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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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June 19, 1984

Docket Nos. 50-280
and 50-281

Posted
Amdt. 96
to DPR-37

Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 97 to Facility Operating License No. DPR-32 and Amendment No. 96 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated May 4, 1983, as supplemented September 23, 1983, and January 11 and February 3, 1984.

These amendments revise the Technical Specifications to incorporate the requirements of Appendix I of 10 CFR Part 50 as Radiological Effluent Technical Specifications (RETS).

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Joseph D. Neighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 97 to DPR-32
2. Amendment No. 96 to DPR-37
3. Safety Evaluation

cc: w/enclosures
See next page

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Units 1 and 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated May 4, 1983, as supplemented September 23, 1983, and January 11 and February 3, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 97, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented July 1, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 19, 1984

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated May 4, 1983, as supplemented September 23, 1983, and January 11 and February 3, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 96 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented July 1, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 19, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 97 FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 96 FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
iii	iii
1.0-3	1.0-3
1.0-4	1.0-4
1.0-5	1.0-5
1.0-7	1.0-7
-----------	1.0-8
-----------	1.0-9
-----------	1.0-10
3.1-17a	3.1-17a
3.1-18	3.1-18
3.7-2	3.7-2
-----------	3.7-2a
3.7-9	3.7-9
-----------	3.7-9a
-----------	3.7-20a
-----------	3.7-20b
3.10-4	3.10-4
3.11-1 thru 3.11-6a	3.11-1 thru 3.11-23
4.1-7	4.1-7
4.1-8	4.1-8
-----------	4.1-8a
-----------	4.1-8b
-----------	4.1-8c
4.9-1 thru 4.9-5	4.9-1 thru 4.9-15
5.1-1	5.1-1
-----------	Figure 5.1-1
6.1-8	6.1-8
-----------	6.1-8a
6.1-14	6.1-14
-----------	6.1-14a
6.4-3	6.4-3
6.6-10 thur 6.6-14	6.6-10 thru 6.6-14
6.6-16a	6.6-17
-----------	6.8-1
-----------	6.8-2
-----------	6.9-1
-----------	6.9-2

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. <u>Gaseous Radwaste Treatment System</u>

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. <u>Process Control Program (PCP)</u>

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. <u>Purge - Purging</u>

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. <u>Solidification</u>

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.

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/cc) 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 tyroid does 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".

E. Minimum Temperature for Criticality Specifications

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.

Basis

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 Monitor	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) Air Ejector Flow Rate Measuring Device	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) Ventilation Vent Flow Rate Monitor	1	3
(e) Sampler Flow Rate Measuring Device	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.

12. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

13. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

14. Two trains of the control and relay room emergency ventilation system shall be operable. With one train inoperable for any reason, demonstrate the other train is operable by performing the test in Specification 4.20.A.1. With both trains inoperable comply with Specification 3.10.B.

15. Containment purge shall be filtered through the high efficiency particulate air filters and charcoal absorbers.

B. If any one of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

C. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

D. The requirements of 3.0.1 are not applicable.

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. <u>Dose-I-131, Tritium, and Radionuclides in Particulate Form</u>

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. Explosive Gas Mixture

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. Gas Storage Tanks

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. <u>Radiological Environmental Monitoring</u>

1. <u>Monitoring Program</u>

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. <u>Land Use Census</u>

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 a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated 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 4.1-1 (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

Amendment No.97 and No.96

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

TABL -1(b)
RADIOACTIVE GASEOUS EFFLUENT MONITORIN 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 Monitor	D	N.A.	R	N.A.
(e) Sampler Flow Rate Measuring Device	D	N.A.	S.A.	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) Air Ejector Flow Rate Measuring Device	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) Ventilation Vent Flow Rate Monitor	D	N.A.	R	N.A.
(e) Sampler Flow Rate Measuring Device	D	N.A.	S.A.	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 the specific activity of the primary reactor coolant is ≤ 2200 μCi/gm dose equivalent Xe-133. Under the conditions which result in a specific activity > 2200 μCi/gm dose equivalent Xe-133, the Waste Gas Decay Tanks shall be sampled once per day.

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.

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]	$1x10^{-4}$
B. Containment Purge	PR Each Purge Grab Sample	PR Each Purge	Principle Gamma Emitters[b]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. Process and Ventilation Vent	W[c] Grab Sample	W[c]	Principal Gamma Emitters[b]	$1x10^{-4}$
			H-3	$1x10^{-6}$
D. Condenser Air Ejector	W[c] Grab Sample	W[c]	Principle Gamma Emitter[b]	$1x10^{-4}$
			H-3	$1x10^{-6}$
E. Containment Hog /Depressurization	PR Grab Sample	PR Each Release	Principal Gamma Emitters	$1x10^{-4}$
			H-3	$1x10^{-6}$
	Continuous[d]	Charcoal[f] Sample	I - 131	$1x10^{-11}$
	Continuous[d]	Particulate[f] Sample	Principle Gamma Emitters[b]	$1x10^{-10}$
	Continuous[d]	Composite[f] Particulate Sample	Gross Alpha	$1x10^{-10}$

TABLE 4.9 (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. Containment Hog /Depressurization (Continued)	Continuous[d]	Composite[f] Particulate Sample	Sr - 89, Sr - 90	$1x10^{-10}$
F. Release Types as Listed in A, B, C Above	Continuous[d]	W[e] Charcoal Sample	I-131	$1x10^{-12}$
	Continuous[d]	W[e] Particulate Sample	Principle Gamma Emitters[b]	$1x10^{-11}$
	Continuous[d]	W Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	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.

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

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: a) 1 sample from close to the site boundary location of the highest calculated annual average groundlevel D/Q. b) 5 sample locations 6-8 km distance located in a concentric ring around station. c) 1 sample from a control location 15-30 km distant, providing valid background data.	Continuous Sampler operation with sample collection weekly	Radioiodine Cannister I-131 Analysis Weekly Particulate Sampler Gross beta radio - activity analysis following filter change; Gamma isotopic analysis of composite (by location) quarterly
3. WATERBORNE			
a) Surface	a) 1 sample upstream b) 1 sample downstream	Monthly Sample	Gamma isotopic analysis monthly; 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

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):

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

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 _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated 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

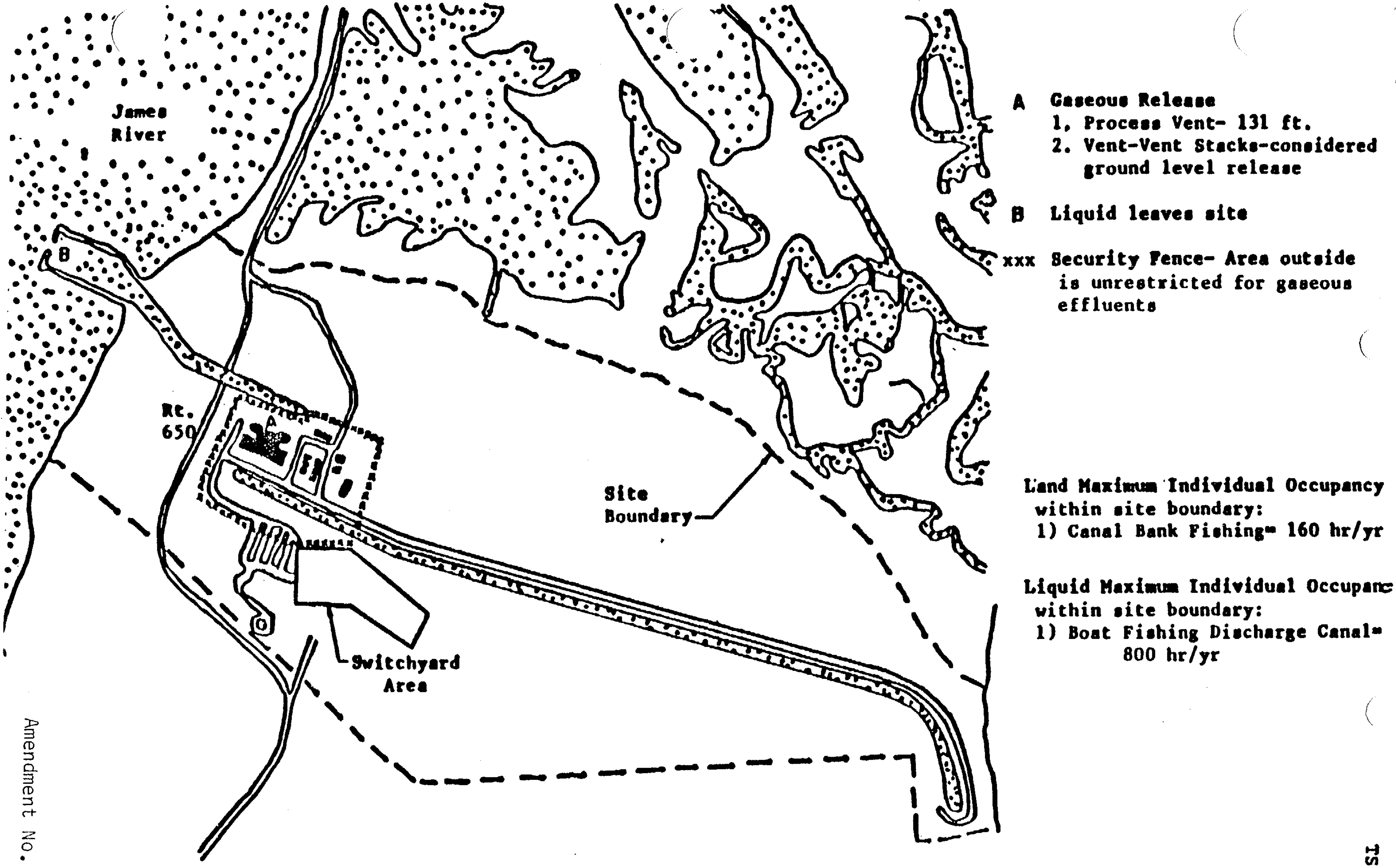

Figure 5.1-1

Map Defining Unrestricted Areas for Radioactive Gaseous and Liquid Effluents

0 ¼mile
Scale

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. Semi-Annual 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 Tables 1, 2, and 3 of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year 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 waste systems:

1. Shall be reported to the Commission in the Semi-Annual 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;

---

*Major changes to radioactive waste treatment systems may be reported to the Commission in the annual update to the FSAR in lieu of reporting changes in the Semi-Annual Radioactive Effluent Release Report.

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.

NUCLEAR REGULATORY COMMISSI
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. DPR-32
AND AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. DPR-37
VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, the Virginia Electric and Power Company has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. The Virginia Electric and Power Company made application with the Commission by letter dated May 4, 1983, as supplemented September 23, 1983, and January 11 and February 3, 1984, which requested changes to the Technical Specifications appended to Facility Operating License Nos. DPR-32 and DPR-37 for Surry Power Station Unit Nos. 1 and 2. The proposed Technical Specifications update those portions of the Technical Specifications addressing radioactive waste management and make them consistent with the current staff positions as expressed in NUREG-0472. These revised Technical Specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR Parts 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64; and with 10 CFR Part 50, Appendix B.

2.0 BACKGROUND AND DISCUSSION

2.1 Regulations

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Section 50. 36a, "Technical Specifications on Effluents from Nuclear Power Reactors", provides that each license authorizing operation of a nuclear power reactor will include technical specifications that (1) require compliance with applicable provisions of Part 20.106, "Radioactivity in Effluents to Unrestricted Areas"; (2) require that operating procedures developed for the control of effluents be established and followed; (3) require that equipment installed in the radioactive waste system be maintained and used; and (4) require the periodic submission of reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents, any quantities of radioactive materials released that are significantly above design objectives, and such other information as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

10 CFR Part 20, "Standards for Protection Against Radiation," paragraphs 20.105(c), 20.106(g), and 20.405(c), require that nuclear power plant and other licensees comply with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded.

10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases for radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage; and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

10 CFR Part 50, Appendix B, establishes quality assurance requirements for nuclear power plants.

10 CFR Part 50, Appendix I, Section IV, provides guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0472 provides radiological effluent technical specifications for pressurized water reactors which the staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of certain key values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent technical specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment sytems.

The above NUREG documents address all of the radiological effluent technical specifications needed to assure compliance with the guidance and requirements provided by the regulations previously cited. However, alternative approaches to the preparation of radiological effluent technical specifications and alternative radiological effluent technical specifications may be acceptable if the staff determines that the alternatives are in compliance with the regulations and with the intent of the regulatory guidance.

The standard radiological effluent technical specifications can be grouped under the following categories:

(1) Instrumentation

(2) Radioactive effluents

(3) Radiological environmental monitoring

(4) Design features

(5) Administrative controls.

Each of the specifications under the first three categories is comprised of two parts: the limiting condition for operation and the surveillance requirements. The limiting condition for operation provides a statement of the limiting condition, the times when it is applicable, and the actions to be taken in the event that the limiting condition is not met.

In general, the specifications established to assure compliance with 10 CFR Part 20 standards provide, in the event the limiting conditions of operation are exceeded, that without delay conditions are restored to within the limiting conditions. Otherwise, the facility is required to effect approved shutdown procedures. In general, the specifications established to assure compliance with 10 CFR Part 50 provide, in the event the limiting conditions of operation are exceeded, that within specified times corrective actions are to be taken, alternative means of operation are to be employed, and certain reports are to be submitted to the NRC describing these conditions and actions.

The specifications concerning design features and administrative controls contain no limiting conditions of operation or surveillance requirements.

Table 1 indicates the standard radiological effluent technical specifications that are needed to assure compliance with the particular provisions of the regulations described in Section 1.0.

3.0 EVALUATION

The enclosed report (TER-C5506-113/114) was prepared for us by Franklin Research Center (FRC) as part of our technical assistance contract program. Their report provides their technical evaluation of the compliance of the Licensee's May 4, 1983 submittal with NRC provided criteria. The staff has reviewed this TER and agrees with the evaluation.

The proposed radiological effluent technical specifications for Surry Power Station Unit Nos. 1 and 2 have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0472 (the Surry Power Station is comprised of two pressurized water reactors) and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

The proposed changes will not remove or relax any existing requirement needed to provide reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Tab Relation Between Provisions of the Regulations and the dard Radiological Effluent Technical Specifications for Pr urized Water Rea
and Boiling Water Reactors

● Indicate the specifications that are needed to assure compliance with the identified provision of the regulations.

Provisions of Title 10 Code of Federal Regulations	Standard Radiological Effluent Technical Specifications																												
	Instrumentation		Radioactive Effluents																Rad. Envir. Monitoring			Design Features	Administrative Control						
			Liquid				Gaseous																						
							PWR/BWR				PWR		BWR																
	Rad. Liquid Effl. Monitoring	Rad. Gas. Effl. Monitoring	Effluent Concentration	Dose	Liquid Radwaste Treatment	Liquid Holdup Tanks	Dose Rate	Dose Noble Gases	Dose I-131, Trit., and Part.	Explosive Gas Mixture	Gaseous Radwaste Treatment	Gas Storage Tanks	Gaseous Radwaste Treatment	Ventilation Exhaust Treatment	Main Condenser	Mark I or II Containment	Solid Radioactive Waste	Total Dose	Rad. Env. Monitoring Program	Land Use Census	Interlab. Comparison Program	Site Boundaries*	Review and Audits	Procedures	Reports	Record Retention	Process Control Program	Offsite Dose Calc. Manual	Major Changes to Rad. Systems
§ 50.36a Technical specifications on effluents from nuclear power reactors																													
Remain within limits of § 20.106			●			●	●					●				●												●	
Establish and follow procedures to control effluents			●	●	●	●	●	●	●	●	●	●	●	●	●	●	●							●				●	
Maintain and use radioactive waste system equipment				●	●			●	●		●		●	●														●	
Submit reports, semi-annual and other	●	●	●	●	●	●	●	●	●	●	●	●	●	●			●	●									●	●	●
§§ 20.105(c), 20.106(g), 20.405(c) Compliance with 40 CFR 190																		●	●	●	●							●	
Part 50 Appendix A - General Design Criteria																													
Criterion 60 - Control of releases of radioactive materials to the environment	●	●			●	●				●	●	●	●	●	●	●	●							●			●	●	
Criterion 61 - Fuel storage and handling and radioactivity control	●	●																											
Criterion 63 - Monitoring fuel and waste storage	●	●																											
Criterion 64 - Monitoring radioactivity releases	●	●													●				●									●	
Part 50 Appendix B - Quality Assurance Criteria	●	●																			●		●	●			●		
Part 50 Appendix I - Guides to Meet "As Low As Is Reasonably Achievable (ALARA)"																													
Maintain releases within design objectives				●	●			●	●		●			●														●	
Establish surveillance & monitoring program to provide data on:																													
(1) quantities of rad. matls. in effluents	●	●																											
(2) radiation & rad. matls. in the environment																			●						●	●			
(3) changes in use of unrestricted areas																				●									
Exert best efforts to keep releases "ALARA"				●	●			●	●		●		●	●															●
Submit report if calculated doses exceed the design objective				●				●	●		●			●											●				
Demonstrate conform. to des. obj. by calc. proced.				●				●	●		●			●														●	
Part 100															●														

*Note: Needed to fully implement other specifications.

4.0 Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area. The staff has determined that the amendment involves no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 19, 1984

Principal Contributor:

W. Meinke

TECHNICAL EVALUATION REPORT

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION IMPLEMENTATION (A-2)

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNITS 1 AND 2

NRC DOCKET NO. 50-280, 50-281

NRC TAC NO. 8102, 8103

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 4

FRC TASKS 113, 114

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June 15, 1983

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of Surry Power Station Units 1 and 2 with regard to Radiological Effluent Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM).

The evaluation uses criteria proposed by the NRC staff in the Model Technical Specifications for pressurized water reactors (PWRs), NUREG-0472 [1]. This effort is directed toward the NRC objective of implementing RETS which comply principally with the regulatory requirements of the Code of Federal Regulations, Title 10, Part 50 (10CFR50), "Domestic Licensing of Production and Utilization Facilities," Appendix I [2]. Other regulations pertinent to the control of effluent releases are also included within the scope of compliance.

1.2 GENERIC BACKGROUND

Since 1970, 10CFR50, Section 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," has required licensees to provide technical specifications which ensure that radioactive releases will be kept as low as reasonably achievable (ALARA). In 1975, numerical guidance for the ALARA requirement was issued in 10CFR50, Appendix I. The licensees of all operating reactors were required [3] to submit, no later than June 4, 1976, their proposed ALARA Technical Specifications and information for evaluation in accordance with 10CFR50, Appendix I.

However, in February 1976, the NRC staff recommended that proposals to modify Technical Specifications be deferred until the NRC completed the model RETS. The model RETS deals with radioactive waste management systems and environmental monitoring. Although the model RETS closely parallels 10CFR50, Appendix I requirements, it also includes provisions for addressing other issues.

These other issues are specifically stipulated by the following regulations:

- o 10CFR20 [4], "Standards for Protection Against Radiation," Paragraphs 20.105(c), 20.106(g), and 20.405(c) require that nuclear power plants and other licensees comply with 40CFR190 [5], "Environmental Radiation Protection Standards for Nuclear Power Operations," and submit reports to the NRC when the 40CFR190 limits have been or may be exceeded.

- o 10CFR50, Appendix A [6], "General Design Criteria for Nuclear Power Plants," contains Criterion 60 - Control of releases of radioactive materials to the environment; Criterion 63 - Monitoring fuel and waste storage; and Criterion 64 - Monitoring radioactivity releases.

- o 10CFR50, Appendix B [7], establishes the quality assurance required for nuclear power plants.

The NRC position on the model RETS was established in May 1978 when the NRC's Regulatory Requirements Review Committee approved the model RETS: NUREG-0472 for PWRs [1] and NUREG-0473 [8] for boiling water reactors (BWRs). Copies were sent to licensees in July 1978 with a request to submit proposed site-specific RETS on a staggered schedule over a 6-month period. Licensees responded with requests for clarifications and extensions.

The Atomic Industrial Forum (AIF) formed a task force to comment on the model RETS. NRC staff members first met with the AIF task force on June 17, 1978. The model RETS was subsequently revised to reflect comments from the AIF and others. A principal change was the transfer of much of the material concerning dose calculations from the model RETS to a separate ODCM.

The revised model RETS was sent to licensees on November 15 and 16, 1978 with guidance (NUREG-0133 [9]) for preparation of the RETS and the ODCM and a new schedule for responses, again staggered over a 6-month period.

Four regional seminars on the RETS were conducted by the NRC staff during November and December 1978. Subsequently, Revision 2 of the model RETS and additional guidance on the ODCM and a Process Control Program (PCP) were issued in February 1979 to each utility at individual meetings. In response to the NRC's request, operation reactor licensees have subsequently submitted initial proposals on plant RETS and the ODCM. Review leading to ultimate implementation of these documents was initiated by the NRC in 1981 using subcontracted independent teams as reviewers.

As the RETS review process has progressed since September 1981, feedback from the licensees has led the NRC to believe that modification to some provisions in the current version of Revision 2 is needed to better clarify specific concerns of the licensees and thus expedite the entire review process. Starting in April 1982, NRC distributed revised versions of RETS in draft form to the licensees during the site visits. The new guidance on these changes was presented in the AIF meeting on May 19, 1982 [10]. Some interim changes regarding the Radiological Environmental Monitoring Section were issued in August 1982 [11]. With the incorporation of these new changes, NRC issued, in September 1982, a draft version of NUREG-0472, Revision 3 [12], to serve as new guidance for the review teams.

1.3 PLANT-SPECIFIC BACKGROUND

In response to the NRC's request, the Licensee, Virginia Electric and Power Company (VEPCO), submitted a RETS proposal dated March 15, 1979 [13] on behalf of Surry Power Station Units 1 and 2, which was followed by a submittal of the ODCM [14].

In an initial evaluation by the Franklin Reseach Center (FRC), an independent review team, the Licensee's RETS and ODCM submittals were evaluated against the model RETS (NUREG-0472) and assessed for compliance with the stipulated provisions. Copies of the draft review, dated April 26, 1982 [15, 16], were delivered to the NRC and the Licensee prior to a site visit by the reviewers.

The site visit was conducted on May 12-14, 1982 by the reviewers with the participation of plant personnel and the NRC staff. Discussions focused on the initial review of the proposed changes to the RETS and on the technical approaches for an ODCM. The deficiencies in the Licensee's proposed RETS were considered, deviations from NRC guidelines were pointed out, many differences were clarified, and only a few items remained unresolved pending justification by the Licensee. These issues are summarized in Reference 17.

On November 12, 1982, the Licensee submitted an updated draft version of both the RETS [18] and ODCM [19]. The submittals were reviewed by the reviewer (FRC), and deficiencies were identified to the NRC [20]. These deficiencies were transmitted to the Licensee with comments by the NRC staff [21].

The final versions of the Surry RETS [22] and ODCM [23], dated May 4, 1983, were submitted to the NRC and transmitted to the FRC reviewers together with justifications provided by the Licensee. However, the submittal did not include the PCP. The submitted documents were subsequently reviewed. Final evaluation of RETS was detailed in a comparison report [24] which used NUREG-0472, Draft Revision 3 [12] to evaluate the Licensee's submittal. The comparison report also incorporates NRC comments [25] which serve as additional guidelines regarding plant-specific issues.

2. REVIEW CRITERIA

Review criteria for the RETS and ODCM were provided by the NRC in three documents:

NUREG-0472, RETS for PWRs

NUREG-0473, RETS for BWRs

NUREG-0133, Preparation of RETS for Nuclear Power Plants.

Twelve essential criteria are given for the RETS and ODCM:

1. All significant releases of radioactivity shall be controlled and monitored.
2. Offsite concentrations of radioactivity shall not exceed the 10CFR20, Appendix B, Table II limits.
3. Offsite radiation doses shall be ALARA.
4. Equipment shall be maintained and used to keep offsite doses ALARA.
5. Radwaste tank inventories shall be limited so that failures will not cause offsite doses exceeding 10CFR20 limits.
6. Hydrogen and/or oxygen concentrations in the waste gas system shall be controlled to prevent explosive mixtures.
7. Wastes shall be processed to shipping and burial ground criteria under a documented program, subject to quality assurance verification.
8. An environmental monitoring program, including a land-use census, shall be implemented.
9. The radwaste management program shall be subject to regular audits and reviews.
10. Procedures for control of liquid and gaseous effluents shall be maintained and followed.
11. Periodic and special reports on environmental monitoring and on releases shall be submitted.
12. Offsite dose calculations shall be performed using documented and approved methods consistent with NRC methodology.

Subsequent to the publication of NUREG-0472 and NUREG-0473, the NRC staff issued guidelines [26, 27], clarifications [28, 29], and branch positions [30, 31, 32, 33] establishing a policy that requires the licensees of operating reactors to meet the intent, if not the letter, of the model RETS provisions. The NRC branch positions issued since the RETS implementation review began have clarified the model RETS implementation for operating reactors.

Review of the ODCM was based on the following NRC guidelines: Branch Technical Position, "General Content of the Offsite Dose Calculation Manual" [34]; NUREG-0133 [9]; and Regulatory Guide 1.109 [35]. The ODCM format is left to the licensee and may be simplified by tables and grid printouts.

Since the Licensee has not submitted a PCP, the review does not include this specific document.

3. TECHNICAL EVALUATION

3.1 GENERAL DESCRIPTION OF RADIOLOGICAL EFFLUENT SYSTEMS

This section briefly describes the liquid and gaseous radwaste effluent systems, release paths, and control systems installed at Surry Power Station Units 1 and 2; both are PWRs.

3.1.1 Radioactive Liquid Effluent

The liquid waste treatment system for the Surry plant is common to both Units 1 and 2 (Figure 1). Two systems currently exist for treating liquid wastes. These are the boron recovery system and the liquid waste treatment system. The boron recovery system treats effluents collected in primary drain tanks and letdown from the primary coolant that is diverted from the chemical and volume control system (CVCS). The liquid waste treatment system processes the liquid waste originating from containment, auxiliary building, and decontamination building sumps, and from laboratory drains. These effluents are all released in batches, which join to form the liquid radwaste effluent line, providing ultimate discharges to the bay via the discharge tunnel.

Other liquid lines that also lead to the discharge tunnel for release are the service water and the condensate polishing chemical waste, which are discharged on a continuous basis.

Other than the above effluent release pathways, the turbine building (floor drain) sump discharges effluents directly to the station storm drain system.

3.1.2 Radioactive Gaseous Effluent

The gaseous waste treatment system for the Surry plant is also common to both Units 1 and 2 (Figure 2).

The process effluent from the CVCS is stored in the decay tank, from which the effluents are released with other streams into the process vent,

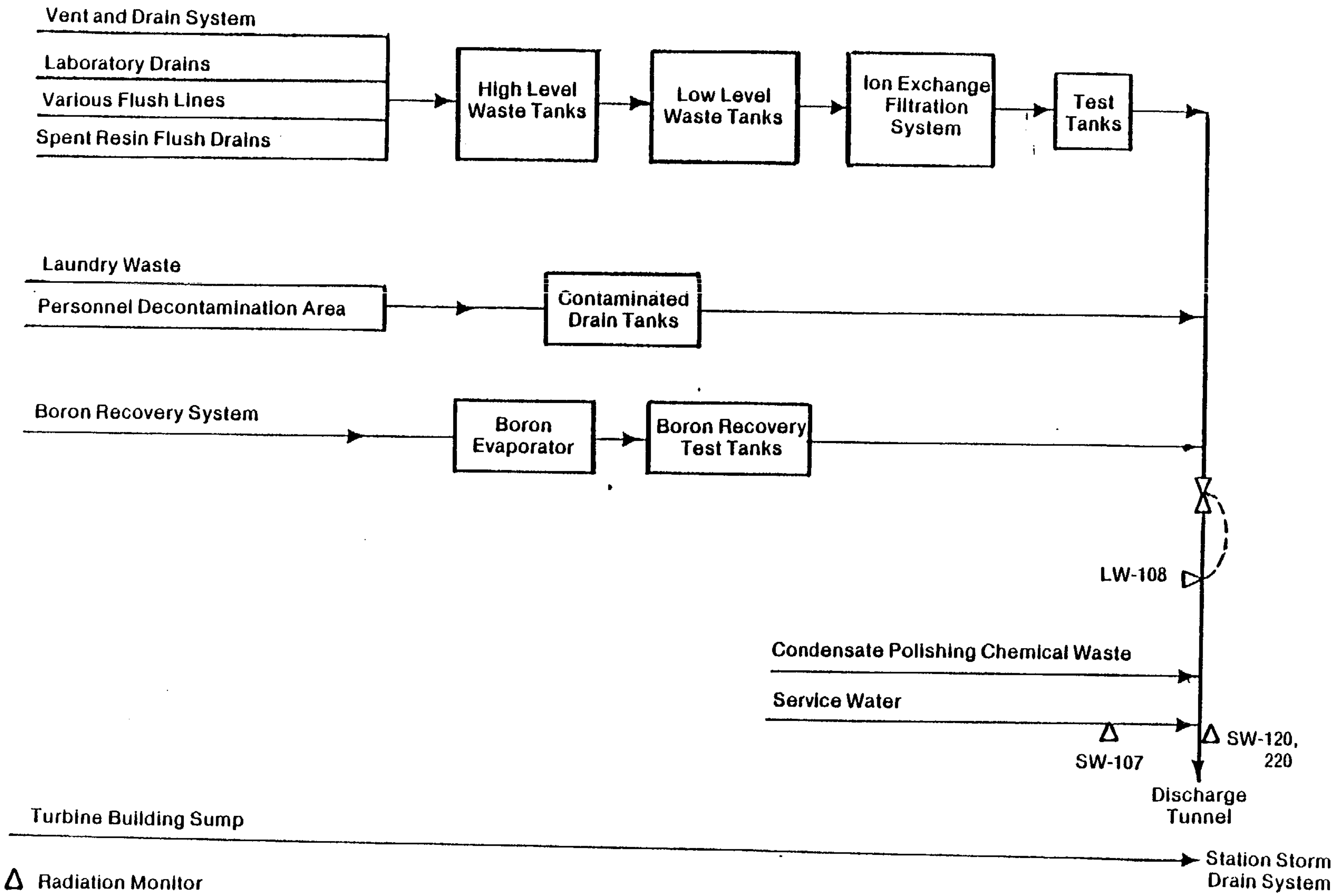

Figure 1. Liquid Radwaste Treatment Systems, Effluent Paths, Turbine Building Sump, Station Storm Drain System, and Controls for Surry Power Station Units 1 and 2

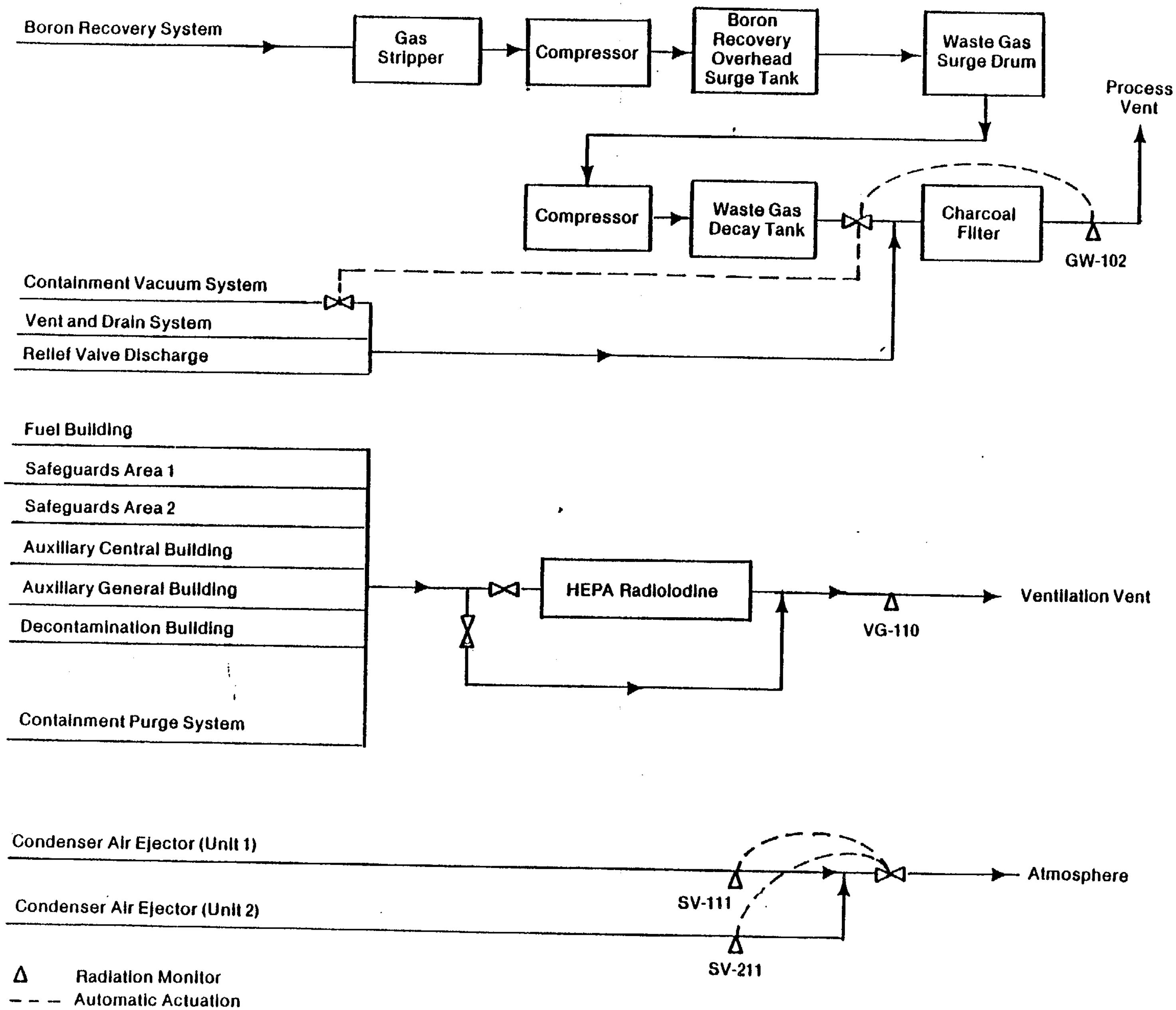

Figure 2. Gaseous Radwaste Treatment Systems, Effluent Paths, and Controls for Surry Power Station Units 1 and 2

where releases are considered as mixed mode. Substreams such as the containment vacuum system, the vent and drain system, and the relief valve discharge also lead to the process vent for release.

The ventilation vent system services the containment purge system, the auxiliary building, the decontamination building, and the fuel building. Effluents from the condenser air ejectors (Units 1 and 2) are discharged separately to the atmosphere. Both releases from the ventilation vent system and the air ejectors are considered at ground level.

The steam generator blowdown does not form an effluent pathway since it is a closed loop system and is recirculated through the condensate polisher for processing.

3.2 RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

The evaluation of the Licensee's proposed RETS against the provisions of NUREG-0472 included the following:

- a review of information provided by the Licensee in the 1979 proposed submittals [13, 14]
- resolution of problem areas in that submittal by means of a site visit [17]
- review of the Licensee's November 12, 1982 draft submittals [18, 19]
- review of the Licensee's May 4, 1983 final submittals [22, 23].

3.2.1 Effluent Instrumentation

The objective of the RETS with regard to effluent instrumentation is to ensure that all significant liquid and gaseous effluent releases are monitored. The RETS specify that all effluent monitors be operable and that alarm/trip setpoints be determined in order to ensure that radioactive levels do not exceed the maximum permissible concentration (MPC) set by 10CFR20. To further ensure that the instrumentation functions properly, surveillance requirements are also needed in the specifications.

3.2.1.1 Radioactive Liquid Effluent Monitoring Instrumentation

A radiation monitor (LW-108) has been installed for the liquid radwaste effluent line (Figure 1), which provides automatic isolation in the event that an excessive level of radioactivity is detected. The Licensee has also provided a monitor (SW-107) for the service water and two monitors (SW-120, 220) for the discharge tunnel.

The Licensee has committed to install composite samplers and flow rate recorders in the storm drain effluents, to which the turbine building sumps discharge. The Licensee indicated that such equipment has been installed and will be maintained according to station procedures. However, this monitoring equipment is not included in the Licensee's RETS submittal. The Licensee has provided justification that the turbine building sumps effluent line is not a normal radioactive effluent pathway. Further, steps have been taken to eliminate normally radioactive systems from discharging into the turbine building sump. A recent design change, for instance, has rerouted the piping tunnel sump effluents into the radwaste system, thus precluding the turbine building sump contamination.

It is determined that the Licensee's proposed RETS submittal and supporting justifications on liquid effluent monitoring instrumentation have satisfied the provisions set forth in the model RETS and thus meet the intent of NUREG-0472.

3.2.1.2 Radioactive Gaseous Effluent Monitoring Instrumentation

The plant process vent is provided with a monitoring system capable of monitoring noble gases, iodines, and particulates. The noble gas monitor (GW-102) has the capability of automatically isolating the releases from the waste gas decay tank and the containment vacuum system. Radiation monitors are also installed at the condenser air ejectors; monitor SV-111 for Unit 1 and monitor SV-211 for Unit 2 both have the automatic isolation capability on the discharges. Radiation monitor VG-110 monitors the effluent releases through the ventilation vent.

The proposed monitoring capabilities provided by the Licensee meet the intent of NUREG-0472 for radioactive gaseous effluent monitoring instrumentation.

3.2.2 Concentrations and Dose Rates of Effluents

3.2.2.1 Liquid Effluent Concentration

In Section 3.11.A.1 of the Licensee's submittal, a commitment is made to maintain the concentration of radioactive liquid effluents released from the site to the unrestricted areas to within 10CFR20 limits, and if the concentration of liquid effluents to the unrestricted area exceeds these limits, it will be restored without delay to a value equal to or less than the MPC values specified in 10CFR20. Both batch and continuous releases are sampled and analyzed periodically in accordance with a sampling and analysis program (Table 4.9-1 of the Licensee's submittal), which meets the intent of NUREG-0472.

3.2.2.2 Gaseous Effluent Dose Rate

In Section 3.11.B.1 of the Licensee's submittal, a commitment is made to maintain the offsite gaseous dose rate from the site to areas at and beyond the site boundary to within 10CFR20 limits, and if the concentration of gaseous effluents exceeds these limits or the equivalent dose values, it will be restored without delay to a value equal to or less than these limits.

The radioactive gaseous waste sampling and analysis program (Table 4.9-2 of the Licensee's submittal) provides adequate sampling and analysis of the vent discharges, including the substreams, and therefore meets the intent of NUREG-0472.

3.2.3 Offsite Doses from Effluents

The objective of the RETS with regard to offsite doses from effluents is to ensure that offsite doses are kept ALARA, are in compliance with the dose specifications of NUREG-0472, and are in accordance with 10CFR50, Appendix I, and 40CFR190. The Licensee has made a commitment to (1) meet the quarterly and yearly dose limitations for liquid effluents, per Section 3.11.1.2 of NUREG-0472 [1]; (2) restrict the air doses for beta and gamma radiation in unrestricted areas as specified in 10CFR50, Appendix I, Section II.B; (3) maintain the dose level to the maximally exposed member of the public from releases of iodine-131, tritium, and particulates with half-lives greater than 8 days within the design objectives of 10CFR50, Appendix I, Section II.C; and (4) limit the annual dose to the maximally exposed member of the public due to releases of radioactivity and radiation from uranium fuel cycle sources to within the requirements of 40CFR190. This satisfies the intent of NUREG-0472.

3.2.4 Effluent Treatment

The objective of the RETS with regard to effluent treatment is to ensure that wastes are treated to keep releases ALARA and to satisfy the provisions for technical specifications governing the maintenance and use of radwaste treatment equipment. The Licensee has made a commitment to use the liquid and gaseous radwaste treatment system when the projected doses averaged over 31 days exceed 25% of the annual dose design objectives, prorated monthly. The Licensee has also made a commitment to use the ventilation exhaust treatment system if the monthly projected dose exceeds the limits prescribed in NUREG-0472. This meets the intent of 10CFR50, Appendix I, Section II.D. The Licensee has also made a commitment to project the monthly dose in accordance with the ODCM. This also meets the intent of NUREG-0472.

3.2.5 Tank Inventory Limits

The objective of the RETS with regard to tank inventory limits is to ensure that the rupture of a radwaste tank would not cause offsite doses greater than the limits set in 10CFR20 for nonoccupational exposure. Citing the overflow protection of outside tanks and the absence of potable water supply downstream of station effluents, the Licensee does not anticipate the necessity of having such a specification for liquid storage tanks. For gas storage tanks, a curie limit of 24,600 curies has been set for noble gases which are considered to be represented by xenon-133. The Licensee has proposed to perform surveillance of the gas storage tank at least once per month when radioactive materials are being added to the tank. The proposed surveillance frequency is less frequent than the once per 24 hours specified by the model RETS; however, the Licensee stated in the cover letter of the submittal [22] that the frequency will be increased to once per day when the specific activity of the coolant is greater than or equal to 2.20×10^3 µci/gm dose equivalent xenon-133. The Licensee's justification and commitment to comply with tank inventory limits have satisfied the intent of NUREG-0472.

3.2.6 Explosive Gas Mixtures

The objective of the RETS with regard to explosive gas mixtures is to prevent hydrogen explosions in the waste gas systems. The Licensee has stated that the waste gas holdup system is designed to withstand a hydrogen/oxygen explosion. The Licensee has made a commitment to maintain a safe concentration in this system. The Licensee has also proposed a hydrogen monitor and an oxygen monitor to fulfill this commitment, which is consistent with the provisions of NUREG-0472.

3.2.7 Solid Radwaste System

The objective of the RETS with regard to the solid radwaste system is to ensure that radwaste will be properly processed and packaged before it is shipped to a burial site, in accordance with 10CFR71 and Specification 3.11.3 of NUREG-0472. The Licensee has made a commitment to establish a PCP to show compliance with this objective. The Licensee has provided assurance that 10CFR20 requirements will also be met, thereby satisfying the intent of NUREG-0472.

3.2.8 Radiological Environmental Monitoring Program

The objectives of the RETS with regard to environmental monitoring are to ensure that (1) an adequate full-area-coverage (land and water inclusive) monitoring program exists; (2) the requirements of 10CFR50, Appendix I for technical specifications on environmental monitoring are satisfied; and (3) the Licensee maintains both a land-use census and interlaboratory comparison program.

In all cases, the Licensee has followed NUREG-0472 guidelines, including the Branch Technical Position dated November 1979 [31], and has provided an adequate number of sample locations for pathways identified. The Licensee's methods of analysis and maintenance of yearly records satisfy the NRC guidelines and meet the intent of 10CFR50, Appendix I. The Licensee has also made a commitment to document the environmental monitoring sample locations in the ODCM, which meets the intent of NUREG-0472. The specification for the land use census satisfies the provisions of Section 3.12.2 of NUREG-0472 by providing for an annual census in the specified areas. The Licensee participates in an interlaboratory comparison program approved by the NRC and reports the results in the Annual Radiological Environmental Operating Report, which also meets the intent of NUREG-0472.

3.2.9 Audits and Reviews

The objective of the RETS with regard to audits and reviews is to ensure that audits and reviews of the radwaste and environmental monitoring programs are properly conducted. The Licensee's administrative structure designates the station nuclear safety and operating committee (SNSOC) and the quality assurance department (QA) as the two groups responsible for reviews and audits, respectively. Their responsibilities also include the ODCM, PCP, and QA program. The two committees encompass the responsibility for reviews and audit; this meets the intent of NUREG-0472.

3.2.10 Procedures and Records

The objective of the RETS with regard to procedures is to satisfy the provisions for written procedures for implementing the ODCM, the PCP, and the QA program. It is also an objective of RETS to properly retain the documented records in relation to the environmental monitoring program and certain QA procedures. The Licensee has made a commitment to establish, implement, and maintain written procedures for the PCP, the ODCM, and the QA program. The Licensee intends to retain the records of the radiological environmental monitoring program, as well as the records of quality assurance activities, for the duration of the facility operating license. It is thus determined that the Licensee has met the intent of NUREG-0472.

3.2.11 Reports

In addition to the reporting requirements of Title 10, Code of Federal Regulations (10CFR), the objective of the RETS with regard to administrative controls is to ensure that appropriate periodic and special reports are submitted to the NRC.

The Licensee has made a commitment to follow applicable reporting requirements stipulated by 10CFR regulations and also the following reports specified by NUREG-0472:

1. Annual Radiological Environmental Operating Report. In Section 6.6.1.b of the Licensee's submittal, a commitment is made to provide an annual radiological environmental operating report that includes summaries, interpretations, and statistical evaluation of the results of the environmental surveillance program. The report also includes the results of land use censuses, and participation in an interlaboratory comparison program specified by Specification 3.12.3 of NUREG-0472.

2. Semiannual Radioactive and Effluent Release Report. In Section 6.6.1.c of the Licensee's submittal, a commitment is made to provide semiannual radioactive effluent and solid waste release reports which include a summary of radioactive liquid and gaseous effluents and solid waste released, an assessment of offsite doses, and a list of unplanned releases. Listing of new location for dose calculations identified by the land use census as well as any changes to ODCM, PCP, and major changes to radioactive waste treatment systems are also included in the report.

3. Special Report. The Licensee has made a commitment to file a 30-day special report to the NRC under the following conditions as prescribed by the proposed specifications:

   - exceeding liquid effluent dose and concentration limits according to the proposed Specifications 3.11.A-2 and 3.11.A-1, respectively

- exceeding gaseous effluent dose and dose rate limits according to the proposed Specifications 3.11.B-2, 3.11.B-1, and 3.11.B-3, respectively
- exceeding the projected monthly dose limits without treating the radioactive liquid or gaseous waste according to the proposed Specifications 3.11.A-3 and 3.11.B-4, respectively
- exceeding total dose limits according to the proposed Specification 3.11.C
- exceeding the reporting levels of proposed Table 4.9-4 for the radioactivity measured in the environmental sampling medium.

These reporting commitments have satisfied the provisions of NUREG-0472.

3.2.12 Implementation of Major Programs

One objective of the administrative controls is to ensure that implementation of major programs, such as the PCP, ODCM, and major changes to the radioactive waste treatment system, follows appropriate administrative procedures. The Licensee has made a commitment to review, report, and implement major programs such as the PCP, ODCM, and major changes to the radioactive waste treatment system. This commitment meets the intent of NUREG-0472.

3.2.13 Design Features

The objective of the RETS with regard to design features is to provide a map of the site area defining the site boundary and unrestricted areas within the site boundary, as well as defining points of release for liquid and gaseous effluents and points where liquid effluents leave the site. The Licensee has provided an acceptable Figure 5.1-1, except that a scale should be provided for the figure.

3.3 OFFSITE DOSE CALCULATION MANUAL (ODCM)

As specified in NUREG-0472, the ODCM is to be developed by the Licensee to document the methodology and approaches used to calculate offsite doses and maintain the operability of the effluent system. As a minimum, the ODCM should provide equations and methodology for the following topics:

- alarm and trip setpoint on effluent instrumentation
- liquid effluent concentration in unrestricted areas
- gaseous effluent dose rate at or beyond the site boundary
- liquid and gaseous effluent dose contributions
- liquid and gaseous effluent dose projections.

In addition, the ODCM should contain flow diagrams, consistent with the systems being used at the station, defining the treatment paths and the components of the radioactive liquid, gaseous, and solid waste management systems. A description and the location of samples in support of the environmental monitoring program are also needed in the ODCM.

3.3.1 Evaluation

The Licensee has followed the methodology of NUREG-0133 [9] to determine the alarm and trip setpoints for the liquid and gaseous effluent monitors. To ensure that the MPC, as specified in 10CFR20, will not be exceeded even in the case of simultaneous discharge, the Licensee will adjust the setpoints according to the apportionment of the radioactivity released from each respective effluent line.

The Licensee has demonstrated the method of calculating the radioactive liquid concentration by describing in the ODCM the means of collecting and analyzing representative samples prior to and after releasing liquid effluents into the circulating water discharge. The method provides added assurance of compliance with 10CFR20 for liquid releases.

Methods are also included for showing that dose rates at or beyond the site boundary due to noble gases, iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days are in compliance with 10CFR20. In this calculation, the Licensee has considered effluent releases from the process vent, the ventilation vent, and condenser air ejectors;

releases from the process vent are treated as mixed mode, and releases from the ventilation vent and air ejectors are treated as ground level. In all cases, the Licensee has used the highest annual average values of relative concentration (X/Q) and relative deposition (D/Q) to determine the controlling locations. The Licensee intends to use the maximally exposed individual and the critical organ as the reference receptor. For noble gases, the Licensee has considered the total body dose and the skin dose resulting from gamma and beta radiation, respectively. For iodine-131, tritium, and particulates, the Licensee has considered the inhalation pathway for estimating the doses. The Licensee has demonstrated that the described methods and relevant parameters have followed the conservative approaches provided by NUREG-0133 and Regulatory Guide 1.109.

Evaluation of the cumulative dose is to ensure that the quarterly and annual dose design objectives specified in RETS are not exceeded.

For liquid releases, the Licensee has identified fish and invertebrate consumption as the two viable pathways. In the calculation, the Licensee has used a near-field dilution factor specific to the plant; all other key parameters follow the suggested values given in Regulatory Guide 1.109. The Licensee has used the maximally exposed adult individual as the reference receptor. To correctly assess the cumulative dose, the Licensee intends to estimate the dose once per 31 days.

Evaluation of the cumulative dose from noble gas releases includes both beta and gamma and air doses at and beyond the site boundary. The critical organs under consideration are the total body and skin for gamma and beta radiation, respectively. Again, the Licensee has used the maximum (X/Q) values as discussed earlier and has followed the methodology and parameters of NUREG-0133 and Regulatory Guide 1.109.

For iodine-131, tritium, and particulates with half-lives greater than 8 days, the Licensee has provided a method to demonstrate that cumulative doses calculated from the release meet both quarterly and annual design objectives. The Licensee has demonstrated a method of calculating the dose using maximum annual average (X/Q) values for the inhalation pathway and has included (D/Q) values for the grass-cow-milk pathway for ingestion, for which the Licensee considered the infant to be the critical age group and thyroid to be the critical organ. This approach is consistent with the methodology of NUREG-0133.

Using the existing methodology for gaseous and liquid dose calculations, the Licensee has demonstrated a procedure to project the monthly dose and to ensure that the design objectives for the liquid radwaste system and the gaseous radwaste system are not exceeded.

Adequate flow diagrams defining the effluent paths and components of the radioactive liquid and gaseous waste treatment systems have been provided by the Licensee. Radiation monitors specified in the Licensee-submitted RETS are also properly identified in the flow diagrams.

The Licensee has provided a detailed description of many sampling locations in the ODCM, but has not provided a table and figure(s) containing specific parameters of distance and direction sector from the centerline of the reactor for each and every sample location in RETS Table 4.9-3 on environmental monitoring. Futhermore, the Licensee's May 4, 1983 ODCM submittal is incomplete, leaving out oysters, crabs, and fish as well as Figures 13.0, 13.1, and 13.2 provided in the earlier draft submittal.

In summary, the Licensee's ODCM uses documented and approved methods that are consistent with the methodology and guidance in NUREG-0133, and, therefore, is an acceptable reference, except for an incomplete listing and description of samples in the environmental monitoring program.

4. CONCLUSIONS

The Licensee submitted the same Radiological Effluent Technical Specifications (RETS) and Offsite Dose Calculation Manual (ODCM) for both Units 1 and 2 of Surry Nuclear Power Station. Table 1 summarizes the results of the final review and evaluation of the RETS submittal. Comments apply equally to Units 1 and 2.

The following conclusions were reached:

1. The Licensee's proposed RETS, submitted May 4, 1983 [22], meets the intent of the NRC staff's "Standard Radiological Effluent Technical Specifications," NUREG-0472, for Surry Power Station Units 1 and 2.

2. The Licensee's ODCM, submitted May 4, 1983 [23], uses documented and approved methods that are applicable to Surry Power Station Units 1 and 2 and are consistent with the criteria of NUREG-0133. It is an acceptable reference except for an incomplete listing and description of samples in the environmental monitoring program.

Table 1. Evaluation of Proposed Radiological Effluent Technical Specifications (RETS), Surry Power Station Units 1 and 2

	Technical Specifications			
	NRC Staff Std. RETS NUREG-0472 (Section)*	Licensee Proposal (Section)	Replaces or Updates Existing Tech. Specs. (Section)	Evaluation
Effluent Instrumentation	3/4.3.3.3.10 3/4.3.3.3.11	3.7.E	3.7	Meets the intent of NRC criteria
Radioactive Effluents	3/4.11.1.1 3/4.11.2.1	3.11.A-1 3.11.B-1	3.11	Meets the intent of NRC criteria
Offsite Doses	3/4.11.1.2, 3/4.11.2.2, 3/4.11.2.3, 3/4.11.4	3.11.A-2, 3.11.B-2, 3.11.B-3, 3.11.C	To be added	Meets the intent of NRC criteria
Effluent Treatment	3/4.11.1.3 3/4.11.2.4	3.11.A-3 3.11.B-4	To be added	Meets the intent of NRC criteria
Tank Inventory Limits	3/4.11.1.4 3/4.11.2.6	3.11.B-6	To be added	Meets the intent of NRC criteria
Explosive Gas Mixtures	3/4.11.2.5B	3.11.B-5	To be added	Meets the intent of NRC criteria
Solid Radioactive Waste	3/4.11.3	3.11.E	To be added	Meets the intent of NRC criteria
Environmental Monitoring	3/4.12.1	3.11.D	4.9	Meets the intent of NRC criteria
Audits and Reviews	6.5.1, 6.5.2	6.1.C-1 6.1.C-2 6.1.C-3	6.1	Meets the intent of NRC criteria
Procedures and Records	6.8, 6.10	6.4, 6.5	6.4	Meets the intent of NRC criteria
Reports	6.9	6.6	6.6	Meets the intent of NRC criteria
Implementation of Major Programs	6.13, 6.14, 6.15	6.8, 6.9	To be added	Meets the intent of NRC criteria

*Section number sequence is according to NUREG-0472, Rev. 3, Draft 7' [12].

5. REFERENCES

1. "Radiological Effluent Technical Specifications for Pressurized Water Reactors," Rev. 2
   NRC, July 1979
   NUREG-0472

2. Title 10, Code of Federal Regulations, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, 'As Low As Is Reasonably Achievable,' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents"

3. Title 10, Code of Federal Regulations, Part 50, Appendix I, Section V, "Effective Dates"

4. Title 10, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation"

5. Title 40, Code of Federal Regulations, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"

6. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

7. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

8. "Radiological Effluent Technical Specifications for Boiling Water Reactors," Rev. 2
   NRC, July 1979
   NUREG-0473

9. "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, A Guidance Manual for Users of Standard Technical Specifications"
   NRC, October 1978
   NUREG-0133

10. C. Willis and F. Congel (NRC)
    "Summary of Draft Contractor Guidance of RETS"
    Presented at the AIF Environmental Subcommittee Meeting, Washington, DC
    May 19, 1982

11. F. Congel (NRC)
    Memo to RAB Staff (NRC)
    Subject: Interim Changes in the Model Radiological Effluent Technical Specifications (RETS)
    August 9, 1982

12. "Radiological Effluent Technical Specifications for Pressurized Water Reactors," Rev. 3, Draft 7', intended for contractor guidance in reviewing RETS proposals for operating reactors
NRC, September 1982
NUREG-0472

13. Amendments to Operating License for Surry Units 1 and 2
"Proposed Technical Specifications, Change No. 73"
March 15, 1979

14. Surry Offsite Dose Calculation Manual (Draft)
Virginia Electric and Power Company, July 30, 1980
NRC Docket Nos. 50-280, 281

15. "Comparison of Specification NUREG-0472, Radiological Effluent Technical Specifications for PWRs, vs. Licensee Submittal of Radiological Effluent Technical Specifications for Surry Power Station Units 1 and 2" (Draft)
Franklin Research Center, April 26, 1982

16. Technical Review of Offsite Dose Calculation Manual for Surry Power Station Units 1 and 2 (Draft)
Franklin Research Center, April 26, 1982

17. S. Pandey/A. Cassel (FRC)
Letter of Transmittal to W. Meinke (NRC)
Subject: Trip report on site visit to Surry Power Station Units 1 and 2
July 6, 1982

18. Surry Radiological Effluent Technical Specifications (RETS)
Virginia Electric and Power Company, November 12, 1982
NRC Docket Nos. 50-280, 50-281

19. Surry Offsite Dose Calculation Manual
Virginia Electric and Power Company, November 12, 1982
NRC Docket Nos. 50-280, 50-281

20. S. Pandey/S. Chen (FRC)
Letter of Transmittal to W. Meinke (NRC)
Subject: Comments on Surry RETS and ODCM Draft Submittals
January 10, 1983

21. W. Meinke (NRC)
Letter of Transmital to S. Pandey (FRC)
Subject: Summary of Telecon with the Licensee (VEPCO) on RETS and ODCM Submittals
February 1, 1983

22. Surry Radiological Effluent Technical Specifications
Virginia Electric and Power Company
May 4, 1983
NRC Docket Nos. 50-280, 50-281

23. Surry Offsite Dose Calculations Manual
Virginia Electric and Power Company
May 4, 1983
NRC Docket Nos. 50-280, 50-281

24. "Comparison of Specification NUREG-0472, Radiological Effluent Technical Specifications for PWRs, vs. Licensee Submittal of Radiological Effluent Technical Specifications for Surry Power Station Units 1 and 2"
Franklin Research Center, June 15, 1983

25. W. Meinke (NRC)
Letter of Transmittal to S. Pandey (FRC)
Subject: Comments on Plant-Specific Issues Regarding Surry RETS Submittal
June 10, 1983

26. C. Willis (NRC)
Letter to Dr. S. Pandey (FRC)
Subject: Changes to RETS requirements following meeting with Atomic Industrial Forum (AIF)
November 20, 1981

27. C. Willis (NRC)
Letter to Dr. S. Pandey (FRC)
Subject: Control of explosive gas mixture in PWRs
December 18, 1981

28. C. Willis and F. Congel (NRC)
"Status of NRC Radiological Effluent Technical Specification Activities"
Presented at the AIF Conference on NEPA and Nuclear Regulations,
Washington, D.C.
October 4-7, 1981

29. C. Willis (NRC)
Memo to P. C. Wagner (NRC)
"Plan for Implementation of RETS for Operating Reactors"
November 4, 1981

30. W. P. Gammill (NRC)
Memo to P. C. Wagner (NRC)
"Current Position on Radiological Effluent Technical Specifications (RETS) Including Explosive Gas Controls"
October 7, 1981

31. "An Acceptable Radiological Environmental Monitoring Program"
Radiological Assessment Branch Technical Position, Revision 1
November 1979

32. W. P. Gammill/F. J. Congel (NRC)
Memo to ETSB/RAB (NRC)
"Radiological Effluent Technical Specifications (RETS) Provisions for I-133"
November 29, 1982

33. Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40CFR190)
NRC, February 1980
NUREG-0543

34. "General Contents of the Offsite Dose Calculation Manual," Revision 1
Branch Technical Position, Radiological Assessment Branch
February 8, 1979

35. Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I
NRC, October 1977
Regulatory Guide 1.109, Revision 1

LICENSE AUTHORITY FILE CC

UNITED STATES
NUCLEAR REGULATORY COMMISSION **DO NOT REMOVE**
WASHINGTON, D. C. 20555

April 30, 1984

Docket Nos. 50-280
and 50-281

Posted
Correction to
Amdt. 94
for DPR-37

Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

By letter dated April 10, 1984, you identified some oversight errors in your proposed Technical Specification changes dated September 13, 1983, as supplemented. These errors were a failure to change certain referenced paragraph numbers which were revised. We issued Amendment Nos. 95 and 94 on February 24, 1984 for Surry Units 1 and 2 with these errors. Please find enclosed Technical Specification pages which make the necessary corrections.

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

Mr. W. L. Stewart
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