liquid additions to the tank.

** If tank level indication is not provided, vertification will be done by visual inspection.

\# The R18 channel is an in-line channel which requires periodic decontamination. Any count rate indication above 10,000 cpm constitutes a CHANNEL CHECK for compliance purposes.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (1-R41C)	P	P	R(3)	Q(1)	*
b. Oxygen Monitor	D	N.A.	Q(4)	M	**
2. PLANT VENT HEADER SYSTEM #					
a. Noble Gas Activity Monitor (1-R16 or 1-R41C)	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sample	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	N.A.	*
e. Sampler Flow Rate Monitor	W	N.A.	R	N.A.	*

The following process streams are routed to the plant vent where they are effectively monitored by the instruments described:

(a) Condenser Air Removal System
(b) Auxiliary Building Ventilation System
(c) Fuel Handling Building Ventilation System
(d) Radwaste Area Ventilation System
(e) Containment Purges

TABLE 4.3-13 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power)

3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power)

3. Instrument indicates a downscale failure. (Indication on instrument drawer in Control Equipment Room only for 1R16)

4. Instrument controls not set in operate mode. (Applicable to 1-R16 only)

(3) The initial CHANNEL CALIBRATION was performed using appropriate liquid or gaseous calibration sources obtained from reputable suppliers. The activity of the calibration sources were reconfirmed using a multi-channel analyzer which was calibrated using one or more NBS standards.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and

2. Four volume percent oxygen, balance nitrogen.

* At all times

** During waste gas holdup system operation.

TABLE 4.11-1 (Continued)

TABLE NOTATION

b. 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 to assure representative sampling.

c. The principal gamma emitters for which the LLD specification applies exclusively are 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 detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

d. 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.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

* The LLD for Ce-144 shall be 2×10^{-6} uCi/ml.

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined in Table 4.11.1

b. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144* for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

c. Sampling and analysis shall also be performed following shutdown, startup or a THERMAL POWER change (exceeding 15 percent of RATED THERMAL POWER within one hour) unless:

1. Analysis shows that the DOSE EQUIVALENT I-131 concentrations in the primary coolant has not increased more than a factor of three.

2. The noble gas activity monitor shows that effluent activity has not exceeded the monitor "warning"setpoint.

d. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area whenever spent fuel is in the spent fuel pool.

f. 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 Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

*The LLD for Ce-144 shall be 2×10^{-6} uCi/ml.

TABLE 4.11-2 (Continued)

TABLE NOTATION

g. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change (exceeding 15 percent of RATED THERMAL POWER in one hour) and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not exceeded the monitor "warning" setpoint.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% 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 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits 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 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

c. The provision of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of 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 oxygen monitor required OPERABLE by Table 3.3-13.

SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3. The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive waste to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3 and 3.0.4, and 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3. The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spend resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFCATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2. 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. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation.

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 in Specification 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.11.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via) the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. 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 lieu of a Licensee Event Report and pursuant to Specification 6.9.1.11, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Q in lieu of the garden census. Specifications for broadleaf vegetation sampling in Table 3.12-1.4c shall be followed, including analysis of control samples.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification 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. 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.

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of a uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

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 total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 47 psig and an air temperature of 271°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 143.7 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.05 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plant," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.*** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same 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 (Figure 5.1-3) during the report period. 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. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most 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. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CRF Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

***In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee 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.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

(4) - Manual SSPS functional input check every 18 months.

(5) - Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - If not performed in the previous 24 hours, conduct a functional test of the Manual Reactor Trip Switches (using voltmeters).

(10) - If not performed in the previous 24 hours, conduct a functional test of:

- Reactor Trip Breaker UV Trip (via SSPS)
- Reactor Trip Breaker Shunt Trip (via manual pushbutton controls)

(11) - Perform a functional test of:

- Reactor Trip Breaker UV Trip (via SSPS) and conduct response time testing of UV/Breakers (event recorders)
- Reactor Trip Breaker Shunt Trip (via manual pushbutton controls)

(12) - Perform periodic maintenance on Reactor Trip Breakers and Reactor Trip Bypass Breakers semiannually as follows:

a. response time testing, (3 times) (visicorder) trend data
b. trip bar lift force measurements
c. UV output force measurement
d. dropout voltage check
e. servicing/lubrication/adjustments (See Table 3.3-1 Notation ###)
f. repeat testing steps (a-d) following any necessary actions at step (e)

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line (2-R18)	1	26
b. Steam Generator Blowdown Line (2-R19 A, B, C, and D)	4	27
2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Containment Fan Coolers - Service Water Line (2-R13 A, B, C) Discharge	3	28
b. Chemical Waste Basin Line (R37)	1	28
3. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	29
b. Steam Generator Blowdown Line	4	29
4. TANK LEVEL INDICATING DEVICES		
a. Temporary Outside Storage Tanks as Required	1	30

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line (2-R18)	D	P#	R(3)	Q(1)
b. Steam Generator Blowdown Line (2-R19 A, B, C, and D)	D	M'	R(3)	Q(1)
2. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Containment Fan Coolers - Service Water Line (2-R13 A, B, C) Discharge	D	M	R(3)	Q(2)
b. Chemical Waste Basin Line (R37)	D	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	N.A.
b. Steam Generator Blowdown Line	D(4)	N.A.	R	N.A
4. TANK LEVEL INDICATING DEVICES**				
a. Temporary Outside Storage Tanks as Required	D*	N.A.	R	Q

TABLE 4.3-12 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power) (Automatic Isolation only)

3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power) (Indication only)

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION was performed using appropriate liquid or gaseous calibration sources obtained from reputable suppliers. The activity of the calibration sources were reconfirmed using a multi-channel analyzer which was calibrated using one or more NBS standards.

(4) 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.

* During liquid additions to the tank.

** If tank level indication is not provided, vertification will be done by visual inspection.

\# The R18 channel is an off-line channel which requires periodic decontamination. Any count rate indication above 10,000 cpm constitutes a CHANNEL CHECK for compliance purposes.

TABLE 4.3-13 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power) (Automatic Isolation only)

3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels at or above the alarm/trip setpoint.

2. Circuit failure. (Loss of Power) (Indication only)

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION was performed using appropriate liquid or gaseous calibration sources obtained from reputable suppliers. The activity of the calibration sources were reconfirmed using a multi-channel analyzer which was calibrated using one or more NBS standards.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and

2. Four volume percent oxygen, balance nitrogen.

* At all times

** During waste gas holdup system operation.

TABLE 4.11-1 (Continued)

TABLE NOTATION

b. 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 to assure representative sampling.

c. The principal gamma emitters for which the LLD specification applies exclusively are 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 detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

d. 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.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

* The LLD for Ce-144 shall be 2×10^{-6} uCi/ml.

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined in Table 4.11.1.

b. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144* for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

c. Sampling and analysis shall also be performed following shutdown, startup or a THERMAL POWER change (exceeding 15 percent of RATED THERMAL POWER within one hour) unless:

   1. Analysis shows that the DOSE EQUIVALENT I-131 concentrations in the primary coolant has not increased more than a factor of three.

   2. The noble gas activity monitor shows that effluent activity has not exceeded the monitor "warning" setpoint.

d. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area whenever spent fuel is in the spent fuel pool.

f. 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 Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

*The LLD for Ce-144 shall be 2×10^{-6} uCi/ml.

TABLE 4.11-2 (Continued)

TABLE NOTATION

g. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change (exceeding 15 percent of RATED THERMAL POWER in one hour) and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not exceeded the monitor "warning" setpoint.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% 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 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits 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 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

c. The provision of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of 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 oxygen monitor required OPERABLE by Table 3.3-13.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification 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. 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.

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of a uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

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 total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is conistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2. 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. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation.

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 in Specification 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.11.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via) the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. 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 lieu of a Licensee Event Report and pursuant to Specification 6.9.1.11, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Q in lieu of the garden census. Specifications for broadleaf vegetation sampling in Table 3.12-1.4c shall be followed, including analysis of control samples.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 47 psig and an air temperature of 271°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 143.7 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.05 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1 of the FSAR, with alowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,811 $\pm$ 100 cubic feet at a nominal T_{avg} of 581.0°F.

Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plant," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.*** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same 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 (Figure 5.1-3) during the report period. 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. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most 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. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CRF Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

***In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee 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.