ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

conditions for operation defined in Section 3, and (2) it has been tested periodically in accordance with Section 4 and meets its performance requirements.

E. Protective Instrumentation Logic

1. Analog Channel

An arrangement of components and modules as required to generate a single protective action digital signal when required by a unit condition. An analog channel loses its identity when single action signals are combined.

2. Logic Channel

A logic channel is a group of relay contact matrices which operate in response to the digital output signal from the analog channel to generate a protective action signal.

F. Degree of Redundancy

The difference between the number of operable channels and the minimum number of channels monitoring a specific parameter which when tripped will cause an automatic system trip.

G. Instrumentation Surveillance

1. Channel Check

The qualitative assessment of channel behavior during operation by observation. This determination shall include , where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation on channels measuring the same parameter.

2. Channel Functional Test

   Injection of a simulated signal into an analog channel as close to the sensor as practicable or makeup of the logic combinations in a logic channel to verify that it is operable, including alarm and/or trip initiating action.

3. Channel Calibration

   Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarm, or trip, and shall be deemed to include the channel functional test.

4. Source Check

   A source check shall be qualitative assessment of radiation monitor response when the channel sensor is exposed to a radioactive source.

H. Containment Integrity

Containment integrity is defined to exist when:

1. All non-automatic containment isolation valves, except those required for intermittent operation in the performance of normal operational activities, are locked closed and under administrative control. Non-automatic containment isolation valves may be opened intermittently

for operational activities provided that they are under administrative control and are capable of being closed immediately if required.

2. Blind flanges are installed where required.
3. The equipment access hatch is properly closed and sealed.
4. At least one door in the personnel air lock is properly closed and sealed.
5. All automatic containment isolation valves are operable or are locked closed under administrative control.
6. The uncontrolled containment leakage satisfied Specification 4.4.

I. Reportable Occurrence

1. Definition: Refer to Technical Specification 6.6, Station Reporting Requirements for the definitions and examples of the two categories of Reportable Occurrence Reports

   a. Prompt Notification With Written Followup.

   b. Thirty Day Written Reports

K. Low Power Physics Tests

Low power physics tests conducted below 5% of rated power which measure fundamental characteristics of the core and related instrumentation.

L. Fire Suppression Water System

A Fire Suppression Water Systems shall consist of: a water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

M. Offsite Dose Calculation Manual (ODCM)

An Offsite Dose Calculation Manual shall be a manual containing the methodology and parameters to be used in the calculation of offsite dose due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints and the specific monitoring locations of the environmental radiological monitoring program.

N. Dose Equivalent I-131

The dose equivalent I-131 shall be that concentration of I-131 (microcurie/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 thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

O. Gaseous Radwaste Treatment System

A gaseous radwaste treatment system is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

P. Process Control Program (PCP)

The process control program shall contain the current formula, 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 Part 20, 10 CFR Part 71 and Federal and State regulations and other requirements governing the disposal of the waste.

Q. Purge - Purging

Purge or purging is 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.

R. Solidification

Solidification shall be the conversion of wet waste into a form that meets shipping and burial ground requirements.

S. Ventilation Exhaust Treatment System

A ventilation exhaust treatment system is 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 Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

T. Venting

Venting is 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 not provided or required during venting. Vent, used in system names, does not imply a venting process.

U. Site Boundary

The site boundary shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

V. Unrestricted Area

An unrestricted area shall be any area at or beyond the site boundary where access is not controlled by the licensee for purpose 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.

W. Member(s) of the Public

Member(s) of the public shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the license who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

E. <u>Minimum Temperature for Criticality</u>
<u>Specifications</u>

1. Except during low power physics tests, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is more positive than:

   a. + 3pcm/°F at less than 50% of rated power, or

   b. + 3pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

2. In no case shall the reactor be made critical with the reactor coolant temperature below DTT+10°F, where the value of DTT+10°F is as determined in Part B of this specification.

3. When the reactor coolant temperature is below the minimum temperature as specified in E-1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to primary coolant depressurization.

<u>Basis</u>

During the early part of a fuel cycle, the moderator temperature coefficient may be calculated to be slightly positive at coolant temperatures in the power operating range. The moderator coefficient will be most positive at the beginning of cycle life, when the boron concentration in the coolant is the greatest. Later in the cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be less positive or will be negative in the power operating range. At the beginning of cycle life, during pre-operational physics tests, measurements are made to determine that the moderator coefficient is less than +3 pcm/°F in the power operating range.

C. In the event of sub-system instrumentation channel failure permitted by Specification 3.7-B, Tables 3.7-1 through 3.7-3 need not be observed during the short period of time and operable sub-system channel are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

D. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in TS Table 3.7-4.

E. The radioactive liquid and gaseous effluent monitoring instrumentation channels shown in Table 3.7-5(a) and Table 3.7-5(b) shall be operable with their alarm/trip setpoints set to ensure that the limits of Specifications 3.11.A.1 and 3.11.B.1 are not exceeded. The alarm trip setpoints of these channels shall be determined and adjusted in accordance with the Offsite Dose Calculation Manual (ODCM).

1. With a radioactive liquid or gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid or gaseous effluents monitored by the affected channel and declare the channel inoperable or change the setpoint so it is acceptably conservative.
2. With less than the minimum number of radioactive liquid or gaseous effluent monitoring instrumentation channels operable, take the action shown in Table 3.7-5(a) or Table 3.7-5(b). Exert best efforts to return the instruments to operable status within 30 days and, if, unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3. The requirements of Specification 3.0.1 and 6.6.2 are not applicable.

F. The accident monitoring instrumentation for its associated operable components listed in TS Table 3.7-6 shall be operable in accordance with the following:

1. With the number of operable accident monitoring instrumentation channels less than the total number of channels shown in TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 7 days or be in at least hot shutdown within the next 12 hours.

2. With the number of operable accident monitoring instrumentation channels less than the minimum channels operable requirement of TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

monitor indication. The pressurizer safety valves utilize an acoustic monitor channel and a downstream high temperature indication channel. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations".

Radioactive Liquid Effluent Monitoring Instrumentation

The radioactive liquid effluent 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 procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability 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 purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to unrestricted areas.

Radioactive Gaseous Effluent Monitoring Instrumentation

The radioactive gaseous effluent 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 procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability 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.

References

(1) FSAR - Section 7.5

(2) FSAR - Section 14.5

(3) FSAR - Section 14.3.2

(4) FSAR - Section 11.3.3

TABLE 3.7-5(a)
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNEL OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
(a) Liquid Radwaste Effluent Line	1	1
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
(a) Circulating Water Discharge Line	1	2
(b) Component Cooling Service Water Effluent Line	1	2
3. FLOW RATE MEASUREMENT DEVICES		
(a) Liquid Radwaste Effluent Line	1	3

ACTION 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases shall be suspended.

ACTION 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for principal gamma emitters, as defined in TS Table 4.9-1.

ACTION 3 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway shall be suspended.

TABLE 3.7-5(b)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. PROCESS VENT SYSTEM		
(a) Noble Gas Activity Monitor-Providing Alarm and Automatic Termination of Release	1	1
(b) Iodine Sampler	1	2
(c) Particulate Sampler	1	2
(d) Process Vent Flow Rate Measuring Device	1	3
(e) Sampler Flow Rate Measuring Device	1	3
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM		
(a) Hydrogen Monitor	1	4
(b) Oxygen Monitor	1	4
3. CONDENSER AIR EJECTOR SYSTEM		
(a) Gross Activity Monitor	2 (one per unit)	1
(b) Flow Rate Monitor	2 (one per unit)	3
4. VENTILATION VENT SYSTEM		
(a) Noble Gas Activity Monitor	1	1
(b) Iodine Sampler	1	2
(c) Particulate Sampler	1	2
(d) Flow Rate Monitor	1	3
(e) Sampler Flow Rate Monitor	1	3

ACTION - 1 With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this path may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION - 2 With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via the effected path may continue provided samples are continously collected within one hour with auxiliary sampling equipment as required in Table 4.9-2.

ACTION - 3 With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION - 4 With the number of channels operable less than required by the minimum channels operable requirement, operation of this waste gas hold up system may continue provided grab samples are collected at least once per 24 hours and analyzed within the following 4 hours.

3.11 EFFLUENT RELEASE

Applicability:

Applies to the controlled release of radioactive liquids and gases from the station.

Objective:

To establish conditions by which gaseous and liquid waste containing radioactive materials may be released, and to assure that all such releases are within the limits specified in 10 CFR 20. In addition, to assure that the releases of liquid and gaseous radioactive wastes to unrestricted areas are as low as reasonably achievable as set forth in Appendix I to 10 CFR 50.

Specification

A. Liquid Effluents

1. Concentration

a. The concentration of radioactive material released in liquid effluents to unrestricted areas (see figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml.

b. With the concentration of radioactive material released in liquid effluents to unrestricted areas exceeding the above limits, without delay restore the concentration to within the above limits.

c. The surveillance requirements for liquid effluents are given in Table 4.9-1.

d. The reporting requirements of section 6.6.2 are not applicable.

2. Dose

a. The dose or dose commitment to the maximum exposed member of the public from radioactive materials in liquid effluents released, from each reactor unit, to unrestricted areas shall be limited:

(i) During any calender quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to the critical organ, and

(ii) During any calender year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to the critical organ

b. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6, 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.

3. Liquid Radwaste Treatment

a. The Liquid Radwaste Treatment System shall be used to reduce the radioactive materials in liquid waste prior to their discharge when the projected dose due to liquid effluent releases to unrestricted areas (see figure 5.1-1) when averaged over 31 days would exceed 0.06 mrem to the total body or 0.2 mrem to the critical organ.

b. With radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days pursuant to Specification 6.6 a Special Report that includes the following information:

(i) Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or sub-system, and the reason for the inoperability,

(ii) Action(s) taken to restore the inoperable equipment to operable status, and

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

B. Gaseous Effluents

1. Dose Rate

a. The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary (see figure 5.1-1) shall be limited to the following:

(i) For noble gases: less than or equal to 500 mrems/yr. to the total body and less than or equal to 3000 mrems/yr. to the skin, and

(ii) For iodine-131, 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 the critical organ.

b. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

c. The reporting requirements of section 6.6.2 are not applicable.

2. Dose-Noble Gases

a. The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site to areas at and beyond the site boundary (see figure 5.1-1) shall be limited to the following:

(i) During any calender quarter: less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

(ii) During any calender year: less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

b. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6, 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.

3. Dose-I-131, Tritium, and Radionuclides in Particulate Form

a. The dose to the maximum exposed member of the public from all I-131, from tritium, and from all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site to areas at and beyond the site boundary (see figure 5.1-1) shall be limited to the following:

(i) During any calender quarter: less than or equal to 7.5 mrems to the critical organ and,

(ii) During any calender year: less than or equal to 15 mrems to the critical organ.

b. With the calculated dose from the release of I-131, tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the commission within 30 days, pursuant to Specification 6.6, a Special Report that identifies the cause(s) for exceeding the limit 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.

4. Gaseous Radwaste Treatment

a. The appropriate portions of the Gaseous Radwaste Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, from the site to areas at and beyond the site boundary (see Figure 5.1-1) would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation when averaged over 31 days.

b. The Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, from the site to areas at and beyond the site boundary (see Figure 5.1-1) would exceed 0.3 mrem to the critical organ when averaged over 31 days.

c. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6, a Special Report that includes the following information:

(i) Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or sub-systems, and the reason for the inoperability,

(ii) Action(s) taken to restore the inoperable equipment to operable status, and

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

5. <u>Explosive Gas Mixture</u>

a. The concentration of hydrogen or oxygen in the waste gas holdup system shall be limited to less than or equal to 4% by volume.

b. With the concentration of hydrogen or oxygen in the waste gas holdup system exceeding the limit, restore the concentration to within the limit within 48 hours.

6. <u>Gas Storage Tanks</u>

a. The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 24,600 curies of noble gases (considered as Xe-133).

b. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all addition of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

C. Total Dose

1. The annual (calender year) dose or dose commitment to the maximum exposed member of the public due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or the critical organ (except the thyroid, which shall be limited to less than or equal to 75 mrems).

2. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.A.2, 3.11.B.2 or 3.11.B.3, calculations should be made including direct radiation contribution from the reactor units and from outside storage tanks to determine whether the limits of Specification 3.11.C.1 above have been exceeed. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6, a Special Report that defines the corrective action to be taken to reduce subsequent releases 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 Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to the maximum exposed member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calender 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.

D. Radiological Environmental Monitoring

1. Monitoring Program

a. The radiological environmental monitoring program shall be conducted as specified in Table 4.9-3.

b. With the radiological environmental monitoring program not being conducted as specified in Table 4.9-3, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

c. 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 4.9-4 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to

Specification 6.6, 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 the maximum exposed member of the public is less than the calendar year limits of Specifications 3.11.A.2, 3.11.B.2, and 3.11.B.3. When more than one of the radionuclides in Table 4.9-4 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 4.9-4 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to the maximum exposed member of the public is equal to or greater than the calendar year limits of Specifications 3.11.A.2, 3.11.B.2 and 3.11.B.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.

d. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 4.9-3, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program.

In lieu of a Licensee Event Report and pursuant to Specification 6.6, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples 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).

2. Land Use Census

a. 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. (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/Qs in lieu of the garden census.)

b. 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 4.9.C, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.6.

c. 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 3.11.D.1.a, 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 the 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 6.6, 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).

3. Interlaboratory Comparison Program

   a. Analyses shall be performed on radioactive materials (which contain nuclides produced at nuclear power stations) supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission. The Interlaboratory Comparison Program is described in the ODCM.

   b. 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 Specification 6.6.

E. Solid Radioactive Waste

1. Solidification of radioactive waste shall be conducted in accordance with a Process Control Program.
2. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

3. Surveillance requirements for solidification are described in Specification 4.9.K.

F. The requirements of Specifications 3.0.1 and 6.6.2 are not applicable.

Basis

Liquid Effluent Concentration

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. 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 the maximum exposed member of the public and (2) the limits of 10 CFR Part 20.106(e) 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.

Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

Liquid Effluent Dose

This specification is provided to implement the requirements of Sections II.A, 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.A of Appendix I. The Specifications 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 releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculations in the ODCM 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 the maximum exposed 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, "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 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

Liquid Radwaste Treatment

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 specification 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.

Gaseous Effluents Dose Rate

This specification 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. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, 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 the maximum exposed member of the public, either within or outside the site boundary to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For the maximum exposed members of the public, who may at times be within the site boundary the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to less than or equal to 500 mrems/year to the total 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.

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

Dose - Noble Gases

This specification 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 Specifications 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 releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements in section 4.9 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 the maximum exposed member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculations 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 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. The ODCM equations provided for determining the air doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

Dose - I-131, Tritium, and Radionuclides In Particulate Form

This specification 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 Specification 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 releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements in section 4.9 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 the maximum exposed member of the public through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods 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 specifications for I-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent on 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 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.

Gaseous Radwaste Treatment

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 specification 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.

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.

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 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 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.

Total Dose

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have now been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to the maximum exposed member of the public will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to the maximum exposed 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 the maximum exposed 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 Part 20.405c, 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 the limits of 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 Specifications 3.11.A and 3.11.B. An individual is not considered the maximum exposed 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.

Monitoring Program

The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of the maximum exposed 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. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.9-5 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.6.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

Land Use Census

This specification 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 are made if required by the results of this census. The best information from the 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.

Interlaboratory Comparison Program

The requirement for participation in an 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 reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

Solid Radioactive Waste

This specification implements the requirements of 10 CFR 50.36a and General Design Criteria 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, and mixing and curing times, as appropriate.

TABLE [illegible]1 (Continued)

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
10.	Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A	R	N.A.	
14.	Boric Acid Tank Level	*D	R	N.A.	
15.	Refueling Water Storage Tank Level	S	R	M	
16.	Boron Injection Tank Level	W	N.A.	N.A.	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	Reactor Containment Pressure-CLS	*D	R	M(1)	1) Isolation Valve signal and spray signal
19.	Boric Acid Control	N.A.	R	N.A.	
20.	Containment Sump Level	N.A.	R	N.A.	
21.	Accumulator Level and Pressure	S	R	N.A.	
22.	Containment Pressure-Vacuum Pump System	S	R	N.A.	
23.	Steam Line Pressure	S	R	M	

TABLE 4.1-1

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
24.	Turbine First Stage Pressure	S	R	M	
25.	Emergency Plan Radiation Instr.	*M	R	M	
26.	Environmental Radiation Monitors	*M	N.A.	N.A.	TLD Dosimeters
27.	Logic Channel Testing	N.A.	N.A.	M	
28.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	R	
29.	Turbine Trip Setpoint	N.A.	R	R	Stop valve closure or low EH fluid pressure
30.	Seismic Instrumentation	M	R	M	
31.	Reactor Trip Breaker	N.A.	N.A.	M	
32.	Reactor Coolant Pressure (Low)	N.A.	R	N.A.	
33.	Auxiliary Feedwater				
	a. Steam Generator Water Level Low-Low	S	R	M	
	b. RCP Undervoltage	S	R	M	
	c. S.I.	(All Safety Injection surveillance requirements)			
	d. Station Blackout	N.A.	R	N.A.	
	e. Main Feedwater Pump Trip	N.A.	N.A.	R	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
34. LOSS OF POWER				
a. 4.16 KV Emergency Bus undervoltage (Loss of voltage)	N.A.	R	M	
b. 4.16 KV Emergency Bus undervoltage (Degraded voltage)	N.A.	R	M.	

S - Each shift
D - Daily
W - Weekly
NA - Not applicable
SA - Semiannually
Q - Every 90 effective full power days
M - Monthly
P - Prior to each startup if not done previous week
R - Each Refueling Shutdown
BW - Every two weeks
AP - After each startup if not done previous week

* See Specification 4.1D

TABLE 4.1-1(a)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	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	D	PR	R	Q
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
(a) Circulating Water Discharge Line	D	M	R	Q
(b) Component Cooling Service Water System Effluent Line	D	M	R	Q
3. FLOW RATE MEASUREMENT DEVICES				
(a) Liquid Radwaste Effluent Line	D	N.A.	R	N.A.

D - Daily
M - Monthly
R - Each Refueling Shutdown
Q - Quarterly
PR - Prior to each release
N.A. - Not Applicable

TABLE 4.1-1(b)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. PROCESS VENT SYSTEM				
(a) Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release	D	M*	R	Q
(b) Iodine Sampler	W	N.A.	N.A.	N.A.
(c) Particulate Sampler	W	N.A.	N.A.	N.A.
(d) Process Vent Flow Rate Measuring Device	D	N.A.	R	N.A.
(e) Sampler Flow Rate Monitor	D	N.A.	SA	N.A.
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
(a) Hydrogen Monitor	D	N.A.	Q(1)	M
(b) Oxygen Monitor	D	N.A.	Q(2)	M
3. CONDENSER AIR EJECTOR SYSTEM				
(a) Gross Activity Monitor	D	M	R	Q
(b) Flow Rate Monitor	D	N.A.	R	N.A.
4. VENTILATION VENT SYSTEM				
(a) Noble Gas Activity Monitor	D	M	R	Q
(b) Iodine Sampler	W	N.A.	N.A.	N.A.
(c) Particulate Sampler	W	N.A.	N.A.	N.A.
(d) Flow Rate Monitor	D	N.A.	R	N.A.
(e) Sampler Flow Rate Monitor	D	N.A.	SA	N.A.

(1) - The channel calibration shall include the use of standard gas samples containing a nominal:
1. one volume percent hydrogen, balance nitrogen, and
2. four volume percent hydrogen, balance nitrogen.

(2) - 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.

D - Daily
W - Weekly
M - Monthly
R - Each Refueling Shutdown
SA - Semi-annually
NA - Not Applicable
Q - Quarterly
* - Monthly and prior to each Waste Gas Decay Tank Release

4.9 EFFLUENT SAMPLING AND RADIATION MONITORING SYSTEM

Applicability

Applies to the periodic monitoring and recording of radioactive effluents.

Objective

To ascertain that radioactive releases are maintained as low as practicable and within the limits set forth in 10 CFR 20 and 10 CFR 50 Appendix I.

Specification

A. All radiation monitor channels shall be checked, calibrated and tested as indicated in Tables 4.1-1(a) and 4.1-1(b).

B. Radioactive liquid waste shall be sampled and analyzed according to the sampling and analyses program of Tables 4.9-1.

The results of the radioactivity analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.A.1.a.

C. Cumulative dose contributions from liquid and gaseous effluents (including noble gases, I-131, tritium and radionuclides in particulate form) shall be determined in accordance with the ODCM at least once per 31 days.

D. Doses due to liquid and gaseous releases shall be projected at least once per 31 days in accordance with the ODCM.

E. The dose rate due to noble gases in gaseous effluents shall be determined continuously to be within the limits of Specification 3.11.B.1 in accordance with the methods and procedures of the ODCM.

The dose rate due to Iodine-131, Tritium, and all radionuclides in particulate form with half life greater than 8 days, in gaseous effluents shall be determined to be within the limits of Specification 3.11.B.1 in accordance with the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.9-2.

F. The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the limits of Specification 3.11.B.5 by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen or oxygen monitors required operable by Table 3.7-5(b) of Specification 3.7.E.

G. The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.11.B.6 at least once per month when radioactive materials are being added to the tank.

H. The radiological environmental monitoring samples shall be collected pursuant to Table 4.9-3 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 4.9-3, the detection capabilities required by Table 4.9-5.

I. The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.6.

J. A summary of the results obtained as part of the Interlaboratory Comparison Program required in Specification 3.11.D.3 shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.6.

K. 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 radioactive waste (i.e. wet radioactive waste as defined in the PCP).

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 solidification parameters determined by the Process Control Program.

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 3 consecutive initial test specimens demonstrate solidification. The Process Control Program shall be modified as required, as provided in Specification 6.8.A, to assure solidification of subsequent batches of waste.

TABLE 4.9-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (uCi/ml)
A. Batch Releases[b]	PR Each Batch	PR Each Batch	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
	PR One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	PR Each Batch	M Composite[d]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	PR Each Batch	Q Composite[d]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
B. Continuous Releases[e]	Continuous[f]	W Composite[f]	Principal Gamma Emitters[f]	$5x10^{-7}$
			I-131	$1x10^{-6}$
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	Continuous[f]	M Composite[f]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continuous[f]	Q Composite[f]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

W - Weekly
M - Monthly
Q - Quarterly
PR - Prior to each release
NA - Not Applicable

TABLE 4.9-1 (Continued)

TABLE NOTATION

[a] The LLD is defined, for purposes of this specification, 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):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcuries per unit mass or volume),

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

E is the counting efficiency (as counts per disintegration),

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

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

TABLE 4.9-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an 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.

[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 appropriate methods will be used to obtain representative sample for analysis.

[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, at levels exceeding the LLD, 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.

[f]To be representative of the quantities and concentrations of radioactive materials in liquid effluents, composite sampling shall employ appropriate methods which will result in a speciman representative of the effluent release.

TABLE 4.9

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (uCi/ml)
A. Waste Gas Storage Tank	PR Each Tank Grab Sample	PR Each Tank	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment Purge	PR Each Purge Grab Sample	PR Each Purge	Principle Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C. Process and Ventilation Vent	W[c] Grab Sample	W[c]	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
D. Condenser Air Ejector	W[c] Grab Sample	W[c]	Principle Gamma Emitter[b]	1×10^{-4}
			H-3	1×10^{-6}
			I-131	1×10^{-12}
E. Release Types as Listed in A, B, C Above	Continuous[d]	W[e] Charcoal Sample	I-131	1×10^{-12}
	Continuous[d]	W[e] Particulate Sample	Principle Gamma Emitters[b]	1×10^{-11}
	Continuous[d]	W Composite Particulate Sample	Gross Alpha	1×10^{-11}

TABLE 4.9[illegible] (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (uCi/ml)
E. Release Types as Listed in A, B, C Above (Continued)	Continuous[d]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[d]	Noble Gas Monitor	Noble Gases Gross Beta & Gamma	$1x10^{-6}$

W - Weekly
Q - Quarterly
PR - Prior to each release

TABLE 4.9-2 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of this specification, 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):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcuries per unit mass or volume),

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

E is the counting efficiency (as counts per disintegration),

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

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

TABLE 4.9-2 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an *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.

[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-135m, and 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 nuclides with half life greater than 8 days, that are measurable and identifiable at the level above LLD, together with the above nuclides, shall also be identified and reported.

[c]Sampling and analyses shall also be performed following shutdown, startup, or a thermal power change exceeding 15 percent of the rated thermal power which occurs within a one hour period. When (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

[d]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.B.1, 3.11.B.2 and 3.11.B.3.

[e]Samples shall be changed at least once per week and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per day for at least 1 week 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 1 day are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement applies only if (1) analysis shows that the DOSE EQUIVALENT I-131 Concentration in the primary coolant has increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has increased more than a factor of 3.

TABLE 4.9-3

ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLE AND SAMPLE LOCATION	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. DIRECT RADIATION	About 40 Routine Monitoring stations to be placed as follows: 1) Inner Ring in general area of site boundary with station in each sector. 2) Outer Ring 6 to 8 km from the site with a station in each sector. 3) The balance of the 8 dosimeters should be placed in special interest areas such as population centers nearby residents, schools, and in 2 or 3 areas to serve as controls.	QUARTERLY	Gamma Dose QUARTERLY

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLE AND SAMPLE LOCATION	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. AIRBORNE			
Radioiodines and Particulates	Samples from 7 locations:	Continuous Sampler operation with sample collection weekly	<u>Radioiodine Cannister</u>
			I-131 Analysis Weekly
	a) 1 sample from close to the site boundary location of the highest calculated annual average groundlevel D/Q.		<u>Particulate Sampler</u>
	b) 5 sample locations 6-8 km distance located in a concentric ring around station.		Gross beta radio - activity analysis following filter change;
	c) 1 sample from a control location 15-30 km distant, providing valid background data.		Gamma isotopic analysis of composite (by location) quarterly
3. WATERBORNE			
a) Surface	a) 1 sample upstream	Monthly Sample	Gamma isotopic analysis monthly;
	b) 1 sample downstream		Composite for tritium analysis quarterly.
b) Ground	Sample from 1 or 2 sources	Quarterly	Gamma isotopic and tritium analysis quarterly
c) Sediment from shoreline	1 sample from downstream area with existing or potential recreational value	Semi-Annually	Gamma isotopic analysis semi-annually
d) Silt	5 samples from vicinity of the station	Semi-Annually	Gamma isotopic analysis semi-annually

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLE AND SAMPLE LOCATION	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. INGESTION			
a) Milk	a) 4 samples from milking animals in the vicinity of station.	Monthly	Gamma isotopic and I-131 analysis monthly
	b) 1 sample from milking animals at a control location (15-30 km distant)		
b) Fish and Invertebrates	a) 3 sample of oysters in the vicinity of the station.	Bi-Monthly	Gamma isotopic on edibles
	b) 5 samples of clams in the vicinity of the station.	Bi-Monthly	Gamma isotopic on edibles
	c) 1 sampling of crabs from the vicinity of the station.	Annually	Gamma isotopic on edibles
	d) 2 samples of fish from the vicinity of the station (catfish, white perch, eel)	Semi-Annually	Gamma isotopic on edibles
c) Food Products	a) 1 sample corn	Annually	Gamma isotopic on edible portion
	b) 1 sample soybean	Annually	Gamma isotopic on edible portion
	c) 1 sample peanuts	Annually	Gamma isotopic on edible portion

TABLE 4.9-4

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	30,000				
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	

TABLE 4.9-5

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	0.01				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	10	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	60			60		
La-140	15			15		

Note: 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.

TABLE 4.9-5 (Continued)

TABLE NOTATION

[a]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

[b]Table 4.9-5 indicates acceptable detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs). The LLD is defined, for purposes of these specifications, 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):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as picocuries per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute), E is the counting efficiency (as counts per disintegration),

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

is the radioactive decay constant for the particular radionuclide, and

t for environmental samples is the elapsed time between sample collection (or end of the sample collection period) and time of counting

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

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.6.3.b.

5.0 DESIGN FEATURES

5.1 SITE

Applicability

Applies to the location and boundaries of the site for the Surry Power Station.

Objective

To define those aspects of the site which will affect the overall safety of the installation.

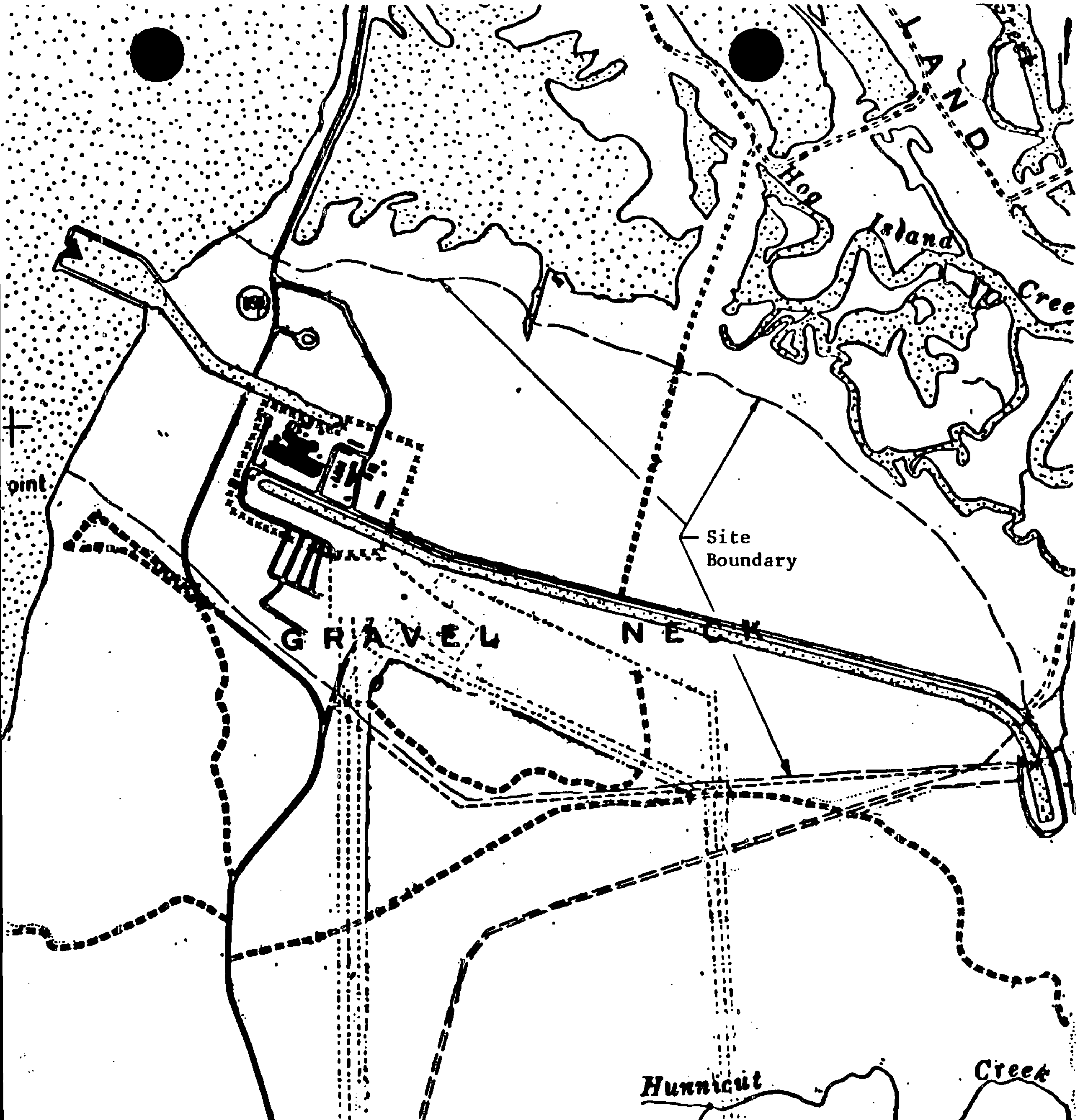
Specification

The Surry Power Station is located in Surry County, Virginia, on property owned by Virginia Electric and Power Company on a point of land called Gravel Neck which juts into the James River. It is approximately 46 miles SE of Richmond, Virginia, 17 miles NW of Newport News, Virginia, and 25 miles NW of Norfolk, Virginia. The minimum distance from a reactor centerline to the site exclusion boundary as defined in 10CFR100 is 1,650 ft. This is the distance for Unit 1, which is controlling. A map of the site is shown in TS Figure 5.1-1.

References

FSAR section 2.0 Site

FSAR Section 2.1 General Description



Gaseous Release
1. Process Vent- 131 ft.
2. Vent-Vent Stacks-considered ground level release

▲ Liquid leaves site

xxxxx Security Fence- Area outside is unrestricted for gaseous effluents

Land Maximum Individual Occupancy within site boundary:
1) Canal Bank Fishing= 160 hr/yr

Liquid Maximum Individual Occupancy within site boundary:
1) Boat Fishing Discharge Canal = 800 hr/yr

Figure 5.1-1

Map Defining Unrestricted Areas for Radioactive Gaseous and Liquid Effluents

9. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

10. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

11. Review of every unplanned onsite release of radioactive material to the environs exceeding the limits of Specification 3.11, including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the chairman of the Station Nuclear Safety and Operating Committee.

12. Review of changes to the Process Control Program and the offsite Dose Calculation Manual.

g. Authority

The SNSOC shall:

1. Recommend to the Station Manager written approval or disapproval of items considered under (1) through (4) above.
2. Render determinations in writing with regard to whether or not each item considered under (1) through (5) above constitutes an unreviewed safety question.
3. Provide written notification within 24 hours to the Vice President-Nuclear Operations and the Director-Safety Evaluation and Control of disagreement between SNSOC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1-A above.

h. Records

The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Vice President-Nuclear Operations and to the Director-Safety Evaluation and Control.

6. The Station Security Plan and implementing procedures at least once per 24 months.

7. Any other area of facility operation considered appropriate by the Executive Manager-Quality Assurance or the Senior Vice President-Power Operations.

8. The Station Fire Protection Program and implementing procedures at least once per 24 months.

9. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

10. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

11. The radiological environmental monitoring program at least once per 12 months.

12. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

13. The Process Control Program and implementing procedures for processing and packaging of radioactive waste at least once per 24 months.

b. Authority

The Quality Assurance Department shall report to and advise the Executive Manager-Quality Assurance, who shall advise the Senior Vice President-Power Operations on those areas of responsibility specified in 6.1.C.3.a above.

f. Entrance to areas with radiation levels in excess of 1 R/hr shall require the use of the "buddy system," whereby a minimum of two individuals maintain continuous visual and/or verbal communication with each other; or other mechanical and/or electrical means to provide constant communication with the individual in the area shall be provided.

g. A Radiation Work Permit system shall be used to authorize and control any work performed in high radiation areas.

h. All buildings or structures, in or around which a high radiation area exists, shall be surrounded by a chain-link fence. The entrance gate shall be locked under administrative control, or continuously guarded to preclude unauthorized entry.

i. Stringent administrative procedures shall be implemented to assure adherence to the restriction placed on the entrance to a high radiation area and the radiation protection program associated thereto.

2. Written procedures shall be established, implemented and maintained covering the activities referenced below:

   a. Process Control Program implementation.

   b. Offsite Dose Calculation Manual implementation.

C. All procedures described in A and B above, and changes thereto, shall be reviewed by the Station Nuclear Safety and Operating Committee and approved by the Station Manager prior to implementation.

radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

3. Unique Reporting Requirements

a. In-service Inspection Evaluation. Special summary technical report shall be submitted to the Director of Reactor Licensing, Office of Nuclear Reactor Regulation, NRC, Washington, D. C. 20555, after five (5) years of operation. This report shall include an evaluation of the results of the in-service inspection program and will be reviewed in light of the technology available at that time.

b. Annual Radiological Environmental Operating Report.[1]

Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports 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, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.11.D.2.a.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all measurements taken during the period pursuant to the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. 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 centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.11.D.3.a; and discussion of all analyses in which the LLD required by Table 4.9-5 was not achievable.

c. Semiannual Radioactive Effluent Release Report[1]

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," 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. This report shall include an assessment of the radiation doses to the maximum exposed members of the public due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. Annual meteorological data collected over the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This meteorological data shall be retained in a file on site and shall be made available to the NRC upon request. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in the

Offsite Dose Calculation Manual (ODCM). The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

If the dose to the maximum exposed member of the public due to the radioactive liquid and gaseous effluents from the station during the previous calendar year exceeds twice the limits of Specification 3.11.A.2, 3.11.B.2, or 3.11.B.3, the dose assessment shall include the contribution from direct radiation. The dose to the maximum exposed member of the public shall show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The Radioactive Effluent Release Reports shall include a list of unplanned releases exceeding the limits of Specifications 3.11A.1.a and 3.11.B.1.a from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.11.D.2.a.

c. With no fire suppression water system operable, within 24 hours; notify the Commission outlining the action taken and the plans and schedule for restoring the system to operable status.

d. With redundant fire suppression water system component inoperable for more than 14 days, submit a Special Report to the Commission within the next 10 days outlining the cause of inoperability and the plans for restoring the component to operable status.

e. With the CO_2 fire protection system inoperable for more than 14 days, submit a Special Report to the Commission within the next 10 days outlining the cause of inoperability and the plans for restoring the system to operable status.

f. With the Records Vault halon fire protection system inoperable for more than 14 days, submit a Special Report to the Commission within the next 10 days outlining the cause of inoperability and the plans for restoring the system to operable status.

g. In the event that the Reactor Vessel Overpressure Mitigating System is used to mitigate a RCS pressure transient, submit a Special Report to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or the administrative controls on the transient and any corrective action necessary to prevent recurrence.

FOOTNOTES

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

2. This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

6.8 Process Control Program and Offsite Dose Calculation Manual

A. Process Control Program (PCP)

Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SNSOC.

2. Shall become effective upon review and acceptance by the SNSOC.

B. Offsite Dose Calculation Manual (ODCM)

Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable by the SNSOC.

2. Shall become effective upon review and acceptance by the SNSOC.

6.9 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS:

A. Licensee initiated major changes to the radioactive solid waste systems:

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by SNSOC. The discussion of each change shall contain:

    a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59.

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

    c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

    d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or in quantity of solid waste that differs from those previously predicted in the license application and amendments thereto;

    e. An evaluation of the change, which shows the expected maximum exposures to an individual in the unrestricted area that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by SNSOC.

2. Shall become effective upon review and acceptance by SNSOC.

ATTACHMENT 2

JUSTIFICATIONS FOR DEVIATING FROM

NUREG-0472 REVISION 3

1. The Radiological Effluent Technical Specifications for Surry Power Station are based upon the Revision 3 of NUREG-0472 dated March 30, 1982. The information applicable to the Surry Power Station has been included in the format of the existing Surry Technical Specifications. Although there might be some deviations from the text of the NUREG-0472, the intent of the same has been met in most of the cases.

2. The existing format of the Table of Contents is used to reflect RETS.

3. All the appropriate definitions have been included in the Surry RETS.

4. In Tables 3.7-5(a) and 4.1-1(a), the Turbine Building Sumps Effluent Line is not included because it is not a normal radioactive effluent pathway. The potential for unplanned and unmonitored releases of radioactivity to the environment from the contamination of normally non-radioactive systems was reviewed by the station staff in response to I. E. Bulletin No. 80-10. Subsequent NRC inspections have also addressed the above issue.

   As a result, Surry has committed to installing composite samplers and flow rate recorders in the storm drain effluents, to which the turbine building sumps empty. This equipment has been installed and will be maintained per station procedures.

   The presence of the samplers limit the possibility of a potential accidental release, discharging from the station without adequately quantifying its magnitude. The daily analysis of the storm drain's hourly composite sample, indicate that this system is a non-contaminated effluent pathway and not subject to normal station radioactive effluents.

   In addition, Surry has taken steps to eliminate normally radioactive systems from discharging into the turbine building sumps. A recent design change re-routed the Piping Tunnel sump effluents into the Radwaste System, thus precluding turbine building sump contamination.

5. The Steam Generator Blowdown Effluent Line is not included in Tables 3.7-5(a) and 4.1-1(a) because it is not an effluent pathway. It is a closed loop system, recirculated through the condensate polisher. Also, Radioactivity Recorders are deleted because they do not provide alarm/trip set point.

6. The flow monitoring instrumentation for the Discharge Canal requested in the surveillance Table 4.1-1(a), is deleted. Pump curves will be used to determine flow.

7. Tank Level Indicating Devices are not included in Table 4.1-1(a). The outside tanks contain overflow protection that discharges to the liquid waste system.

8. Action Statements in Table 3.7-5(a) meet the intent of the Action Statments in NUREG-0472.

9. The channel functional test is not applicable to the Liquid Waste Flow Measuring Device in Table 4.1-1(a), nor the Process Vent and Vent-Vent Flow Rate Measuring Devices in Table 4.1-1(b), because they have no alarm/trip function. The channel check, performed daily, verifies operability of the channel.

10. Channel functional test as defined in the Surry Technical Specification describes the current capability of the effluent monitoring system. The montoring system would require a lengthy backfit that is not deemed to be beneficial. In addition, a channel check is performed every shift by an operator to determine the monitors operability.

11. In Tables 3.7-5(b) and 4.1-1(b), the nomenclature has been changed to describe Surry's effluent monitoring systems. The Waste Gas Holdup System has been changed to Process Vent System, Condenser Evacuation System to Condenser Air Ejector System and Auxiliary Building Ventilation System to Ventilation Vent System.

    The Containment Purge System and the Radwaste Area Ventilation System are part of the Ventilation Vent System. The Steam Generator Blowdown Vent System is no longer in use at Surry.

12. The Limiting Condition of Operation for the Liquid Hold-up Tanks has been deleted due to overflow protection of outside tanks and the absence of potable water supply downstream of station effluents.

13. Refueling Canal and Spent Fuel Pool Area exhaust is discharged to the Ventilation Vent System. The H-3 sampling frequency for this system is stated in Table 4.9-2 of the submittal.

14. I-133 is not included in the monitoring and dose calculations. An assessmemnt of prior releases from Surry Power Station, including the four years from 1979 to 1982, indicates that I-133 release quantities on an annual basis contribute less than 1% to the radioactive pathway doses. A summary of the data is listed below. In an interagency NRC memorandum on Provisions for I-133 in RETS, dated November 29, 1982, the primary concern was in effluent release from BWR's.

DATE	Ci(I-131)	Ci(I-133)	DOSE(I-131)	DOSE(I-133)	DOSE(I-133)
1979	6.17E-3	7.82E-5	1.61E-2	1.89E-6	0.012%
1980	1.68E-2	1.76E-3	4.37E-2	4.24E-5	0.097%
1981	4.50E-2	1.65E-2	1.17E-1	3.98E-4	0.34%
1982	5.71E-2	1.07E-2	1.48E-1	2.58E-4	0.17%

15. The surveillance requirements on the Waste Gas Decay Tanks have been changed to once per month when the specific activity of the coolant is $<$2.20E+3 uCi/gm dose equivalent Xe-133 and once per day when the specific activity is $\geq$ 2.20E+3 uCi/gm dose equivalent Xe-133. At Surry, waste gas is removed from the reactor coolant letdown by the Boron Recovery System. The gases are transferred to the Waste Gas Decay System without processing by the Catalytic Recombiners.

The recombiners, used to strip hydrogen from primary gases, is not a functional system at Surry. Without the hydrogen removal system, hydrogen becomes the major component of the primary offgas. The remaining components of the offgas consist of nitrogen, oxygen and trace amounts of noble gases.

The analysis of the Boron Recovery System by the Surry Environmental Applicants Report and the Update Final Safety Analysis Report, assume that 90-99% of the primary offgas is composed of hydrogen. In an analysis to determine the maximum curie content in the Waste Gas Decay Tank (see Attachment 3) a conservative figure was used that assumes 10% of the primary offgas is composed of radioactive noble gases. The Waste Gas Decay Tank, which stores the primary offgases, is also assumed to contain a maximum of 10% by volume, radioactive noble gases.

The curie limit in the Waste Gas Decay Tank is specified in Technical Specification 3.11.B.6 as 24,600 curies. To meet or exceed this curie limit, assuming 90% hydrogen and 10% noble gas content, the primary coolant activity, processed by the Boron Recovery System, would need to exceed 2278 uCi/ml (Xe-133 equivalent). The specified primary coolant activity should remain at that level throughout the entire waste gas tank filling cycle. Station records show that none of the Waste Gas Decay Tanks releases since Surry's initial criticality, contained more than 2% of the curie content stated in specification 3.11.B.6.

Surveillance tied to the activity of the influent stream should provide a good indication as to when the activity in the tank is approaching specification limit.

16. Since there is no downstream drinking water supply, the default values have been used in Tables 4.9-4 and 4.9-5 wherever applicable.

17. Solid radioactive waste released from site is reported as outlined in Regulatory Guide 1.21 as stated in Specification 6.6.3.c.

18. In Table 4.9-5, Ba-La-140 is separated into Ba-140 and La-140 and Zr-Nb-95 is separated into Zr-95 and Nb-95 to comply with Radiological Assessment Branch Technical Position, Revision 1, November 1979.

19. Surry has an established environmental monitoring program and established methods so that the table notation for Table 4.9-3 is not necessary.

20. The Quality Assurance Program is not included in Specification 6.4.B.2 because a Station QA program, with written procedures exists at the station. Scheduled audits, quality control, and document control of the Health Physics programs are addressed in the Nuclear Power Station Quality Assurance Manual. More specifically, the environmental program, which encompasses radiological and non-radiological controls, is audited on a scheduled basis with surveillance audits in the interim.

ATTACHMENT 3

ATTACHMENT 3

A. Subject: Justification for the establishment of the surveillance requirements on the Waste Gas Decay Tanks at Surry Power Station.

B. References:

1. Surry Power Station Updated Final Safety Analysis Report, Chapter 11.

2. Surry Power Station Environmental Applicants Report.

3. Surry Power Station, Radiological Effluent Technical Specification Submittal.

C. Introduction:

The Surry Power Station RETS submittal (reference 3) states: "The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 24,600 curies of noble gases (considered as Xe133)".

To verify that the decay tanks do not exceed the above limit, a justification for the surveillance requirements is presented which considers the physical limitations of the tanks design pressure and design volume, and the assumed composition of the primary offgases.

D. Approach:

Reference 1 and 2 describe the Boron Recovery Systems stripping of primary gases from the reactor coolant. Also described in these references is the processing of the primary offgas to ensure that the hydrogen level in the offgas is maintained below the explosive mixture level.

Within the description of this processing, assumptions were made that approximate the normal and maximum hydrogen concentration in the primary offgas.

Assuming that this primary gas is transferred from the Boron Recovery System to an empty Waste Gas Decay Tank and the transfer of this gas continues until the tank is full, then a primary gas activity was determined that would give the total tank activity of 24,600 curies.

No assumption of radiogas decay was taken into account.

Assuming 100% efficiency of Boron Recovery System to strip primary gases, the activity determined above would be the action level, when reached in reactor coolant sampling, that would require a daily sampling of the Waste Gas Decay Tanks.

E. Assumptions:

1. All of the noble gases and 0.1% of the radioiodines in the letdown are removed at the gas stripper and transferred to the Waste Gas Decay Tank.

2. No radioactive decay is assumed for the gases stripped from the primary system.

3. Waste Gas Decay Tank volume is 434 $ft.^3$.

4. Maximum pressure 115 psig.

5. 10% of primary offgas is composed of noble gases (Xe133,

   - Reference 2, page A-12.

   - Reference 1, page 11.2-26.

   - Reference 1, Table 11.2-1.

F. Calculations:

1. Tank volume:

$$\text{Volume} = \frac{(434\ ft^3)\ 129.7\ psia}{14.7\ psia} = 3.83 \times 10^3\ ft^3 = \underline{1.08 \times 10^8 mls}$$

2. Noble Gas volume:

$$1.08 \times 10^8 mls \times 0.10 = \underline{1.08 \times 10^7 mls}$$

3. Reactor Coolant Activity:

$$\frac{24{,}600\ \text{Curies}}{1.08 \times 10^7\ \text{mls}} \times 10^6\ uCi/Ci = \underline{2.278 \times 10^3}\ \frac{uCi}{ml}$$

Attachment 4