Attachment 1

Proposed Technical Specification Changes

Surry Units 1 and 2

Virginia Electric and Power Company

TECHNICAL SPECIFICATIONS
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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

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specifications 6.6.B.2 and 6.6.B.3.

N. Dose Equivalent I-131

The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

O. Gaseous Radwaste Treatment System

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

P. Process Control Program (PCP)

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

Q. Purge - Purging

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

R. 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.

S. 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.

T. Site Boundary

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

U. 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.

V. 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.

C. In the event of subsystem instrumentation channel failure permitted by Specification 3.7.B2, Tables 3.7-2 and 3.7-3 need not be observed during the short period of time an operable subsystem channel is 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 explosive gas monitoring instrumentation channels shown in Table 3.7-5(a) shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.A.1 are not exceeded.

1. With an explosive gas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, declare the channel inoperable and take the action shown in Table 3.7.5(a).

2. With less than the minimum number of explosive gas monitoring instrumentation channels operable, take the action shown in Table 3.7-5(a). Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission (Region II) to explain why this inoperability was not corrected in a timely manner.

4. The steam line high differential pressure limit is set well below the differential pressure expected in the event of a large steam line break accident as shown in the safety analysis. (3)

5. The high steam line flow differential pressure setpoint is constant at 40% full flow between no load and 20% load and increasing linearly to 110% of full flow at full load in order to protect against large steam line break accidents. The coincident low T_{avg} setting limit for SIS and steam line isolation initiation is set below its hot shutdown value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break. (3)

Accident Monitoring Instrumentation

The operability of the accident monitoring instrumentation in Table 3.7-6 ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. On the pressurizer PORV's, the pertinent channels consist of limit switch indication and acoustic 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." Potential accident effluent release paths are equipped with radiation monitors to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. The effluent release paths monitored are the Process Vent Stack, Ventilation Vent Stack, Main Steam Safety Valve and Atmospheric Dump Valve discharge and the Auxiliary Feedwater Pump Turbine Exhaust. These monitors meet the requirements of NUREG 0737.

Instrumentation is provided 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 or Appendix A to 10 CFR Part 50.

Containment Hydrogen Analyzers

Continuous indication of hydrogen concentration in the containment atmosphere is provided in the control room over the range of 0 to 10 percent hydrogen concentration.

These redundant, qualified hydrogen analyzers are shared by Units 1 and 2 with the capability of measuring containment hydrogen concentration for the range of 0 to 10 percent and the installation of instrumentation to indicate and record this measurement.

A transfer switch with control circuitry is provided for the capability of Unit 1 to utilize both analyzers or for Unit 2 to utilize both analyzers.

Each unit's hydrogen analyzer will receive a transferable power supply from Unit 1 and Unit 2. This will ensure redundancy for each unit.

Indication of Unit 1 and Unit 2 hydrogen concentration is provided on Unit 1 PAMC panel and Unit 2 PAMC panel. Hydrogen concentration is also recorded on qualified recorders. In addition, each hydrogen analyzer is provided with an alarm for trouble/high hydrogen content. These alarms are located in the

References

(1) FSAR - Section 7.5

(2) FSAR - Section 14.5

(3) FSAR - Section 14.3.2

TABLE 3.7-5

AUTOMATIC FUNCTIONS
OPERATED FROM RADIATION MONITORS ALARM

MONITOR CHANNEL	AUTOMATIC FUNCTION AT ALARM CONDITIONS	MONITORING REQUIREMENTS	ALARM SETPOINT µCi/cc
1. Component cooling water radiation monitors	Shuts surge tank vent valve HCV-CC-100	See Specification 3.13	Twice Background
2. Containment particulate and gas monitors (RM-RMS-159 & RM-RMS-160, RM-RMS-259 & RM-RMS-260)	Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specification 3.10	Particulate $\leq 9 \times 10^{-9}$ Gas $\leq 1 \times 10^{-5}$
3. Manipulator crane area monitors (RM-RMS-162 & RM-RMS-262)	Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specification 3.10	$\leq$ 50 mrem/hr

TABLE 3.7-5(a)

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Waste Gas Holdup System Explosive Gas Monitoring System		
(a) Hydrogen Monitor	1	1
(b) Oxygen Monitor	1	1

ACTION 1 - With the number of channels operable less than required by the minimum channels operable requirement, operation of this waste gas hold up system may continue provided grab samples are collected at least once per 24 hours and analyzed within the following 4 hours.

3.11 RADIOACTIVE GAS STORAGE

Applicability

Applies to the storage of radioactive gases.

Objective

To establish conditions by which gaseous waste containing radioactive materials may be stored.

Specification

A. Explosive Gas Mixture

1. The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.
    a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.
    b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the affected tank and reduce the concentration of oxygen to less than or equal to 4% by volume, then take the above action.
2. The requirements of Specification 3.0.1 are not applicable.

B. Gas Storage Tanks

1. 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).
2. 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 limits.
3. The requirements of Specification 3.0.1 are not applicable.

Basis

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.

TABLE 4.1-1A

EXPLOSIVE MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Waste Gas Holdup System Explosive Gas Monitoring System			
(a) Hydrogen Monitor	D	Q(1)	M
(b) Oxygen Monitor	D	Q(2)	M

(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
M - Monthly
Q - Quarterly

4.9 RADIOACTIVE GAS STORAGE MONITORING SYSTEM

Applicability

Applies to the periodic monitoring of radioactive gas storage.

Objective

To ascertain that waste gas is stored in accordance with Specification 3.11.

Specification

A. The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the limits of Specification 3.11.A 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(a) of Specification 3.7.E.

B. The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.11.B at least once per month when the specific activity of the primary reactor coolant is $\leq$ 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.

N. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

O. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

9. Records of the service lives of all hydraulic and mechanical snubbers on safety-related systems, including the data at which the service life commences and associated installation and maintenance records.

10. Records of the annual audit of the Station Emergency Plan and implementing procedures.

11. Records of the annual audit of the Station Security Plan and implementing procedures.

12. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

B. Unique Reporting Requirements

1. Inservice 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 5 years of operation. This report shall include an evaluation of the results of the inservice inspection program and will be reviewed in light of the technology available at that time.

2. Annual Radiological Environmental Operating Report[1].

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

3. Semiannual Radioactive Effluent Release Report[3]

The Semiannual Radioactive Effluent Release Report 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 report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

4. Containment Leak Rate Test

Each containment integrated leak rate test shall be the subject of a summary technical report. Upon completion of the initial containment leak rate test specified by proposed Appendix J to 10 CFR 50, a special report shall, if that Appendix is adopted as an effective rule, be submitted to the Director, Division of Reactor Licensing, USNRC, Washington, D. C. 20555, and other containment leak rate tests specified by Appendix J that fail to meet the acceptance criteria of the appendix, shall be the subject of special summary technical reports pursuant to Section V.B of Appendix J:

a. "Report of Test Results - The initial Type A tests shall be subject of a summary technical report submitted to the Commission approximately 3 months after the conduct of the test. This report shall include a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable to the initial test, and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting the acceptance criteria."

"For periodic tests, leakage rate results of Type A, B, and C tests that meet the acceptance criteria of Sections III.A.7, III.B.3, respectively, shall be reported in the licensee's periodic operating report. Leakage test results of Type A, B, and C tests that fail to meet the acceptance criteria of Sections III.A.7, III.B.3, and III.C.3, respectively, shall be reported in a separate summary report that includes an analysis and interpretation of the test data, the least squares fit analysis of the test data, the instrument error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included."

C. Special Reports

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 Section 20.407 of 10 CFR Part 20.

3. A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6.8 PROCESS CONTROL PROGRAM AND OFFSITE DOSE CALCULATION MANUAL

A. Process Control Program (PCP)

Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.12. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

    b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall require review and acceptance by the SNSOC and the approval of the Station Manager prior to implementation.

B. Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.12. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall require review and acceptance by the SNSOC and the approval of the Station Manager prior to implementation.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

Attachment 2

Discussion of Proposed Changes

Surry Units 1 and 2

Virginia Electric and Power Company

Discussion of Proposed Technical Specification Change

Introduction

The proposed changes to the Surry Units 1 and 2 Technical Specifications removes the Radiological Effluent Technical Specifications. These specifications are being removed to the Offsite Dose Calculation Manual (ODCM) or the Process Control Program (PCP). Technical Specifications relating to these documents are also being amended due to their expanded role.

Background

This proposed change is based on the NRC's Generic Letter 89-01 dated January 31, 1989. The letter stated that the NRC will approve a Technical Specification amendment to delete the Radiological Effluent Technical Specifications if the requirements are relocated to the ODCM or PCP. The letter was very specific about what changes are acceptable and warned that proposed amendments that deviate from its guidance will require a longer, more detailed review. It stated that conforming amendment requests will be expeditiously reviewed. This proposed change follows the guidance in the letter. Some changes were needed due to differences between Surry and Standard Technical Specifications.

One requirement was corrected when it was moved to the ODCM. The correction was necessary due to an error in Specification 3.11.A.3.a, which requires that the liquid radwaste treatment system be used to reduce monthly projected doses due to liquid effluents to 0.06 mrem whole body and 0.2 mrem to the critical organ. The phrase "from each unit" was inadvertently omitted from the Specification. The basis of the Specification is that in order to keep effluents as low as reasonably achievable, the limits were set at a "suitable fraction" (1/4) of the limits in Section II.A of Appendix I, 10 CFR 50. The Appendix I limits are on a per reactor basis. The missing phrase is present in the equivalent North Anna Specification and in Specification 3.11.1.3 of the draft Revision 5 of the Standard Technical Specifications for Westinghouse PWRs. Note that the ODCM is intended to apply to both North Anna and Surry and it could be confusing to have an exception for Surry where it is not logically expected. The ODCM Section 6.2.4.a has therefore been corrected to be consistent with North Anna and Standard Technical Specifications and Appendix I.

Description of the Proposed Change

1. In the index, item 3.11 "Effluent Release" is changed to "Radioactive Gas Storage."

2. In the index, item 4.9, the phrase "Effluent Sampling and Radiation" is changed to "Radioactive Gas Storage."

3. In the index, item 6.9 is deleted.

4. Specification 1.0.M, the ODCM definition, is replaced with item number 1.17 from Enclosure 3 of the Generic Letter, except references to Specifications 6.8.4, 6.9.1.3 and 6.9.1.4 are changed to 6.4, 6.6.B.2 and 6.6.B.3 respectively. The revision reflects the expanded role of the ODCM.

5. Specification 1.0.P, the PCP definition, is replaced with item number 1.22 from Enclosure 3 of the Generic Letter. This adds references to 10CFR61 and burial ground requirements which were previously included in "other requirements."

6. Section 1.0.R is deleted. The requirements are added to the PCP. Definitions S through W are re-lettered R through V.

   Although not reflected in all of the titles, Specifications 3.7, 3.11, 4.1 and 4.9 cover waste gas storage and radioactive effluents. The following changes delete effluent monitoring requirements, which have been added to the ODCM, but retain the gas storage monitoring requirements.

7. The phrase "radioactive liquid and gaseous effluent" in Specification 3.7.E is replaced with "explosive gas."

8. The phrase "and Table 3.7-5(b)" is deleted.

9. The phrase "Specifications 3.11.A.1 and 3.11.B.1" is changed to "Specification 3.11.A.1."

10. The last sentence of 3.7.E, before 3.7.E.1 is deleted.

11. The phrase "a radioactive liquid or gaseous effluent" in 3.7.E.1 is changed to "an explosive gas."

12. The phrase "without delay suspend the release of radioactive liquid or gaseous effluents monitored by the affected channel and" in 3.7.E.1 is deleted.

13. The phrase "or change the setpoint so it is acceptably conservative" in 3.7.E.1 is replaced with "and take the action shown in Table 3.7-5(a)."

14. The phrase "radioactive liquid or gaseous effluent" in Specification 3.7.E.2 is changed to "explosive gas."

15. The phrase "or Table 3.7-5(b)" in 3.7.E.2 is deleted.

16. The phrase "explain in the next Semiannual Radioactive Effluent Release Report" in 3.7.E.2 is replaced with "submit a Special Report to the Commission (Region II) to explain."

17. The paragraph titled "Automatic Function Operated from Radiation Monitors" in the basis section, page 3.7-8 is deleted.

18. On page 3.7-9, the paragraph titled "Radioactive Liquid Effluent Monitoring Instrumentation" is deleted.

19. The first two sentences of the next paragraph are deleted.

20. In the next sentence the phrase "This instrumentation also includes provisions" is changed to "Instrumentation is provided."

21. Reference number four on page 3.7-9c is deleted.

22. In Table 3.7-5, items 1, 3, 4, and 7 are deleted. The remaining items are renumbered. References to Specification 4.9 are deleted. The words "and exhaust" are deleted to reflect the removal of the purge exhaust fans by a previous design change.

23. Table 3.7-5(a) is deleted.

24. Table 3.7-5(b) is changed to 3.7-5(a). In the title, "Radioactive Gaseous Effluent" is changed to "Explosive Gas." Items 1, 3 and 4 and Action items 1, 2 and 3 are deleted. "Action 4" is renumbered "Action 1." The page number is changed to 3.7-20a.

25. The title of section 3.11 is changed to "Radioactive Gas Storage."

26. The "Applicability" section of 3.11 is changed to: "Applies to the storage of radioactive gases."

27 Under "Objective," "and liquid" is deleted, "released" is changed to "stored" and everything after "released" is deleted.

28. All of 3.11.A and sections 3.11.B.1 through 3.11.B.4 are deleted.

29. Our letter, serial number 90-297, dated May 25, 1990 proposed changes to section 3.11.5. The Specification in Attachment 1 includes these changes, which are indicated by a double bar. In addition to the previously proposed changes, the "5" in 3.11.B.5 is changed to "A" and the subsection labels "a" and "b" are changed to "1" and "2." A new subsection 3 is added: "The requirements of Specification 3.0.1 are not applicable." The new subsection is needed

because section 3.11.F is to be deleted. The Specifications are moved to page 3.11-1.

30. The "6" in 3.11.B.6 is changed to "B" and the subsection labels "a" and "b" are changed to "1" and "2." Because section 3.11.F is to be deleted, a new subsection 3 is added: "The requirements of Specification 3.0.1 are not applicable." The Specifications are moved to page 3.11-1.

31. Sections 3.11.C through 3.11.F are deleted.

32. In the 3.11 Bases section, everything except the "Explosive Gas Mixture" and "Gas Storage Tanks" subsections is deleted. The remaining sections are moved to page 3.11-2.

33. Table 4.1-1A is deleted.

34. Table 4.1-1B is changed to Table 4.1-1A. In the title "Radioactive Gaseous Effluent" is changed to "Explosive Gas." Items 1, 3 and 4 and the "Source Check" column and all frequency footnotes except "D," "M" and"Q" are deleted. Item 2 is renumbered and the page number is changed to 4.1-8c

35. In the title of section 4.9, "Effluent Sampling and Radiation" is changed to "Radioactive Gas Storage."

36. Under "Applicability," "and recording" is deleted and "effluents" is changed to "gas storage."

37. The "Objective" section of 4.9 is changed to "To ascertain that waste gas is stored in accordance with Specification 3.11."

38 Sections 4.9.A through 4.9.E and 4.9.H through 4.9.K are deleted. The requirements have been added to the ODCM.

39. The labels for subsections F and G are changed to "A" and "B" and they are moved to page 4.9-1. References to Specifications 3.11.B.5 and 3.11.B.6 and Table 3.7-5(b) are changed to 3.11.A, 3.11.B and 3.7-5(a) respectively.

40. Tables 4.9-1 through 4.9-5 are deleted. The requirements are added to the ODCM.

41. Two new subsections, N and O are added to section 6.4. These are the same as sections 6.8.4.g and 6.8.4.h of Enclosure 3 of Generic Letter 89-01 except in 6.8.4.g, paragraph 10, which does not apply to PWRs, is deleted and paragraph 11 is renumbered 10. The additions are programmatic requirements deleted elsewhere.

42. A new item number 12 is added to section 6.5.B: "Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM."

43. Sections 6.6.B.2 and 6.6.B.3 are replaced with the text of sections 6.9.1.3 and 6.9.1.4 from Enclosure 3 of the Generic Letter. The change simplifies the requirements for the Annual Radiological Environmental Operating Report and the Semi-Annual Radioactive Effluent Release Report. Details have been added to the ODCM. The pages in the remainder of section 6.6 are renumbered.

44. 6.8.A and 6.8.B are replaced with the text of section 6.13 and 6.14 of the Generic Letter's Enclosure 3 except all references to Specification "6.10.3.o" are changed to "6.5.B.12" and "URG" is changed to "SNSOC." Also, subsection labels "a," "b" and "c" are changed to "1," "2" and "3" and labels "1" and "2" are changed to "a" and "b."

45. Section 6.9 is deleted. The requirements are added to the PCP.

Safety Analysis

Although the proposed changes simplify the Technical Specifications, there is no reduction in requirements because of additions to the ODCM and PCP. The following table outlines the disposition of each requirement removed from the Technical Specifications.

Specification	Addition
1.0.R	PCP
3.7.E, 4.1 and 4.9.A (liquid effluents)	ODCM 6.2.2 TS 6.4.N.1
3.7.E, 4.1 and 4.9.A (gaseous effluents)	ODCM 6.3.2 TS 6.4.N.1
3.11.A.1	ODCM 6.2.1 TS 6.4.N.2-3
3.11.A.2	ODCM 6.2.3 TS 6.4.N.4-5
3.11.A.3	ODCM 6.2.4 TS 6.4.N.6
3.11.B.1	ODCM 6.3.1 TS 6.4.N.3 TS 6.4.N.7
3.11.B.2	ODCM 6.3.3 TS 6.4.N.5 TS 6.4.N.8
3 11.B.3	ODCM 6.3.4 TS 6.4.N.5 TS 6.4.N.9
3.11.B.4	ODCM 6.3.5 TS 6.4.N.6
3.11.C	ODCM 6.4 TS 6.4.N.10
3.11.D.1	ODCM 6.5.1 TS 6.4.O.1
3.11.D.2	ODCM 6.5.2 TS 6.4.O.2
3.11.D.3	ODCM 6.5.3 TS 6.4.O.3
3.11.E	PCP
4.9.B	ODCM 6.2.5
4.9.C	ODCM 6.2.3 ODCM 6.3.3
4.9.D	ODCM 6.2.4 ODCM 6.3.5

Specification	Addition
4.9.E	ODCM 6.3.1
	ODCM 6.3.3
	ODCM 6.3.4
4.9.H	ODCM 6.5.1
4.9.I	ODCM 6.5.2
4.9.J	ODCM 6.5.3
4.9.K	PCP
6.6.B.2	ODCM 6.6.1
6.6.B.3	ODCM 6.6.2
6.9	PCP

Attachment 3

10 CFR 50.92 Evaluation

Surry Units 1 and 2

Virginia Electric and Power Company

Basis for No Significant Hazards Determination

The proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92 because operation of Surry Units 1 and 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequence of an accident previously evaluated. This change does not alter the conditions or assumptions of any accident analysis.

(2) create the possibility of a new or different kind of accident from any accident previously identified. This change does not alter the conditions or assumptions of any accident analysis. This is not an actual hardware change.

(3) involve a significant reduction in a margin of safety. This change does not alter the conditions or assumptions of any accident analysis. It is not an actual hardware change.

Therefore, pursuant to 10 CFR 50.92, based on the above considerations, it has been determined that this change does not involve a significant hazards consideration.

JAN 1 [illegible]

SAFETY EVALUATION NO. ______________

STATION/UNIT(S): Surry 1+2

SAFETY EVALUATION FORM

PART A - RESOLUTION SUMMARY REPORT

(1) List the governing document(s) for which the safety evaluation is being performed: Technical Specifications

(2) Briefly describe the change, test, or experiment being evaluated: Radiological Effluent Tech Specs (RETS) are being moved to the Offsite Dose Calculation Manual and the Process Control Program.

(3) Briefly describe the purpose for this change, test, or experiment: To simplify Technical Specifications as authorized by Generic Letter 89-01.

Based on the information contained herein, the following is required and is attached (check as appropriate):

✓ a 10 CFR 50.59 safety evaluation (PART D, QUESTIONS 1-4)

____ a 10 CFR 72.48 safety evaluation (SPS/ISFSI only - PART D, QUESTIONS 1-6)

Briefly state the major issues considered, the reason for the change, test, or experiment should be allowed, and why an unreviewed safety question does or does not exist (a simple statement of conclusion alone is insufficient; attach additional sheets if needed):
The reason is convenience. It does not involve an unreviewed safety question because the requirements are only being moved.

JAN 1 3 1990

SAFETY EVALUATION NO. ____________________

STATION/UNIT(S): Surry 1+2

PART A - RESOLUTION SUMMARY REPORT (Continued)

Recommended approval - Cognizant Supervisor: ____________________________

____Approved ____Disapproved ____Approved as modified ____Requires further evaluation

SNSOC Chairman ________________________________ Date ____________________

Comments: ____

____

JAN 1 [illegible]

PART B - APPLICABLE REFERENCES

(1) Identify applicable UFSAR section(s): 11

(2) Identify applicable Technical Specification section(s): 1.0, 3.7, 3.11, 4.1, 4.9, 6.4, 6.5, 6.6, 6.8, 6.9

(3) Identify any other references used in this review: Offsite Dose Calculation Manual
Radioactive Waste Process Control Program.

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION

NOTE: Items denoted by a double asterisk (**) which are answered with a "YES," require Engineering approval.

_____ Yes ** ✓ No 1. Will the operation of any safety related system or component as described in the SAR and/or the Technical Specifications be altered? This includes abandonment of equipment or extended periods of equipment out of service.
Explain: The requirements are only being moved. Operations will be unchanged.

_____ Yes ** ✓ No 2. Will the activity alter the performance characteristics of any safety related system or component? (Note: Action Statements, jumpers, and temporary modifications should be reviewed.)
Explain: The change relocates the RETS with only format changes.

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION (continued) JAN 1 5 1990

_____ Yes __✓__ No 3. Will the ability of operators to control or monitor the plant be reduced in any way?
Explain: The change relocates RETS with only format changes.

_____ Yes __✓__ No 4. If a jumper is involved, are testing requirements as stated on the jumper adequate to ensure operability after installation as well as after removal?
Explain: No jumper is involved.

_____ Yes ** __✓__ No 5. Could the proposed activity affect reactivity? If "Yes," explain (the Reactor Engineer/designee must approve the explanation by initialing):

(Rx. Eng.____________)

_____ Yes __✓__ No 6. Will the activity significantly increase the potential for personnel injury or equipment damage?
Explain: The change only relocates RETS with format changes.

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION (Continued) JAN 1 . 1993

____Yes ✓ No 7. Will the activity create or increase the levels of radiation or airborne radioactivity? If so, will the change result in a significant unreviewed environmental impact, a significant increase in occupational exposure, or significant change to dose to operators performing tasks outside the filtered air boundary during a DBA (GDC-19). If "Yes," explain (the Superintendent of Health Physics/designee must approve the explanation by initialing):

The Specifications to be relocated include ~~##~~ some dealing with rad-effluents but the basic requirements are unchanged.
(Supt. of H.P. ____)

____Yes ✓ No 8. Will the activity change or decrease the effectiveness of the emergency plan? If "Yes," explain (the Emergency Preparedness Coordinator/designee must approve the explanation by initialing):

The requirements involved are for non-emergency operation. They are being moved with only format changes
(E.P. Coordinator ____)

____Yes ✓ No 9. Will the consequences of failure for this activity affect the ability of systems or components to perform safety functions? Briefly describe the modes and consequences of failure considered during this evaluation:

The change only relocated RETS without substantive change.

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION (continued)

JAN 1 [illegible]

____ Yes ** ✓ No 10. Will the activity cause equipment to be exposed (or potentially exposed) to adverse conditions including those created be temperature, pressure, humidity, or radiation? If adverse conditions are possible, could these conditions lead to equipment failure, or a dangerous atmosphere?
Explain: No hardware or operational changes are involved.

____ Yes ✓ No 11. Could the failure of the activity feedback into protective circuitry?
Explain: The change only relocates RETS without substantive change.

____ Yes ** ✓ No 12. Could the activity cause a loss of separation of instrument channels/trains or electrical power supplies?
Explain: The change only relocates RETS without substantive change.

____ Yes ✓ No 13. Will the activity involve the addition or deletion of any electrical loads on the vital bus?
Explain: The change only relocates RETS without substantive change.

____ Yes ✓ No 14. Will the activity adversely affect the ability of a system or component to maintain its integrity or code requirements? Will the activity add or adversely affect components in the ASME XI/ISI program?
Explain:
The change only relocates RETS without substantive change.

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION (continued)

JAN 1 [illegible]

____ Yes ✓ No 15. Will the activity reconfigure, eliminate, or add components and/or piping to the single or two-phase erosion/corrosion piping inspection program?
Explain: The activity only relocates RETS without substantive change

____ Yes ✓ No 16. Will additional surveillance requirements, as defined in the Technical Specifications, be necessitated by the activity?
Explain: Some surveillance requirements are relocated, but none are added.

____ Yes ✓ No 17. Will the applicable Technical Specification basis description be altered by the activity?
Explain: Although the basis description is relocated when the corresponding specification is moved, the actual basis is unchanged.

____ Yes ✓ No 18. Will the activity result in a violation of any Limiting Conditions for Operation (LCO's), as defined in the Technical Specifications?
Explain: Some LCOs will be relocated, but compliance with them will be unchanged.

____ Yes ✓ No 19. Were any other concerns or items identified during this review? If "Yes," explain: ____

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION (continued)

JAN 1

NOTE: THESE ITEMS ARE INCLUDED FOR CONSIDERATION OF POTENTIAL IMPACT. IF THE ANSWER TO ANY OF THE FOLLOWING QUESTIONS IS "YES", A DETAILED REVIEW MUST BE PERFORMED, AND THE RESULTS OF THIS REVIEW MUST BE DOCUMENTED ON A SEPARATE SHEET WHICH REFERENCES THE SAFETY EVALUATION NUMBER AND THE RESPECTIVE PART C ITEM NUMBER. ATTACHMENT 2 PROVIDES GUIDELINES FOR THE DETAILED ENGINEERING REVIEW OF SOME OF THESE ITEMS.

20. STATION SECURITY

____ Yes __✓__ No — Will the activity deactivate a security-related system or breach a security barrier?

21. FIRE PROTECTION/APPENDIX R:

____ Yes __✓__ No — a. Will the activity add or eliminate any combustible material from plant areas?

____ Yes ** __✓__ No — b. Will the activity change or affect and plant structure that acts as a fire barrier?

____ Yes __✓__ No — c. Will the activity impact the performance of an existing fire protection or detection system?

22. EQUIPMENT QUALIFICATION/CLASSIFICATION

____ Yes ** __✓__ No — a. Will the activity adversely affect any Class 1E electrical equipment located in a potentially harsh environment (as designated by the Environmental Zone Descriptions/EZDs)?

____ Yes ** __✓__ No — b. Will the activity have the potential to alter any of the environmental parameters identified in the EZDs?

____ Yes ** __✓__ No — c. Will the activity have the potential to affect any of the electrical distribution systems (i.e., 4KV, 480V, 120VAC, etc.)?

____ Yes __✓__ No — d. Will the activity change or affect equipment on the EQML or Q-List.

____ Yes __✓__ No — e. Will the activity add, eliminate, or have the potential to affect ASME Section XI equipment?

____ Yes __✓__ No — f. Will the activity change a setpoint in the PLS Document?

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION (continued)

JAN 1

23. SEISMIC

_____Yes __✓__No Could the activity be adversely affected by a seismic event or could the activity affect surrounding equipment during a seismic event?

24. HUMAN FACTORS

_____Yes __✓__No a. Will the activity change instrumentation or controls in the control room or on the auxiliary shutdown panel?

_____Yes __✓__No b. Will the activity alter the control room or the auxiliary shutdown panels?

25. SAFETY PARAMETER DISPLAY SYSTEM/ERF

_____Yes __✓__No a. Will the activity change any of the equipment
** associated with the SPDS/ERF, including SPDS/ERF computer inputs?

26. STATION COMPUTERS

_____Yes __✓__No a. Will the activity have a significant
** potential to modify or add software to station computers?

27. ENVIRONMENTAL IMPACT/FLOODING

_____Yes __✓__No a. Will the activity impact more than one-fourth of an acre of land, work in navigable waters, wells, dams, or wetlands, and/or involve any wastes or discharges?

_____Yes __✓__No b. Will the activity involve changes to site terrain, features, or structures?

_____Yes __✓__No c. Will the activity have a significant
** potential to expose safety related equipment to flooding via fluid system equipment/piping malfunction or failure?

_____Yes __✓__No 28. REG. GUIDE 1.97
** Will the activity have a significant potential to modify equipment and/or instrumentation associated with Reg. Guide 1.97 variables?

PART C - ITEMS CONSIDERED BY THIS SAFETY EVALUATION (continued)

JAN 1

29. HEATING-VENTILATION-AIR-CONDITIONING

___ Yes ** _✓_ No — a. Will the activity have a significant potential to increase the heating or cooling loads in plant areas and/or to plant equipment?

___ Yes ** _✓_ No — b. Will the activity change the existing ventilation system in any way?

___ Yes ** _✓_ No — c. Will the activity change any building structures, including walls, ceilings, windows, doors, or floors, such that existing HVAC systems may be affected?

___ Yes _✓_ No — 30. HEAVY LOADS
Will the activity involve heavy loads (including the transfer of heavy loads in areas housing safety related equipment)?

___ Yes _✓_ No — 31. Will there be an introduction of detrimental materials into the containment or other plant areas? (For example, Zinc, and Aluminum alloys are not allowed in the containment because of the potential generation of H2 gas from chemical reactions with these materials. If "Yes," explain: ______________________________

PART D - 10 CFR 50.59 SAFETY EVALUATION

Note: This section is based on the results of the items considered in PART C, and therefore must be completed subsequent to PART C.

UNREVIEWED SAFETY QUESTION DETERMINATION:

1. Which accidents previously evaluated in the SAR were considered?
Radioactive releases in general.
Waste gas decay tank rupture.

_____ Yes ✓ No a. Could the activity increase the probability of occurrence for the accidents identified above? State the basis for your conclusion:
The activity does not change the requirements, it only moves them. WGDT inventory and H_2 and O_2 Specifications will not be moved.

_____ Yes ✓ No b. Could the activity increase the consequences of the accidents identified above? State the basis for your conclusion:
The activity does not change the requirements, it only moves them. WGDT inventory, H_2 and O_2 Specifications will not be moved.

_____ Yes ✓ No c. Could the activity create the possibility for an accident of a different type than was previously evaluated in the SAR? State the basis for your conclusion: The change only involves effluent monitoring and only moves the requirements.

PART D - 10 CFR 50.59 SAFETY EVALUATION (continued)

JAN [illegible]

2. What malfunctions of equipment related to safety previously evaluated in the SAR were considered?

   WGDT rupture.

____ Yes ✓ No a. Could the activity increase the probability of occurrence for the malfunctions identified above? State the basis for your conclusion:
Inventory, H_2 and O_2 limits will be retained in Tech Specs.

____ Yes ✓ No b. Could the activity increase the consequences of the malfunctions identified above? State the basis for your conclusion:
Inventory limits will be retained in Tech Specs.

____ Yes ✓ No c. Could the activity create the possibility for a malfunction of equipment of a different type than was previously evaluated in the SAR? State the basis for your conclusion:
The activity will not change the facility or how it is operated.

____ Yes ✓ No 3. Has the margin of safety of any part of the Technical Specifications as described in the BASES section been reduced?
Explain:
The Specifications to be moved are not safety related and are to be moved with only format changes

✓ Yes ____ No 4. Does the proposed change, test, or experiment require a change to the Technical Specifications?
Explain:
The change relocates Specifications to procedures.

PART D - 10 CFR 50.59 SAFETY EVALUATION (Continued)

JAN 1

____ Yes ✓ No 5. Does the proposed change, test, or experiment involve a significant unreviewed environmental impact? (10CFR72.48 ONLY)
Explain: The RETS are being moved without substantive change.

____ Yes ✓ No 6. Does the proposed change, test, or experiment involve a significant increase in occupational exposure? (10CFR72.48 ONLY) State the basis for your conclusions:
The RETS are being moved without substantive change.

NOTE: IF THE RESPONSE TO QUESTIONS 1-4 (ABOVE) IS "NO," THE PROPOSED ACTIVITY MAY BE IMPLEMENTED, FOLLOWING SNSOC APPROVAL, PROVIDING THAT COMPLETE DOCUMENTATION IS MAINTAINED. IF THE RESPONSE TO ANY PART OF QUESTIONS 1-4 IS "YES," AN APPLICATION FOR AMENDMENT TO THE OPERATING LICENSE MUST BE SUBMITTED AND APPROVED BY THE NRC PRIOR TO IMPLEMENTATION OF THE CHANGE, TEST, OR EXPERIMENT. IN ADDITION, FOR THE SURRY ISFSI, IF THE RESPONSE TO QUESTION 5 OR 6 IS "YES", AN APPLICATION FOR AMENDMENT TO THE ISFSI LICENSE MUST ALSO BE SUBMITTED AND APPROVED PRIOR TO IMPLEMENTING THE CHANGE, TEST, OR EXPERIMENT.

BASED ON THE PRECEDING, THE PROPOSED ACTIVITY (✓) WILL ~~OR ( ) WILL NOT RESULT IN AN UNREVIEWED SAFETY QUESTION AND/OR~~ REQUIRE A LICENSING AMENDMENT.

Prepared by: Robert M Neil Title Staff Engineer

Signature: Robert M Neil Date: 3/19/90

Reviewed by: ____________ Date: ____________

____________ Date: ____________

Design Authority
Reviewed by: ____________ Title ____________

Signature: ____________ Date: ____________

(Documenting concurrence of ** items in Part C answered "YES")
(May be N/A)

Attachment 4

Offsite Dose Calculation Manual

Virginia Electric and Power Company

Station Administrative Procedure

VIRGINIA POWER

Title: Offsite Dose Calculation Manual

Lead Department: Radiological Protection

Procedure Number:	Revision Number:	Effective Date:
VPAP-2103	0	05/31/90

Surry Power Station		North Anna Power Station	
Approved by: SNSOC Chairman	3.23.90 Date	Approved by: SNSOC Chairman	3/22/90 Date
Approved by: Station Manager	4/3/90 Date	Approved by: Station Manager	4/19/90 Date

Approved by: Vice President - Nuclear Operations 4-27-90 Date

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TABLE OF CONTENTS

ATTACHMENTS

1.0 PURPOSE

The Offsite Dose Calculation Manual (ODCM) establishes the requirements of the Radioactive Effluent and Radiological Environmental Monitoring Programs. Methodology and parameters are provided for calculation of offsite doses resulting from radioactive gaseous and liquid effluents, for gaseous and liquid effluent monitoring alarm/trip setpoints, and for conduct of the Environmental Monitoring Program. Requirements are given for the completion of the Annual Radiological Environmental Operating Report and the Semi-Annual Radioactive Effluent Release Report required by Station Technical Specifications. Calculation of offsite doses due to radioactive liquid and gaseous effluents are performed to assure that:

- Concentration of radioactive liquid effluents to the UNRESTRICTED AREA will be limited to the concentration levels of 10 CFR 20, Appendix B, Table II, column 2 for radionuclides other than dissolved or entrained noble gases;
- Exposure to the maximum exposed MEMBER OF THE PUBLIC in the UNRESTRICTED AREA from radioactive liquid effluents will not result in doses greater than the liquid dose limits of 10 CFR 50, Appendix I;
- Dose rate at and beyond the SITE BOUNDARY from radioactive gaseous effluents will be limited to the annual dose rate limits of 10 CFR 20;
- Exposure to the maximum exposed MEMBER OF THE PUBLIC in the UNRESTRICTED AREA from radioactive gaseous effluents will not result in doses greater than the gaseous dose limits of 10 CFR 50, Appendix I; and
- Exposure to the maximum exposed MEMBER OF THE PUBLIC will not exceed 40 CFR 190 dose limits

2.0 SCOPE

This procedure is applicable to the Radioactive Effluent and Environmental Monitoring Programs performed at Surry and North Anna Stations.

3.0 REFERENCES/COMMITMENT DOCUMENTS

3.1 References

3.1.1 10 CFR 20, Standards for Protection Against Radiation

3.1.2 10 CFR 50, Domestic Licensing of Production and Utilization Facilities

3.1.3 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operations

3.1.4 TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites

3.1.5 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, Rev. 1, U.S. NRC, June 1974

3.1.6 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 50, Appendix I, Rev. 1, U.S. NRC, October 1977

3.1.7 Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light - Water - Cooled Reactors, Rev. 1, U.S. NRC, July 1977

3.1.8 Surry and North Anna Technical Specifications (Units 1 and 2)

3.1.9 NUREG-0324, XOQDOQ, Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations, U.S. NRC, September 1977

3.1.10 NUREG/CR-1276, Users Manual for the LADTAP II Program, U.S. NRC, May, 1980

3.1.11 NUREG-0597, User's Guide to GASPAR Code, U.S. NRC, June, 1980

3.1.12 Radiological Assessment Branch Technical Position on Environmental Monitoring, November, 1979, Rev. 1

3.1.13 NUREG-0133, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Stations", October, 1978

3.1.14 NUREG-0543, February 1980, Methods for Demonstrating LWR Compliance With the EPA Uranium Fuel Cycle Standard (40 CFR Part 190)

3.1.15 NUREG-0472, Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors, Rev. 3, March 1982

3.1.16 Environmental Measurements Laboratory, DOE HASL 300 Manual

3.1.17 NRC Generic Letter 89-01, Implementation of Programmatic Controls for Radiological Effluent Technical Specifications (RETS) in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program

3.1.18 UFSAR (Surry and North Anna)

3.1.19 Nuclear Reactor Environmental Radiaiton Monitoring Quality Control Manual, IWL-0032-361

3.2 Commitment Documents

None

4.0 DEFINITIONS

NOTE: Terms which are defined in Surry and North Anna Technical Specifications appear as all capitalized letters in the text of this procedure for identification.

4.1 Channel Calibration

CHANNEL CALIBRATION is defined as the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

4.2 Channel Check

CHANNEL CHECK is defined as 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 channels measuring the same parameter.

4.3 Channel Functional Test

A CHANNEL FUNCTIONAL TEST is defined as:

4.3.1 Analog Channels

The injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

4.3.2 Bistable Channels

The injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

4.4 Dose Equivalent I-131

DOSE EQUIVALENT I-131 is defined as 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. Surry's definition of DOSE EQUIVALENT I-131 allows use of thyroid dose conversion factors from NRC Regulatory Guide 1.109, Revision 1.

4.5 Frequency Notations

NOTE: Frequencies are allowed a maximum extension of 25%.

Frequency notations are defined as follows:

NOTATION	FREQUENCY
D - Daily	At least once per 24 hours
W - Weekly	At least once per 7 days
M - Monthly	At least once per 31 days release
Q - Quarterly	At least once per 92 days
SA - Semi-annually	At least once per 184 days
R - Refueling	At least once per 18 months
S/U - Startup	Prior to each reactor startup
P - Prior to release	Completed prior to each release
N.A. - Not applicable	Not applicable

4.6 Gaseous Radwaste Treatment System

A GASEOUS RADWASTE TREATMENT SYSTEM is the 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. North Anna's Technical Specifications define system composition as the waste gas decay tanks, regenerative heat exchanger, waste gas charcoal filters, process vent blowers, waste gas surge tanks and waste gas diaphragm compressor.

4.7 General Nomenclature

χ = Chi: concentration at a point at a given instant (curies per cubic meter)

D = Deposition: quantity of deposited radioactive material per unit area (curies per square meter)

Q = Source strength (instantaneous; grams, curies, etc.)
= Emission rate (continuous; grams per second, curies per second, etc.)
= Emission rate (continuous line source; grams per second per meter, etc.)

4.8 Member of the Public

MEMBER OF THE PUBLIC shall include individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee 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.

4.9 Operable - Operability

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified functions, and when all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its functions are also capable or performing their related support functions.

4.10 Purge - Purging

PURGE or PURGING is defined as 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.

4.11 Rated Thermal Power

RATED THERMAL POWER shall be a total reactor core heat transfer rate to reactor coolant of:

- Surry: 2441 Megawatt Thermal (MWt)
- North Anna: 2893 MWt

4.12 Site Boundary

The SITE BOUNDARY is defined as that line beyond which the land is not owned, leased, or otherwise controlled by Virginia Power.

4.13 Source Check

A SOURCE CHECK is defined as the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

4.14 Special Report

A report submitted to the NRC in accordance with Technical Specification requirements: (Surry Technical Specification 6.2) (North Anna Technical Specification 6.9.2)

4.15 Thermal Power

THERMAL POWER is defined as the total reactor core heat transfer rate to the reactor coolant.

4.16 Unrestricted Area

UNRESTRICTED AREA is defined as any area at or beyond the SITE BOUNDARY where access is not controlled by Virginia Power for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional or recreational purposes.

4.17 Ventilation Exhaust Treatment System

VENTILATION EXHAUST TREATMENT SYSTEM is defined as the 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 absorbers 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.

5.0 RESPONSIBILITIES

5.1 Health Physics

Health Physics is responsible for:

5.1.1 Establishing and maintaining necessary procedures for sampling and monitoring radioactive effluents and the environment

5.1.2 Performing and documenting surveys, sampling, and analyses of plant effluents and environmental monitoring.

5.1.3 Performing trend analysis on plant effluents and recommending actions to correct adverse trends.

5.1.4 Preparing Effluent and Environmental Monitoring Program records.

5.2 Operations Department

The Operations Department is responsible for requesting samples, analysis, and authorization to release effluents.

6.0 INSTRUCTIONS

NOTE: Meteorological, liquid and gaseous pathway analyses are presented in Attachments 28 and 29, Meteorological, Liquid and Gaseous Pathway Analysis (Surry and North Anna, respectively).

6.1 Sampling and Monitoring Criteria

6.1.1 Surveys, sampling, and analyses shall be performed with instruments calibrated for the type and range of radiation monitored and the nature of the discharge monitored.

6.1.2 Installed monitoring systems shall be calibrated for the type and range of radiation or parameter monitored.

6.1.3 A sufficient number of survey points or samples shall be taken to adequately assess the status of the discharge monitored.

6.1.4 Samples shall be representative of the volume and nature of the monitored discharge.

6.1.5 Surveys, sampling, analyses, and monitoring records shall be accurately and legibly documented and sufficiently detailed so that the meaning and intent is clear.

6.1.6 Surveys, analyses, and monitoring records shall be reviewed for trends, completeness, and accuracy.

6.2 Liquid Radioactive Waste Effluents

6.2.1 Liquid Effluent Concentration Limitations

a. Liquid waste concentrations from the site will not exceed the following applicable limits:

1. For radionuclides (other than dissolved or entrained noble gases) the concentration released in liquid effluents to UNRESTRICTED AREAS shall be limited to those specified in 10 CFR 20, Appendix B, Table II, Column 2.

2 For dissolved or entrained noble gases, the concentration shall be limited to 2E-4 µCi/ml.

b. If the concentration of liquid effluents released from the site exceed the above limits, promptly restore concentrations to within limits.

c. Daily concentrations of radioactive materials in liquid waste to UNRESTRICTED AREAS shall meet the following limitation:

$$\frac{\text{Volume of Waste Discharged + Volume of Dilution Water}}{\text{Volume of Waste Discharged} \times \sum_i \frac{\mu Ci/ml_i}{MPC_i}} \geq 1$$

where:

$\mu Ci/ml_i$ = the concentration of nuclide i in the liquid effluent discharge;

MPC_i = the maximum permissible concentration in UNRESTRICTED AREAS of nuclide, i, expressed as µCi/ml from 10CFR Part 20, Appendix B, Table II, for radionuclides other than noble gases and 2E-04 µCi/ml for dissolved or entrained noble gases.

6.2.2 Liquid Monitoring Instrumentation

a. Radioactive liquid effluent monitoring instrumentation channels shown on Attachments 1 and 2, Radioactive Liquid Effluent Monitoring Instrumentation (Surry and North Anna, respectively), shall be OPERABLE with their alarm/trip setpoints set to ensure that limits of step 6.2.1.a are not exceeded.

1. Alarm/trip setpoints of these channels shall be determined and adjusted in accordance with step 6.2.2.d, Setpoint Calculation.
2. If a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required by step 6.2.2.a, perform one of the following:
   - Promptly suspend release of radioactive liquid effluents monitored by affected channel
   - Declare the channel inoperable
   - Change the setpoint to an acceptable conservative value

b. Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Attachments 3 and 4, Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements (Surry and North Anna, respectively).

1. With the number of channels OPERABLE less than the minimum channels required by tables shown in Attachment 1 and 2, perform the ACTION shown in these tables.
2. Attempt to return the instruments to OPERABLE status within 30 days. If unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. **Applicable Monitors**

Liquid effluent monitors for which alarm/trip setpoints are determined are:

1. **Surry**

Release Point	Instrument Number
Liquid Radwaste Effluent Line	LW-108
Service Water System Effluent Line	SW-107
Circulating Water Discharge Line	SW-120, SW-220

2. **North Anna**

Release Point	Instrument Number
Liquid Radwaste Effluent Line	LW-111
Service Water System Effluent Line	SW-108
Condenser Circulating Water	SW-130, SW-230

d. **Setpoint Calculation**

NOTE: This methodology does not preclude the determination of more conservative setpoints.

1. Maximum setpoint values shall be calculated using the following equation:

$$c = \frac{CF}{f}$$

where:

c = the setpoint, in µCi/ml, of the radioactivity monitor measuring the radioactivity concentration in the effluent line prior to dilution;

C = the effluent concentration limit for this monitor used in implementing 10 CFR 20 for the Station, in µCi/ml;

f = the flow setpoint as measured at the radiation monitor location, GPM;

F = the dilution water flow calculated as:

(Surry) F = f + (200,000 GPM x Number of Circ. Pumps in Service)

(N. Anna) F = f + (218,000 GPM x Number of Circ. Pumps in Service)

2. Each of the condenser circulating water channels (Surry: SW-120, SW-220) (North Anna: SW-130, SW-230) monitors the effluent (service water including component cooling service water, circulating water, and liquid radwaste) in the circulating water discharge tunnel beyond the last point of possible radioactive material addition. No dilution is assumed for this pathway. Therefore, the equation in step 1 above becomes:

$$c = C$$

The setpoint for Station monitors used in implementing 10 CFR 20 for the site becomes the effluent concentration limit.

3. In addition, for added conservatism, setpoints are calculated for the liquid radwaste effluent line (Surry: LW-108, North Anna: LW-111) and the component cooling service water system effluent line (Surry: SW-107, North Anna: SW-108).

For the liquid radwaste effluent line, the equation in step 1 becomes:

$$c = \frac{CFK_{LW}}{f}$$

where;

K_{LW} = The fraction of the effluent concentration limit used in implementing 10CFR20 for the site attributable to liquid radwaste effluent line pathway.

For the service water system effluent line, the equation in step 1 becomes:

$$c = \frac{CFK_{SW}}{f}$$

where;

K_{SW} = The fraction of the effluent concentration limit used in implementing 10 CFR 20 for the Station attributable to the service water effluent line pathway.

The sum $K_{LW} + K_{SW} \leq 1.0$.

6.2.3 Liquid Effluent Dose Limit

a. Requirement

At least once per 31 days, perform the dose calculation in subsections 6.2.3.c and 6.2.3.d to ensure that the dose or dose commitment to the maximum exposed MEMBER OF THE PUBLIC from radioactive materials in liquid releases (from each reactor unit) to UNRESTRICTED AREAS shall be limited to the following:

1. During any calendar quarter to:
   - Less than or equal to 1.5 mrem to the total body
   - Less than or equal to 5 mrem to the critical organ
2. During any calendar year to:
   - Less than or equal to 3 mrem to the total body
   - Less than or equal to 10 mrem to the critical organ

b. Action

If the calculated dose from release of radioactive materials in liquid effluents exceeds any of the above limits, prepare and submit to the Commission within 30 days, a Special Report that identifies causes for exceeding limits and defines corrective actions taken to reduce releases of radioactive materials in liquid effluents to ensure that subsequent releases will be in compliance with the above limits.

c. Surry Dose Contribution Calculations

NOTE Thyroid and GI-LLI organ doses must be calculated to determine which is the critical organ for the period being considered.

Dose contributions shall be calculated for all radionuclides identified in liquid effluents released to UNRESTRICTED AREAS based on the following expression:

$$D = t\,F\,M \sum_i C_i A_i$$

where:

D = the cumulative dose commitment to the total body or critical organ, from the liquid effluents for the time period t, in mrem;

t = the length of the time period over which C_i and F are averaged for all liquid releases, hours;

M = the mixing ratio (reciprocal of the dilution factor) at the point of exposure, dimensionless, 0.2 from Appendix 11A, Surry UFSAR;

F = the near field average dilution factor for C_i during any liquid effluent release. Defined as the ratio of the average undiluted liquid waste flow during release to the average flow from the site discharge structure to UNRESTRICTED AREAS;

C_i = the average concentration of radionuclide, i, in undiluted liquid effluent during time period, t, from any liquid releases, in µCi/ml;

A_i = the site related ingestion dose commitment factor to the total body or critical organ of an adult for each identified principal gamma and beta emitter in mrem-ml per hr-µCi. Values for A_i are given in Attachment 5, Liquid Ingestion Pathway Dose Factors For Surry Power Station.

$$A_i = 1.14\text{ E+05 } (21BF_i + 5BI_i)\, DF_i$$

where:

1.14 E+05 = 1 E+06 pCi/µCi x 1 E+03 ml/kg ÷ 8760 hr/yr, units conversion factor;

21 = adult fish consumption, kg/yr, from NUREG-0133;

5 = adult invertebrate consumption, Kg/yr, from NUREG-0133;

BI_i = the bioaccumulation factor for nuclide, i, in invertebrates, pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109, Rev. 1;

BF_i = the bioaccumulation factor for nuclide, i, in fish, pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109, Rev. 1.

DF_i = the critical organ dose conversion factor for nuclide, i, for adults, in mrem/pCi, from Table E-11 of Regulatory Guide 1.109, Rev. 1.

d. **North Anna Dose Contribution Calculations**

NOTE: North Anna's dose contribution calculation for liquid effluents released to UNRESTRICTED AREAS has been modified. The derivation is given in Attachment 6, North Anna Liquid Ingestion Pathway Dose Factor Calculation.

Dose contribution shall be calculated for all radionuclides identified in liquid effluents released to UNRESTRICTED AREAS based on the following expressions:

$$D = \sum_i Q_i \times B_i$$

Where:

D = the cumulative dose commitment to the total body or critical organ, from the liquid effluents for the time period t, in mrem;

B_i = Dose Commitment Factors (mrem/Ci) for adults. Values for B_i are given in Attachment 7, North Anna Liquid Ingestion Pathway Dose Commitment Factors for Adults.

Q_i = Total released activity for the considered time period and the ith nuclide.

$$Q_i = t \times C_i \times \text{Waste Flow}$$

Where:

t = the length of the time period over which C_i and F are averaged for all liquid releases, hours;

C_i = the average concentration of radionuclide, i, in undiluted liquid effluent during time period, t, from any liquid releases, in µCi/ml;

e. **Quarterly Composite Analyses**

For radionuclides not determined in each batch or weekly composite, dose contribution to current monthly or calendar quarter cumulative summation may be approximated by assuming an average monthly concentration based on previous monthly or quarterly composite analyses. However, for reporting purposes, calculated dose contribution shall be based on the actual composite analyses.

6.2.4 Liquid Radwaste Treatment

a. Requirement

1. The Liquid Radwaste Treatment System shall be used to reduce the radioactive materials in liquid waste prior to discharge when projected dose due to liquid effluent, from each reactor unit, to UNRESTRICTED AREAS would exceed 0.06 mrem to total body or 0.2 mrem to the critical organ in a 31 day period.
2. Doses due to liquid releases shall be projected at least once per 31 days.

b. Action

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

1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or sub-system, and the reason for the inoperability.
2. Actions taken to restore inoperable equipment to OPERABLE status.
3. Summary description of actions taken to prevent a recurrence.

c. Projected Total Body Dose Calculation

1. Determine D_{TB} = total body dose from liquid effluents in the previous 31 day period, calculated according to subsection 6.2.3.c or d (Surry and North Anna, respectively).
2. Estimate R_1 = ratio of the estimated volume of liquid effluent releases in the present 31 day period to the volume released in the previous 31 day period.
3. Estimate F_1 = ratio of the estimated liquid effluent radioactivity in the present 31 day period to liquid effluent activity in the previous 31 day period (µCi/ml).
4. Determine PD_{TB} = projected total body dose in a 31 day period.

$$PD_{TB} = D_{TB}\,(R_1F_1)$$

d. **Projected Critical Organ Dose Calculation**

NOTE: Historical data pertaining to the volumes and radioactivity of liquid effluents released in connection with specific Station functions, such as maintenance or refueling outages, shall be used in projections as appropriate.

1. Determine D_o = critical organ dose from liquid effluents in the previous 31 day period, calculated according to subsection 6.2.3.c or d (Surry and North Anna, respectively).
2. Estimate R_1 as in step 6.2.4.c.2.
3. Estimate F_1 as in step 6.2.4.c.3.
4. Determine PD_o = projected critical organ dose in a 31 day period.

$$PD_o = D_o\ (R_1F_1)$$

6.2.5 Liquid Sampling

Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis requirements shown in Attachments 8 and 9, Radioactive Liquid Waste Sampling and Analysis Program (Surry and North Anna, respectively).

6.3 Gaseous Radioactive Waste Effluents

6.3.1 Gaseous Effluent Dose Rate Limitation

a. **Requirement**

Dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY are limited to the following:

1. The dose rate limit for noble gases shall be $\leq$ 500 mrem/year to the total body and $\leq$ 3000 mrem/year to the skin.
2. The dose rate limit for I-131, for tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days shall be $\leq$ 1500 mrem/year to the critical organ.

b. **Action**

1. If the dose rates exceed the above limits, promptly decrease the release rate to within the above limits.

2. Dose rates due to noble gases in gaseous effluents shall be determined continuously to be within the limits specified in subsection 6.3.1.a.

3. Dose rates due to I-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents shall be determined to be within the above limits by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified on Attachments 10 and 11, Radioactive Gaseous Waste Sampling and Analysis Program (Surry and North Anna, respectively).

c. **Calculations of Gaseous Effluent Dose Rates**

1. The dose rate limit for noble gases shall be determined to be within the limit by limiting the release rate to the lessor of:

- $\sum_i [K_{ivv} \dot{Q}_{ivv} + K_{ipv} \dot{Q}_{ipv}] \leq 500$ mrem/yr to the total body;

or,

- $\sum_i [(L_{ivv} + 1.1M_{ivv}) \dot{Q}_{ivv} + (L_{ipv} + 1.1M_{ipv}) \dot{Q}_{ipv}]$ ≤ 3000 mrem/yr to the skin.

where:

Subscripts = vv, refers to vent releases from the building ventilation vent;
pv, refers to the vent releases from the process vent;
i, refers to individual radionuclide;

K_{ivv}, K_{ipv} = The total body dose factor for ventilation vent or process vent release due to gamma emissions for each identified noble gas radionuclide, i, in mrem/yr per Curie/sec. Factors are listed in Attachments 12 and 13, Gaseous Effluent Dose Factors (Surry and North Anna, respectively).

L_{ivv}, L_{ipv} = The skin dose factor for ventilation vent or process vent release due to beta emissions for each identified noble gas radionuclide i, in mrem/yr per Curie/sec. Factors are listed in Attachments 12 and 13.

M_{ivv}, M_{ipv} = The air dose factor for ventilation vent or process vent release due to gamma emissions for each identified noble gas radionuclide, i, in mrad/yr per Curie/sec. Factors are listed in Attachments 12 and 13.

$\dot{Q}_{ivv}$, $\dot{Q}_{ipv}$ = The release rate for ventilation vent or process vent of noble gas radionuclide, i, in gaseous effluents in Curie/sec (per site);

1.1 = The unit conversion factor that converts air dose to skin dose, in mrem/mrad.

2. The dose rate limit for I-131, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days shall be determined to be within the limit by restricting the release rate to:

$$\sum_i [P_{ivv}\ \dot{Q}_{ivv} + P_{ipv}\ \dot{Q}_{ipv}] \leq 1500 \text{ mrem/yr to the critical organ.}$$

where:

P_{ivv}, P_{ipv} = The critical organ dose factor for ventilation vent or process vent for I-131, H-3, and all radionuclides in particulate form with half-lives greater than 8 days for the inhalation pathway, in mrem/yr per Curie/sec. Factors are listed in Attachments 12 and 13.

$\dot{Q}_{ivv}$, $\dot{Q}_{ipv}$ = The release rate for ventilation vent or process vent of I-131, H-3, and all radionuclides, i, in particulate form with half-lives greater than 8 days in gaseous effluents in Curie/sec (per site).

3. All gaseous releases, not through the process vent, are considered ground level and shall be included in the determination of Q_{ivv}.

6.3.2 Gaseous Monitoring Instrumentation

a. Requirement

1. The radioactive gaseous effluent monitoring instrumentation channels shown in Attachments 14 and 15, Radioactive Gaseous Effluent Monitoring Instrumentation (Surry and North Anna, respectively), shall be OPERABLE with alarm/trip setpoints set to ensure that limits specified for noble gases in subsection 6.3.1.a are not exceeded. Alarm/trip setpoints of these channels shall be determined and adjusted in accordance with subsection 6.3.2.d.

2. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Attachments 16 and 17, Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements (Surry and North Anna, respectively).

b. **Action**

1. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above requirement, promptly:
   - Suspend the release of radioactive gaseous effluents monitored by the affected channel; and
   - Declare the channel inoperable; or
   - Change the setpoint so it is acceptably conservative

2. With the number of channels OPERABLE less than the minimum channels required by tables shown in Attachment 14 and 15, take the ACTION shown in these tables.

3. Return the instruments to OPERABLE status within 30 days. If unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. **Applicable Monitors**

Radioactive gaseous effluent monitors for which alarm/trip setpoints are determined are:

1. **Surry**

Release Point	Instrument Number
Process Vent	GW-102, GW-130-1
Condenser Air Ejector	SV-111, SV-211
Ventilation Vent	VG-110, VG-131-1

2. **North Anna**

Release Point	Instrument Number
Process Vent	GW-102, GW-180-1
Condenser Air Ejector	SV-121, SV-221
Ventilation Vent A	VG-104, VG-178-1
Ventilation Vent B	VG-113, VG-179-1

d. **Setpoint Calculations**

1. The setpoint calculations for each monitor listed above shall be determined such that the following relationship is maintained:

$$D \geq D_{pv} + D_{cae} + D_{vv}$$

where:

D = Subsection 6.3.1.a dose limits implementing 10 CFR 20 for the Station, mrem/yr;

D_{pv} = The noble gas Station boundary dose rate from process vent gaseous effluent releases, mrem/yr;

D_{cae} = The noble gas Station boundary dose rate from condenser air ejector gaseous effluent releases, mrem/yr;

D_{vv} = The noble gas Station boundary dose rate from:

Surry: Ventilation vent gaseous effluent releases, mrem/yr

North Anna: Summation of ventilation vent A plus B gaseous effluent releases, mrem/yr

2. Setpoint values shall be determined using the following equation:

$$C_m = \frac{R_m \times 2.12\text{ E-03}}{F_m}$$

where:

m = The release pathway, process vent (pv), ventilation vent (vv) or condenser air ejector (cae);

C_m = The effluent concentration limit implementing subsection 6.3.1.a for the Station, µCi/ml;

R_m = The release rate limit for pathway m determined from methodology in subsection 6.3.1.c, using Xe-133 as nuclide to be released, µCi/sec;

2.12E-03 = CFM per ml/sec;

F_m =The maximum flow rate for pathway m, CFM.

3. According to NUREG-0133, the radioactive effluent radiation monitor alarm/trip setpoints should be based on the radioactive noble gases. It is not considered to be practicable to apply instantaneous alarm/ trip setpoints to integrating monitors sensitive to radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases.

6.3.3 Noble Gas Effluent Air Dose Limit

a. Requirement

1. The air dose in UNRESTRICTED AREAS due to noble gases released in gaseous effluents from each reactor unit from the site at and beyond the SITE BOUNDARY shall be limited to the following:
   - During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation.
   - During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.
2. Cumulative dose contributions for noble gases for the current calendar quarter and current calendar year shall be determined in accordance with subsection 6.3.3.c, Dose Calculations, at least once per 31 days.

b. Action

If the calculated air dose from radioactive noble gases in gaseous effluents exceeds any of the above limits, prepare and submit to the Commission within 30 days, a Special Report that identifies the causes for exceeding the limits and defines corrective actions that have been taken to reduce releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the limits stated in subsection 6.3.3.a.

c. **Noble Gas Effluent Air Dose Calculation**

NOTE: Gaseous releases, not through the process vent, are considered ground level and shall be included in the determination of Q_{ivv}.

1. The air dose to areas at or beyond the SITE BOUNDARY due to noble gases shall be determined by the following:

For gamma radiation:

$$Dg = 3.17E\text{-}08 \sum_i [M_{ivv} \bar{Q}_{ivv} + M_{ipv} \bar{Q}_{ipv}]$$

For beta radiation:

$$Db = 3.17E\text{-}08 \sum_i [N_{ivv} \bar{Q}_{ivv} + N_{ipv} \bar{Q}_{ipv}]$$

Where:

Subscripts = vv, refers to vent releases from the building ventilation vent.
pv, refers to the vent releases from the process vent
i, refers to individual radionuclide

Dg = the air dose for gamma radiation, in mrad

Db = the air dose for beta radiation, in mrad;

M_{ivv}, M_{ipv} = the air dose factors for ventilation vent or process vent release due to gamma emissions for each identified noble gas radionuclide, i, in mrad/yr per Curie/sec. Factors are given in Attachments 12 and 13.

N_{ivv}, N_{ipv} = the air dose factor for ventilation vent or process vent release due to beta emissions for each identified noble gas radionuclide, i, in mrad/yr per Curie/sec. Factors are listed in Attachments 12 and 13.

$\bar{Q}_{ivv}$, $\bar{Q}_{ipv}$ = the release for ventilation vent or process vent of noble gas radionuclide, i, in gaseous effluents for 31 days, quarter, or year as appropriate in Curie (per site);

6.3.4 I-131, H-3, and Radionuclides In Particulate Form Effluent Dose Limit

a. **Requirement**

1. Methods shall be implemented to ensure that the dose to any organ of a MEMBER OF THE PUBLIC from I-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the site to UNRESTRICTED AREAS from each reactor unit shall be limited to the following:
   - During any calendar quarter, to ≤ 7.5 mrem to the critical organ
   - During any calendar year, to ≤ 15 mrem to the critical organ.
2. Cumulative dose contributions to a MEMBER OF THE PUBLIC from I-131, tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to UNRESTRICTED AREAS for the current calendar quarter and current calendar year shall be determined in accordance with subsection 6.3.4.c, Surry Dose Calculations, or subsection 6.3.4.d, North Anna Dose Calculations, at least once per 31 days.

b. **Action**

If 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, prepare and submit to the Commission within 30 days, a Special Report containing the following:

1. Causes for exceeding limits.
2. Corrective actions taken to reduce releases.
3. Proposed corrective actions to be taken to assure that subsequent releases will be in compliance with limits stated in subsection 6.3.4.a.

c. **Surry Dose Calculations**

NOTE: Gaseous releases, not through process vent, are considered ground level and shall be included in the determination of $\tilde{Q}_{ivv}$.

1. The dose to the maximum exposed MEMBER OF THE PUBLIC from I-131, from tritium, and from all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY shall be determined as follows:

$$Dr = 3.17E\text{-}08 \sum_i [(RM_{ivv}\ \tilde{Q}_{ivv} + RM_{ipv}\ \tilde{Q}_{ipv}) + (RI_{ivv}\ \tilde{Q}_{ivv} + RI_{ipv}\ \tilde{Q}_{ipv})]$$

Where:

Subscripts	=	vv, refers to vent releases from the building ventilation vent; pv, refers to the vent releases from the process vent;
Dr	=	the dose to the critical organ of the maximum exposed MEMBER OF THE PUBLIC in mrem.
RM_{ivv}, RM_{ipv}	=	the milk pathway dose factor for ventilation vent or process vent release due to I-131, tritium, and from all radionuclides in particulate form with half-lives greater than 8 days, in mrem/yr per Curie/sec. Factors are listed in Attachment 18, Critical Organ and Inhalation Dose Factors For Surry.
RI_{ivv}, RI_{ipv}	=	the inhalation pathway dose factor for ventilation vent or process vent release due to I-131, tritium, and from all radionuclides in particulate form with half-lives greater than 8 days, in mrem/yr per Curie/sec. Factors are listed in Attachment 18.
$\tilde{Q}_{ivv}$, $\tilde{Q}_{ipv}$	=	the release for ventilation vent or process vent of I-131, tritium, and from all radionuclides in particulate from with half-lives greater than 8 days in Curies (per site).
3.17 E-08	=	the inverse of the number of seconds in a year.

d. North Anna Dose Calculations

NOTE: Gaseous releases, not through process vent, are considered ground level and shall be included in the determination of $\tilde{Q}_{ivv}$.

1. The dose to the maximum exposed MEMBER OF THE PUBLIC from I-131, from tritium, and from all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY shall be determined as follows:

$$Dr = 3.17E\text{-}08 \sum_i [R_{ivv} \tilde{Q}_{ivv} + R_{ipv} \tilde{Q}_{ipv}]$$

Where:

Subscripts	=	vv, refers to vent releases from the building ventilation vent; pv, refers to the vent releases from the process vent;
Dr	=	the dose to the critical organ of the maximum exposed MEMBER OF THE PUBLIC in mrem.
R_{ivv}, R_{ipv}	=	the dose factor for ventilation vent or process vent release due to I-131, tritium, and from all radionuclides in particulate form with half-lives greater than 8 days, in mrem/yr per Curie/sec. Factors are listed in Attachment 19, Critical Organ and Inhalation Dose Factors for North Anna.
$\tilde{Q}_{ivv}$, $\tilde{Q}_{ipv}$	=	the release for ventilation vent or process vent of I-131, tritium, and from all radionuclides in particulate from with half-lives greater than 8 days in Curies (per site).
3.17 E-08	=	the inverse of the number of seconds in a year.

6.3.5 Gaseous Radwaste Treatment

NOTE: Historical data pertaining to the volumes and radioactive concentrations of gaseous effluents released in connection to specific Station functions, such as containment purges, shall be used in the above estimates as appropriate.

a. **Requirement**

1. The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive material in gaseous waste prior to their discharge when 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 would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation averaged over 31 days.
2. 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 would exceed 0.3 mrem to the critical organ averaged over 31 days.
3. Doses due to gaseous releases from the site shall be projected at least once per 31 days based on calculations performed in subsections 6.3.5.c, d, and e.

b. **Action**

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

1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems and the reason for the inoperability.
2. Actions taken to restore the inoperable equipment to OPERABLE status.
3. Summary description of actions taken to prevent a recurrence.

c. **Projected Gamma Dose**

1. Determine D_g = the 31 day gamma air dose in the previous 31 day period calculated according to subsection 6.3.3.c.
2. Estimate R_g = ratio of the estimated volume of gaseous effluent in the present 31 day period to the volume released during the previous 31 day period.
3. Estimate F_g = ratio of the estimated noble gas effluent activity in the present 31 day period to the noble gas effluent activity during the previous 31 day period (µCi/ml).
4. Determine PDg = projected 31 day gamma air dose:

$$PD_g = D_g (R_g \times F_g)$$

d. **Projected Beta Dose**

1. Determine D_b = the 31 day beta air dose in the previous 31 day period, calculated according to subsection 6.3.3.c.
2. Estimate R_g and F_g as in steps 6.3.5.c.2 and 3 above.
3. Determine PD_g = projected 31 day period beta air dose:

$$PD_b = D_b (R_g \times F_g)$$

e. **Projected Maximum Exposed Member of the Public Dose**

1. Determine D_{max} = the 31 day maximum exposed MEMBER OF THE PUBLIC dose in the previous 31 day period, calculated according to subsection 6.3.4.c.
2. Estimate F_i = ratio of the estimated activity from I-131, radioactive materials in particulate form with half-lives greater than 8 days, and tritium in the present 31 day period to the activity of I-131, radioactive materials in particulate form with half-lives greater than 8 days, and tritium in the previous 31 day period (µC/ml).
3. Determine PD_{max} = projected 31 day maximum exposed MEMBER OF THE PUBLIC dose:

$$PD_{max} = D_{max} (R_g \times F_i)$$

6.4 Total Dose Limit to Public From Uranium Fuel Cycle Sources

6.4.1 Requirement

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).

6.4.2 Action

a. If the calculated doses from release of radioactive materials in liquid or gaseous effluents exceed twice the limits of Subsections 6.2.3.a, 6.3.3.a, or 6.3.4.a, calculations shall be made, including direct radiation contribution from the reactor units and from outside storage tanks, to determine whether limits of 6.4.1 have been exceeded.

b. If the limits of 6.4.1 have been exceeded, prepare and submit to the Commission within 30 days, 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 limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include the following:

   1. 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 calendar year that includes the releases covered by this report.
   2. A description of the levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations.
   3. If the estimated doses exceeds the limits of 6.4.1, 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.

6.5 Radiological Environmental Monitoring

6.5.1 Monitoring Program

a. Requirement

1. The Radiological Environmental Monitoring Program shall be conducted as specified in Attachments 20 and 21, Radiological Environmental Monitoring Program (Surry and North Anna, respectively).
2. Samples shall be collected from specific locations given in Attachments 22 and 23, Environmental Sample Locations (Surry and North Anna, respectively).
3. Samples shall be analyzed in accordance with:
   - Requirements of Attachments 20 and 21
   - Detection capabilities required by Attachments 24 and 25, Detection Capabilities for Environmental Sample Analysis (Surry and North Anna, respectively)
   - Guidance of the Radiological Assessment Branch Technical Position on Environmental Monitoring dated November, 1979, Revision No. 1.

b. Action

1. With the radiological environmental monitoring program not being conducted as required in 6.5.1.a, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Technical Specification (Surry T.S. 6.6.B.2) (North Anna T.S. 6.9.1.8), a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

2. If, when averaged over any calendar quarter, the level of radioactivity exceeds the reporting levels of Attachments 26 and 27, Reporting Levels for Radioactivity Concentrations in Environmental Samples (Surry and North Anna, respectively), prepare and submit to the Commission within 30 days, a Special Report that:
   - Identifies the causes for exceeding the limits; and
   - Defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of subsection 6.2.3, 6.3.3, and 6.3.4.

   When more than one of the radionuclides in Attachments 26 and 27 are detected in the sampling medium, this report shall be submitted if:

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

3. When radionuclides other than those listed in Attachment 26 and 27 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of subsections 6.2.3, 6.3.3, and 6.3.4. 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.

4. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Attachment 20 and 21, 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. Identify the cause of the unavailability of samples and identify the new locations for obtaining replacement samples in the next Semi-annual Radioactive Effluent Release Report. Include in the report a revised figure and table for the ODCM reflecting the new locations.

6.5.2 Land Use Census

a. Requirement

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 following:

- Nearest milk animal
- Nearest residence
- Nearest garden of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation

1. The land use census shall be conducted during the growing season at least once per 12 months using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. Results of the land use census shall be included in the Annual Radiological Environmental Operating Report.
2. 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. Specifications for broad leaf vegetation sampling given in Attachments 20 and 21 shall be followed, including analysis of control samples.

b. Action

1. With a land use census identifying locations that yield a calculated dose or dose commitment greater than the values currently being calculated in step 6.3.4.a.2, identify the new locations in the next Semiannual Radioactive Effluent Release Report.
2. With a land use census identifying locations 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, add the new locations to the Radiological Environmental Monitoring Program within 30 days. The sampling locations, excluding the control station location, having the lowest calculated dose or dose commitments (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. Identify the new locations in the next Semiannual Radioactive Effluent Release Report and also include in the report revised figures and tables reflecting the new locations.

6.5.3 Interlaboratory Comparison Program

a. Requirement

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.

b. Action

1. Analyses shall be performed as part of the Environmental Protection Agency's Environmental Radioactivity Laboratory Intercomparison Studies (Cross Check) Program and include:

Program	Cross-Check Of:
Milk	I-131, Gamma, K, Sr-89 and 90
Water	Gross Beta, Gamma, I-131, H-3 (Tritium), Sr-89/90, Blind - any combinations of above radionuclides.
Air Filter	Gross Beta, Gamma, Sr-90

2. 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.

c. Methodology and Results

1. Methodology and results of the cross-check program shall be maintained in the contractor supplied Nuclear Reactor Environmental Radiation Monitoring Quality Control Manual, IWL-0032-361.
2. Results will be reported in the Annual Radiological Environmental Monitoring Report.

6.6 REPORTING REQUIREMENTS

6.6.1 Annual Radiological Environmental Operating Report

Routine Radiological Environmental Operating Reports covering the operation of the units during the previous calendar year shall be submitted prior to May 1 of each year. A single submittal may be made for the Station. Radiological Environmental Operating Reports shall include:

a. Summaries, interpretations, and analysis of trends of results of radiological environmental surveillance activities for the report period, including:

- A comparison (as appropriate) with preoperational studies, operational controls, and previous environmental surveillance reports
- An assessment of the observed impacts of the plant operation on the environment
- Results of land use census per subsection 6.5.2, Land Use Census

b. Results of analysis of radiological environmental samples and of environmental radiation measurements taken per subsection 6.5.1, Monitoring Program. Results shall be summarized and tabulated in the format of the table in the Radiological Assessment Branch Technical Position (Reference 3.1.11).

1. If some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining reasons for missing results.
2. Missing data shall be submitted as soon as possible in a supplementary report.

c. A summary description of the radiological environmental monitoring program.

d. At least two legible maps covering sampling locations keyed to a table giving distances and directions from the centerline of one reactor. One map shall cover stations near the SITE BOUNDARY; a second shall include more distant stations.

e. Results of Station's participation in the Interlaboratory Comparison Program, per Subsection 6.5.3, Interlaboratory Comparison Program.

f. Discussion of deviations from the Station's environmental sampling schedule per Attachment 20 or 21 (as appropriate).

g. Discussion of analyses in which the lower limit of detection (LLD) required by Attachment 24 or 25 (as appropriate) was not achievable.

6.6.2 Semiannual Radioactive Effluent Release Report

a. Requirement

Radioactive Effluent Release Reports covering operation of the units during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. A single submittal may be made for the Station and should combine those sections that are common to both units. Radioactive Effluent Release Reports shall include:

1. A summary of quantities of radioactive liquid and gaseous effluents and solid waste released. Data shall be summarized on a quarterly basis following the format of Regulatory Guide 1.21, Appendix B (Reference 3.1.5).
2. 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 Station during the previous calendar year. This assessment shall be performed in accordance with subsection 6.6.2.b, Dose Assessment, and shall only be included in Radioactive Effluent Release Reports submitted within 60 days after January 1 of each year.
3. A list of unplanned releases from the site to UNRESTRICTED AREAS occurring during the reporting period that exceed the limits set forth in subsections 6.2.1, Liquid Effluent Concentration Limitations, and 6.3.1, Gaseous Effluent Dose Rate Limitation.
4. Major changes made during the reporting period to radioactive liquid, gaseous, and solid waste treatment systems.
5. Changes made to VPAP-2103, Offsite Dose Calculation Manual (see subsection 6.6.4, Changes to the ODCM).
6. A listing of new locations for dose calculations or environmental monitoring identified by the Land Use Census (Subsection 6.5.2).

b. Dose Assessment

1. Radiation doses to individuals due to radioactive liquid and gaseous effluents from the Station during the previous calendar year shall either be calculated in accordance with this procedure or in accordance with Regulatory Guide 1.109. Population doses shall not be included in dose assessments.

2. The dose to the maximum exposed MEMBER OF THE PUBLIC due to the radioactive liquid and gaseous effluents from the Station shall be incorporated with the dose assessment performed above. If the dose to the maximum exposed MEMBER OF THE PUBLIC exceeds twice the limits of Subsections 6.2.3.a.1, 6.2.3.a.2, 6.3.3.a.1, or 6.3.4.a.1, the dose assessment shall include the contribution from direct radiation.

NOTE: NUREG-0543 (Reference 3.1.13), states "There is reasonable assurance that sites with up to four operating reactors that have releases within Appendix I design objective values are also in conformance with the EPA Uranium Fuel Cycle Standard, 40 CFR Part 190".

3. The meteorological conditions during the previous calendar year or historical annual average atmospheric dispersion conditions shall be used for determining the gaseous pathway doses.

6.6.3 Annual Meteorological Data

a. Meteorological data collected over the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

b. Meteorological data shall be retained in a file on site and shall be made available to the NRC upon request.

6.6.4 Changes to the ODCM

Changes to the ODCM shall be:

a. Reviewed and approved by Station Nuclear Safety and Operating Committee (SNSOC) prior to implementation.

b. Documented and records of reviews performed shall be retained as Station records. Documentation shall include:

1. Sufficient information to support the change together with appropriate analyses or evaluations justifying changes.

2. A determination that the change will not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations and will maintain the level of radioactive effluent control required by:
   - 10 CFR 20.106
   - 40 CFR Part 190
   - 10 CFR 50.36a
   - 10 CFR Part 50, Appendix I

c. Submitted to the NRC in the form of a complete legible copy of the entire ODCM as a part of, or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

7.0 RECORDS

The following individual/packaged documents and related correspondence completed as a result of the performance or implementation of this procedure are records. Records shall be transmitted to Records Management in accordance with VPAP-1701, Records Management. These records shall include, but are not be limited to, the following:

- Records of changes to the ODCM in accordance with subsection 6.6.4
- Records of meteorological data in accordance with subsection 6.6.3
- Records of sampling and analyses
- Records of radioactive materials and other effluents released to the environment
- Records of maintenance, surveillances, and calibrations

ATTACHMENT 1

(Page 1 of 1)

SURRY RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
(a) Liquid Radwaste Effluent Line	1	1
2. GROSS BETA OR GAMA 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 Attachment 8, Surry Radioactive Liquid Waste Sampling and Analysis Program.

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.

ATTACHMENT 2

(Page 1 of 2)

NORTH ANNA RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
(a) Liquid Radwaste Effluent Line	1	1
2. GROSS BETA OR GAMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
(a) Service Water System Effluent Line	1	1
(b) Circulating Water System Effluent Line	1	4
3. FLOW RATE MEASUREMENT DEVICES		
(a) Liquid Radwaste Effluent Line	1	2
4. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
(a) Clarifier Effluent Line	1	1
5. TANK LEVEL INDICATING DEVICES (Note 1)		
(a) Refueling Water Storage Tanks	1	3
(b) Casing Cooling Storage Tanks	1	3
(c) PC Water Storage Tanks (Note 2)	1	3
(d) Boron Recovery Test Tanks (Note 2)	1	3

ATTACHMENT 2

(Page 2 of 2)

NORTH ANNA RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

ACTION 1: 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 gross radioactivity (beta or gamma) at a lower limit of detection of at least 1×10^{-7} µCi/g or an isotopic radioactivity at a lower limit of detection of at least 5×10^{-7} µCi/g.

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 the flow rate is estimated at least once per 4 hours during actual releases. Design capacity performance curves generated in situ may be used to estimate flow.

ACTION 3: With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

ACTION 4: With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, make repairs as soon as possible. Grab samples cannot be obtained via this pathway.

NOTE 1: Tanks included in this requirement are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

NOTE 2: This is a shared system with Unit 2.

ATTACHMENT 3

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SURRY 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.

ATTACHMENT 4

(Page 1 of 2)

NORTH ANNA 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	D	R	Q (Note 1)
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
(a) Service Water System Effluent Line	D	M	R	Q (Note 2)
(b) Circulating Water System Effluent Line	D	M	R	Q (Note 2)
3. FLOW RATE MEASUREMENT DEVICES				
(a) Liquid Radwaste Effluent Line	D (Note 3)	N.A.	R	Q
4. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR				
(a) Clarifier Effluent Line	N.A.	N.A.	R	N.A.
5. TANK LEVEL INDICATING DEVICES (Note 6)				
(a) Refueling Water Storage Tank	D (Note 4)	N.A.	R	Q
(b) Casing Cooling Storage Tank	D (Note 4)	N.A.	R	Q
(c) PC Water Storage Tanks (Note 5)	D (Note 4	N.A.	R	Q
(d) Boron Recovery Test Tanks (Note 5)	D (Note 4)	N.A.	R	Q

ATTACHMENT 4

(Page 2 of 2)

NORTH ANNA RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTE 1: The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Instrument controls not set in operate mode.

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

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Instrument controls not set in operate mode.

NOTE 3: CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be make at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

NOTE 4: During liquid additions to the tank.

NOTE 5: This is a shared system with Unit 2.

NOTE 6: Tanks included in this requirement are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

ATTACHMENT 5

(Page 1 of 1)

LIQUID INGESTION PATHWAY DOSE FACTORS FOR SURRY STATION UNITS 1 AND 2

Radionuclide	Total Body A_i $\frac{mrem/hr}{\mu Ci/ml}$	Thyroid A_i $\frac{mrem/hr}{\mu Ci/ml}$	GI-LLI A_i $\frac{mrem/hr}{\mu Ci/ml}$
H-3	2.82E-01	2.82E-01	2.82E-01
Na-24	4.57E-01	4.57E-01	4.57E-01
Cr-51	5.58E+00	3.34E-01	1.40E+03
Mn-54	1.35E+03	-	2.16E+04
Fe-55	8.23E+03	-	2.03E+04
Fe-59	7.27E+04	-	6.32E+05
Co-58	1.35E+03	-	1.22E+04
Co-60	3.82E+03	-	3.25E+04
Zn-65	2.32E+05	-	3.23E+05
Rb-86	2.91E+02	-	1.23E+02
Sr-89	1.43E+02	-	8.00E+02
Sr-90	3.01E+04	-	3.55E+03
Y-91	2.37E+00	-	4.89E+04
Zr-95	3.46E+00	-	1.62E+04
Zr-97	8.13E-02	-	5.51E+04
Nb-95	1.34E+02	-	1.51E+06
Mo-99	2.43E+01	-	2.96E+02
Ru-103	4.60E+01	-	1.25E+04
Ru-106	2.01E+02	-	1.03E+05
Ag-110m	8.60E+02	-	5.97E+05
Sb-124	1.09E+02	6.70E-01	7.84E+03
Sb-125	4.20E+01	1.79E-01	1.94E+03
Te-125m	2.91E+01	6.52E+01	8.66E+02
Te-127m	6.68E+01	1.40E+02	1.84E+03
Te-129m	1.47E+02	3.20E+02	4.69E+03
Te-131m	5.71E+01	1.08E+02	6.80E+03
Te-132	1.24E+02	1.46E+02	6.24E+03
I-131	1.79E+02	1.02E+05	8.23E+01
I-132	9.96E+00	9.96E+02	5.35E+00
I-133	3.95E+01	1.90E+04	1.16E+02
I-134	5.40E+00	2.62E+02	1.32E-02
I-135	2.24E+01	4.01E+03	6.87E+01
Cs-134	1.33E+04	-	2.85E+02
Cs-136	2.04E+03	-	3.21E+02
Cs-137	7.85E+03	-	2.32E+02
Cs-138	5.94E+00	-	5.12E-05
Ba-140	1.08E+02	-	3.38E+03
La-140	2.10E-01	-	5.83E+04
Ce-141	2.63E-01	-	8.86E+03
Ce-143	4.94E-02	-	1.67E+04
Ce-144	9.59E+00	-	6.04E+04
Np-239	1.91E-03	-	7.11E+02

ATTACHMENT 6

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NORTH ANNA LIQUID INGESTION PATHWAY DOSE FACTOR CALCULATION

UNITS 1 AND 2

1.0 EXPRESSION "1"

$$D = t\,F \sum_i f_i\, C_i\, A_i$$

where:

D = the cumulative dose commitment to the total body or critical organ, from the liquid effluents for the time period t, in mrem;

t = the length of time period over which C_i and F are averaged for all liquid releases, hours;

F = the near field average dilution factor for C_i during any liquid effluent release. Defined as the ratio of the average undiluted liquid waste flow during release to the average flow from the Station discharge structure to UNRESTRICTED AREAS;

f_i = the individual dilution multiplication factor to account for increases in concentration of long-lived nuclides due to recirculation, listed on page 4 of 4 of this attachment. "f_i" is the ratio of the total dilution flow over the effective dilution flow.

C_i = the average concentration of radionuclide, i, in undiluted liquid effluent during time period, t, from any liquid releases, in μCi/ml;

A_i = the site related ingestion dose commitment factor to the total body or critical organ of an adult for each identified principal gamma and beta emitter listed on page 4 of 4 of this attachment, in mrem-ml per hr-μCi;

$$A_i = 1.14\ \mathrm{E{+}05}\ (730/D_w\ +\ 21BF_i/D_a)\ DF_i$$

where:

1.14 E+05 = 1 E+06 pCi/μCi x 1 E+03 ml/kg ÷ 8760 hr/yr, units conversion factor;

730 = adult water consumption, kg/yr, from NUREG-0133;

ATTACHMENT 6

(Page 2 of 4)

NORTH ANNA LIQUID INGESTION PATHWAY DOSE FACTOR CALCULATION

UNITS 1 AND 2

D_w = dilution factor from the near field area within one-quarter mile of the release point to the potable water intake for the adult water consumption. Dw includes the dilution contributions from the North Anna Dam to Doswell (0.73), the Waste Heat Treatment Facility (C_c/C_L), and Lake Anna (C_L/C_R). The potable water mixing ratio is calculated as:

$$1 / (C_c / C_L) (C_L / C_R \times 0.73 = C_R / (C_c \times 0.73)$$

where C_c / C_L and C_R are the respective concentrations for the considered nuclide in the Discharge Channel, Waste Heat Treatment Facility (Lagoon) and the Reservoir. Calculation is per expressions 11.2-5, 11.2-6, and 11.2-8 of North Anna's UFSAR.

BF_i = the bioaccumulation factor for nuclide, i, in fish, pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109, Rev. 1.

D_a = dilution factor for the fish pathway, calculated as C_L / C_c where C_L and C_c are the concentrations for the considered nuclide in the Discharge Channel and the Waste Heat Treatment Facility (Lagoon). Calculation is per Expressions 11.2-5, and 11.2-6 of North Anna's UFSAR.

DF_i = the critical organ dose conversion factor for nuclide, i, for adults, in mrem/pCi, from Table E-11 of Regulatory Guide 1.109, Rev. 1.

ATTACHMENT 6

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NORTH ANNA LIQUID INGESTION PATHWAY DOSE FACTOR CALCULATION

UNITS 1 AND 2

2.0 EXPRESSION "2"

Expression "1" is simplified for actual dose calculations by introducing:

$$F = \frac{\text{WASTE FLOW}}{\text{CIRC.(WATER) FLOW + WASTE FLOW}} \approx \frac{\text{WASTE FLOW}}{\text{CIRC. FLOW}}$$

and

$$f_i = \frac{\text{CIRC. FLOW}}{\text{EFFECTIVE DIL. FLOW}_i}$$

Effective dilution flow rates for individual nuclides "i" are listed on Attachment 7, North Anna Liquid Pathway Dose Commitment Factors for Adults. Then the total released activity (Q_i) for the considered time period and the ith nuclide is written as:

$$Q_i = t \times C_i \times \text{WASTE FLOW}$$

and Expression "1" reduces to:

$$D = \sum_i Q_i \frac{A_i}{\text{EFF. DIL. FLOW}_i}$$

For the long lived, dose controlling nuclides the effective dilution flow is essentially the over (dam) flow rate out of the North Anna Lake system (i.e., the liquid pathway dose is practically independent from the circulating water flow rate. However, to accurately assess long range average effects of reduced circulating water flow rates during outages or periods of low lake water temperatures, calculations are based on an average of 7 out of 8 circulating water pumps running at 218,000 gpm = 485.6 cft/sec per pump.

By defining $B_i = A_i / \text{EFF. DIL. FLOW}_i$, the dose calculation is reduced to a two factor formula:

$$D = \sum_i Q_i \times B_i$$

Values for B_i (mrem/Ci) and EFF. DIL. FLOW_i are listed in Attachment 7.

ATTACHMENT 6

(Page 4 of 4)

NORTH ANNA LIQUID INGESTION PATHWAY DOSE FACTOR CALCULATION
UNITS 1 AND 2

Radionuclide	Individual Dilution Multiplication Factor (f_i)	Total Body A_i $\frac{mrem/hr}{\mu Ci/ml}$	Critical Organ A_i $\frac{mrem/hr}{\mu Ci/ml}$
H-3	14.9	6.18E+00	6.18E+00
Na-24	1.0	3.71E+01	3.71E+01
Cr-51	1.7	1.10E+00	-
Mn-54	7.0	8.62E+02	4.52E+03
Fe-55	11.3	1.30E+02	5.56E+02
Fe-59	2.2	9.47E+02	2.47E+03
Co-58	2.8	2.49E+02	1.11E+02
Co-60	13.3	8.27E+02	3.75E+02
Zn-65	6.1	3.28E+04	7.25E+04
Rb-86	1.5	3.53E+04	7.59E+04
Sr-89	2.3	8.70E+02	-
Sr-90	15.8	2.39E+05	-
Y-91	2.5	3.42E-01	-
Zr-95	2.7	2.98E-01	-
Zr-97	1.0	1.50E-04	3.27E-04
Nb-95	1.0	4.87E+01	9.07E+01
Mo-99	1.0	7.48E+00	3.93E+01
Ru-103	2.0	4.10E+00	-
Ru-106	7.6	2.65E+01	-
Ag-110m	6.2	4.94E+00	8.32E+00
Sb-124	2.6	4.37E+01	2.08E+00
Sb-125	11.4	2.46E+01	1.16E+00
Te-125m	2.5	3.23E+02	8.73E+02
Te-127m	3.7	7.82E+02	2.29E+03
Te-129m	1.9	1.52E+03	3.58E+03
Te-131m	1.0	1.12E+02	1.35E+02
Te-132	1.0	5.04E+02	5.37E+02
I-131	1.2	9.66E+01	1.69E+02
I-132	1.0	1.03E-01	2.95E-01
I-133	1.0	3.47E+00	1.14E+01
I-134	1.0	2.15E-02	6.00E-02
I-135	1.0	6.58E-01	1.78E+00
Cs-134	10.3	5.80E+05	7.09E+05
Cs-136	1.3	6.01E+04	8.35E+04
Cs-137	15.8	3.45E+05	5.26E+05
Cs-138	1.0	9.18E-01	1.85E+00
Ba-140	1.3	2.65E+01	5.08E-01
La-140	1.0	4.47E-03	1.69E-02
Ce-141	1.8	2.14E-02	1.89E-01
Ce-143	1.0	1.35E-04	1.22E+00
Ce-144	6.6	1.41E+00	1.10E+01
Np-239	1.0	5.13E-04	9.31E-04

ATTACHMENT 7

(Page 1 of 1)

NAPS LIQUID PATHWAY DOSE COMMITMENT FACTORS FOR ADULTS

(B_i = A_i Fi/CIRC FLOW = A_i/Effluent Dilution $Flow_i$)

Radionuclide	Effective Dilution Flow (cft/sec)	Total Body B_i (mrem/Ci)	Critical Organ B_i (mrem/Ci)
H-3	2.28E+02	2.66E-04	2.66E-04
Na-24	3.39E+03	1.07E-04	1.07E-04
Cr-51	1.99E+03	5.44E-06	N/A
Mn-54	4.88E+02	1.73E-02	9.08E-02
Fe-55	3.01E+02	4.23E-03	1.81E-02
Fe-59	1.57E+03	5.93E-03	1.55E-02
Co-58	1.20E+03	2.04E-03	9.10E-04
Co-60	2.55E+02	3.18E-02	1.44E-02
Zn-65	5.60E+02	5.74E-01	1.27E+00
Rb-86	2.34E+03	1.48E-01	3.18E-01
Sr-89	1.46E+03	5.84E-03	N/A
Sr-90	2.16E+02	1.09E+01	N/A
Y-91	1.34E+03	2.50E-06	N/A
Zr-95	1.27E+03	2.30E-06	1.31E-06
Zr-97	3.39E+03	4.33E-10	9.46E-10
Nb-95	3.25E+03	1.47E-04	2.74E-04
Mo-99	3.30E+03	2.22E-05	1.17E-04
Ru-103	1.68E+03	2.40E-05	N/A
Ru-106	4.48E+02	5.80E-04	N/A
Ag-110m	5.52E+02	8.78E-05	1.48E-04
Sb-124	1.32E+03	3.25E-04	1.55E-05
Sb-125	2.98E+02	8.10E-04	3.80E-05
Te-125m	1.35E+03	2.35E-03	6.35E-03
Te-127m	9.16E+02	8.37E-03	2.46E-02
Te-129m	1.82E+03	8.19E-03	1.93E-02
Te-131m	3.38E+03	3.27E-04	3.92E-04
Te-132	3.27E+03	1.51E-03	1.61E-03
I-131	2.94E+03	3.22E-04	5.62E-04
I-132	3.40E+03	2.98E-07	8.51E-07
I-133	3.39E+03	1.00E-05	3.29E-05
I-134	3.40E+03	6.19E-08	1.73E-07
I-135	3.40E+03	1.90E-06	5.15E-06
Cs-134	3.29E+02	1.73E+01	2.11E+01
Cs-136	2.62E+03	2.25E-01	3.12E-01
Cs-137	2.15E+02	1.57E+01	2.40E+01
Cs-138	3.40E+03	2.65E-06	5.34E-06
Ba-140	2.65E+03	9.83E-05	1.88E-06
La-140	3.36E+03	1.31E-08	4.94E-08
Ce-141	1.85E+03	1.14E-07	1.00E-06
Ce-143	3.37E+03	3.93E-10	3.55E-06
Ce-144	5.14E+02	2.70E-05	2.10E-04
Np-239	3.32E+03	1.51E-09	2.75E-09

ATTACHMENT 8

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SURRY RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml), (Note 1)
A. Batch Releases (Note 2)	P (Each Batch)	P (Each Batch)	Principal Gamma Emitters (Note 3)	$5x10^{-7}$
			I-131	$1x10^{-6}$
	P (One Batch/M)	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	P (Each Batch)	M Composite (Note 4)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P (Each Batch)	Q Composite (Note 4)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
B. Continuous Releases (Note 5)	Continuous (Note 6)	W Composite (Note 6)	Principal Gamma Emitters (Note 6)	$5x10^{-7}$
			I-131	$1x10^{-6}$
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	Continuous (Note 6)	M Composite (Note 6)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continuous (Note 6)	Q Composite (Note 6)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

ATTACHMENT 8

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SURRY RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Note 1: The LLD is defined, for purposes of this requirement, 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):

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

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume).

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

E = the counting efficiency (as counts per disintegration).

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

2.22×10^6 = the number of disintegrations per minute (dpm) per microcurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

Δt = 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.

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.

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SURRY RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Note 2: A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and appropriate methods will be used to obtain representative sample for analysis.

Note 3: 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.

Note 4: A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

Note 5: 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.

Note 6: 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 specimen representative of the effluent release.

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NORTH ANNA RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml), (Note 1)
Batch Releases (Notes 2 and 7)	P (Each Batch)	P (Each Batch)	Principal Gamma Emitters (Note 3)	$5x10^{-7}$
			I-131	$1x10^{-6}$
	P (One Batch/M)	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	P (Each Batch)	M Composite (Note 4)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P (Each Batch)	Q Composite (Note 4)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
Continuous Releases (Note 5)	Continuous (Note 6)	W Composite (Note 6)	Principal Gamma Emitters (Note 6)	$5x10^{-7}$
			I-131	$1x10^{-6}$
			Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	Continuous (Note 6)	M Composite (Note 6)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continuous (Note 6)	Q Composite (Note 6)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

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NORTH ANNA RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Note 1: The LLD is defined, for purposes of this requirement, 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):

$$LLD = \frac{4.66\ s_b}{E \bullet V \bullet 2.22 \times 10^6 \bullet Y \bullet e^{(-\lambda\Delta\tau)}}$$

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume).

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

E = the counting efficiency (as counts per disintegration).

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

2.22×10^6 = the number of disintegrations per minute (dpm) per microcurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

Δt = 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.

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.

ATTACHMENT 9

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NORTH ANNA RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Note 2: A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed as the situation permits, to assure representative sampling.

Note 3: 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.

Note 4: A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

Note 5: 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.

Note 6: To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent releases.

Note 7: Whenever the secondary coolant activity exceeds 10^{-5} μCi/ml, the turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

ATTACHMENT 10

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SURRY RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml), (Note 1)
A. Waste Gas Storage Tank	Prior to release. (Each Tank) (Grab Sample)	Prior to release. (Each Tank)	Principal Gamma Emitters (Note 2)	$1x10^{-4}$
B. Containment PURGE	Prior to release. (Each PURGE) (Grab Sample)	Prior to release. (Each PURGE)	Principal Gamma Emitters (Note 2)	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. Process and Ventilation Vent	Weekly (Grab Sample) (Note 3)	Weekly (Note 3)	Principal Gamma Emitters (Note 2)	$1x10^{-4}$
			H-3	$1x10^{-6}$
D. All Release Types as listed in A, B, and C.	Continuous (Note 4)	Weekly (Note 5) (Charcoal Sample)	I-131	$1x10^{-12}$
	Continuous (Note 4)	Weekly (Note 5) Particulate Sample	Principal Gamma Emitters (Note 2)	$1x10^{-11}$
	Continuous (Note 4)	Weekly Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous (Note 4)	Quarterly Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous (Note 4)	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	$1x10^{-6}$
E. Condenser Air Ejector	Weekly Grab Sample (Note 3)	Weekly (Note 3)	Principle Gamma Emitters (Note 2)	$1x10^{-4}$
			H-3	$1x10^{-6}$

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SURRY RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

F. Containment Hog Depressurization	Prior to release. (Grab Sample)	Prior to release. (Each release	Principle Gamma Emitters	$1x10^{-4}$
			H-3	$1x10^{-6}$
	Continuous (Note 4)	Charcoal Sample (Note 6)	I-131	$1x10^{-11}$
	Continuous (Note 4)	Particulate Sample (Note 6)	Principle Gamma Emitters (Note 2)	$1x10^{-10}$
	Continuous (Note 4)	Composite Particulate Sample (Note 6)	Gross Alpha	$1x10^{-10}$
	Continuous (Note 4)	Composite Particulate Sample (Note 6)	Sr-89, Sr-90	$1x10^{-10}$

ATTACHMENT 10

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SURRY RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Note 1: The LLD is defined, for purposes of this requirement, 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):

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

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume).

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

E = the counting efficiency (as counts per disintegration).

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

2.22×10^6 = the number of disintegrations per minute (dpm) per microcurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

Δt = 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.

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.

ATTACHMENT 10

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SURRY RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Note 2: 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 lives greater than 8 days, that are measurable and identifiable at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

Note 3: Sampling and analysis shall also be performed following shutdown, startup, and whenever a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER occurs within a one hour period, When:

a. Analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and

b. The noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

Note 4: 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 subsections 6.3.1, 6.3.3, and 6.3.4.

Note 5: Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of charging. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement applies if:

a. Analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by a factor of 3; and

b. Noble gas monitor shows that effluent activity has increased more than a factor of 3.

Note 6: 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.

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NORTH ANNA RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (µCi/ml), (Note 1)
A. Waste Gas Storage Tank	Prior to release. (Each Tank Grab Sample)	Prior to release. (Each Tank)	Principal Gamma Emitters (Note 2)	1×10^{-4}
B. Containment PURGE	Prior to release. (Each PURGE Grab Sample)	Prior to release. (Each PURGE)	Principal Gamma Emitters (Note 2)	1×10^{-4}
			H-3	1×10^{-6}
C. Ventilation (1) Process Vent (2) Vent. Vent A (3) Vent. Vent B	Monthly (Grab Sample) (Notes 3,4, and 5)	Monthly (Note 3)	Principal Gamma Emitters (Note 2)	1×10^{-4}
			H-3	1×10^{-6}
D. All Release Types as listed in A, B, and C.	Continuous (Note 4)	Weekly (Charcoal Sample)	I-131	1×10^{-12}
	Continuous (Note 4)	Weekly Particulate Sample	Principal Gamma Emitters (Note 2)	1×10^{-11}
	Continuous (Note 4)	Monthly Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous (Note 4)	Quarterly Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous (Note 4)	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}
E. Cond. Air Ejector Vent Steam Gen. Blowdown Vent	Weekly (Grab Sample)	Weekly	Principle Gamma Emitters (Note 7)	1×10^{-4}
			H-3	1×10^{-6}
F. Containment Vacuum Steam Ejector (Hogger)	Prior to release. (Grab Sample)	Prior to each release	Principle Gamma Emitters (Note 2)	1×10^{-4}
			H-3	1×10^{-6}

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NORTH ANNA RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Note 1: The LLD is defined, for purposes of this requirement, 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):

$$LLD = \frac{4.66\ s_b}{E \bullet V \bullet 2.22 \times 10^6 \bullet Y \bullet e^{(-\lambda \Delta \tau)}}$$

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume).

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

E = the counting efficiency (as counts per disintegration).

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

2.22×10^6 = the number of disintegrations per minute (dpm) per microcurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

Δt = 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.

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.

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NORTH ANNA RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Note 2: 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 peaks that are measurable and identifiable, at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

Note 3: Sampling and analysis shall also be performed following shutdown, startup, and whenever a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER occurs within a one hour period, if:

a. Analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant is greater than 1.0 μCi/gm; and

b. The noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

Note 4: 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 subsections 6.3.1, 6.3.3, and 6.3.4.

Note 5: Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of charging. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement applies if:

a. Analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant is greater than 1.0 μCi/gm and;

b. Noble gas monitor shows that effluent activity has increased more than a factor of 3.

Note 6: Whenever the secondary coolant activity exceeds 10^{-5} μCi/ml, samples shall be obtained and analyzed weekly. The turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

Note 7: 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. 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.

ATTACHMENT 12

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GASEOUS EFFLUENT DOSE FACTORS FOR SURRY POWER STATION

(Gamma and Beta Dose Factors)

χ/Q = 6.0E-05 sec/m^3 at 499 meters N Direction

Dose Factors for Ventilation Vent

Noble Gas Radionuclide	K_{ivv} Total Body mrem/yr / Curie/Sec	L_{ivv} Skin mrem/yr / Curie/Sec	M_{ivv} Gamma Air mrad/yr / Curie/Sec	N_{ivv} Beta Air mrad/yr / Curie/Sec
Kr-83m	4.54E+00	-	1.16E+03	1.73E+04
Kr-85m	7.02E+04	8.76E+04	7.38E+04	1.18E+05
Kr-85	9.66E+02	8.04E+04	1.03E+03	1.17E+05
Kr-87	3.55E+05	5.84E+05	3.70E+05	6.18E+05
Kr-88	8.82E+05	1.42E+05	9.12E+05	1.76E+05
Kr-89	9.96E+05	6.06E+05	1.04E+06	6.36E+05
Kr-90	9.36E+05	4.37E+05	9.78E+05	4.70E+05
Xe-131m	5.49E+03	2.86E+04	9.36E+03	6.66E+04
Xe-133m	1.51E+04	5.96E+04	1.96E+04	8.88E+04
Xe-133	1.76E+04	1.84E+04	2.12E+04	6.30E+04
Xe-135m	1.87E+05	4.27E+04	2.02E+05	4.43E+04
Xe-135	1.09E+05	1.12E+05	1.15E+05	1.48E+05
Xe-137	8.52E+04	7.32E+05	9.06E+04	7.62E+05
Xe-138	5.30E+05	2.48E+05	5.53E+05	2.85E+05
Ar-41	5.30E+05	1.61E+05	5.58E+05	1.97E+05

ATTACHMENT 12

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GASEOUS EFFLUENT DOSE FACTORS FOR SURRY POWER STATION

(Gamma and Beta Dose Factors)

$\chi/Q = 1.0E\text{-}06$ sec/m^3 at 644 meters S Direction

Dose Factors for Process Vent

Noble Gas Radionuclide	K_{ipv} Total Body $\frac{mrem/yr}{Curie/Sec}$	L_{ipv} Skin $\frac{mrem/yr}{Curie/Sec}$	M_{ipv} Gamma Air $\frac{mrad/yr}{Curie/Sec}$	N_{ipv} Beta Air $\frac{mrad/yr}{Curie/Sec}$
Kr-83m	7.56E-02	-	1.93E+01	2.88E+02
Kr-85m	1.17E+03	1.46E+03	1.23E+03	1.97E+03
Kr-85	1.61E+01	1.34E+03	1.72E+01	1.95E+03
Kr-87	5.92E+03	9.73E+03	6.17E+03	1.03E+04
Kr-88	1.47E+04	2.37E+03	1.52E+04	2.93E+03
Kr-89	1.66E+04	1.01E+04	1.73E+04	1.06E+04
Kr-90	1.56E+04	7.29E+03	1.63E+04	7.83E+03
Xe-131m	9.15E+01	4.76E+02	1.56E+02	1.11E+03
Xe-133m	2.51E+02	9.94E+02	3.27E+02	1.48E+03
Xe-133	2.94E+02	3.06E+02	3.53E+02	1.05E+03
Xe-135m	3.12E+03	7.11E+02	3.36E+03	7.39E+02
Xe-135	1.81E+03	1.86E+03	1.92E+03	2.46E+03
Xe-137	1.42E+03	1.22E+04	1.51E+03	1.27E+04
Xe-138	8.83E+03	4.13E+03	9.21E+03	4.75E+03
Ar-41	8.84E+03	2.69E+03	9.30E+03	3.28E+03

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GASEOUS EFFLUENT DOSE FACTORS FOR SURRY POWER STATION

(Inhalation Pathway Dose Factors)

Ventilation Vent $\chi/Q = 6.0E\text{-}05\ sec/m^3$ at 499 meters N Direction

Process Vent $\chi/Q = 1.0E\text{-}06\ sec/m^3$ at 644 meters S Direction

Radionuclide	P_{ivv} mrem/yr / Curie/sec	Pipv mrem/yr / Curie/sec
H-3	6.75E+04	1.12E+03
Cr-51	5.13E+03	8.55E+01
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	3.64E+05	6.07E+03
Te-129m	3.80E+05	6.33E+03
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND
I-131	9.75E+08	1.62E+07

ND - No data for dose factor according to Reg. Guide 1.109, Rev. 1.

ATTACHMENT 13

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GASEOUS EFFLUENT DOSE FACTORS FOR NORTH ANNA POWER STATION

(Gamma and Beta Dose Factors)

$\chi/Q = 9.3E\text{-}06\ sec/m^3$ at 1416 meters SE Direction

Dose Factors for Ventilation Vent

Noble Gas Radionuclide	K_{ivv} Total Body mrem/yr / Curie/Sec	L_{ivv} Skin mrem/yr / Curie/Sec	M_{ivv} Gamma Air mrad/yr / Curie/Sec	N_{ivv} Beta Air mrad/yr / Curie/Sec
Kr-83m	7.03E-01	-	1.79E+02	2.68E+03
Kr-85m	1.09E+04	1.36E+04	1.14E+04	1.83E+04
Kr-85	1.50E+02	1.25E+04	1.60E+02	1.81E+04
Kr-87	5.51E+04	9.05E+04	5.74E+04	9.58E+04
Kr-88	1.37E+05	2.20E+04	1.41E+05	2.72E+04
Kr-89	1.54E+05	9.39E+04	1.61E+05	9.86E+04
Kr-90	1.45E+05	6.78E+04	1.52E+05	7.28E+04
Xe-131m	8.51E+02	4.43E+03	1.45E+03	1.03E+04
Xe-133m	2.33E+03	9.24E+03	3.04E+03	1.38E+04
Xe-133	2.73E+03	2.85E+03	3.28E+03	9.77E+03
Xe-135m	2.90E+04	6.61E+03	3.12E+04	6.87E+03
Xe-135	1.68E+04	1.73E+04	1.79E+04	2.29E+04
Xe-137	1.32E+04	1.13E+05	1.40E+04	1.18E+05
Xe-138	8.21E+04	3.84E+04	8.57E+04	4.42E+04
Ar-41	8.22E+04	2.50E+04	8.65E+04	3.05E+04

ATTACHMENT 13

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GASEOUS EFFLUENT DOSE FACTORS FOR NORTH ANNA POWER STATION

(Gamma and Beta Dose Factors)

$\chi/Q = 1.2E\text{-}06\ sec/m^3$ at 1513 meters S Direction

Dose Factors for Process Vent

Noble Gas Radionuclide	K_{ipv} Total Body mrem/yr / Curie/Sec	L_{ipv} Skin mrem/yr / Curie/Sec	M_{ipv} Gamma Air mrad/yr / Curie/Sec	N_{ipv} Beta Air mrad/yr / Curie/Sec
Kr-83m	9.07E-02	-	2.32E+01	3.46E+02
Kr-85m	1.40E+03	1.75E+03	1.48E+03	2.36E+03
Kr-85	1.93E+01	1.61E+03	2.06E+01	2.34E+03
Kr-87	7.10E+03	1.17E+04	7.40E+03	1.24E+04
Kr-88	1.76E+04	2.84E+03	1.82E+04	3.52E+03
Kr-89	1.99E+04	1.21E+04	2.08E+04	1.27E+04
Kr-90	1.87E+04	8.75E+03	1.96E+04	9.40E+03
Xe-131m	1.10E+02	5.71E+02	1.87E+02	1.33E+03
Xe-133m	3.01E+02	1.19E+03	3.92E+02	1.78E+03
Xe-133	3.53E+02	3.67E+02	4.24E+02	1.26E+03
Xe-135m	3.74E+03	8.53E+02	4.03E+03	8.87E+02
Xe-135	2.17E+03	2.23E+03	2.30E+03	2.95E+03
Xe-137	1.70E+03	1.46E+04	1.81E+03	1.52E+04
Xe-138	1.06E+04	4.96E+03	1.11E+04	5.70E+03
Ar-41	1.06E+04	3.23E+03	1.12E+04	3.94E+03

ATTACHMENT 13

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GASEOUS EFFLUENT DOSE FACTORS FOR NORTH ANNA POWER STATION

(Inhalation Pathway Dose Factors)

Ventilation Vent $\chi/Q = 9.3E\text{-}06\ sec/m^3$ at 1416 meters SE Direction

Process Vent $\chi/Q = 1.2E\text{-}06\ sec/m^3$ at 1513 meters S Direction

Radionuclide	P_{ivv} $\frac{mrem/yr}{Curie/sec}$	Pipv $\frac{mrem/yr}{Curie/sec}$
H-3	1.05E+04	1.35E+03
Cr-51	7.95E+02	1.02E+02
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	5.64E+04	7.28E+03
Te-129m	5.88E+04	7.59E+03
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND
I-131	1.51E+08	1.95E+07

ND - No data for dose factor according to Reg. Guide 1.109, Rev. 1.

ATTACHMENT 14

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SURRY 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. 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
3. 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

ATTACHMENT 14

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SURRY RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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 continuously collected within one hour with auxiliary sampling equipment as required in Attachment 8.

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.

ATTACHMENT 15

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NORTH ANNA 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	2, 4
(b) Iodine Sampler	1	2, 5
(c) Particulate Sampler	1	2, 5
(d) Process Vent Flow Rate Measuring Device	1	1
(e) Sampler Flow Rate Measuring Device	1	1
2. CONDENSER AIR EJECTOR SYSTEM		
(a) Gross Activity Monitor	1	3
(b) Flow Rate Monitor	1	1
3. VENTILATION VENT SYSTEM (Shared with Unit 2)		
(a) Noble Gas Activity Monitor	1 (Note 1)	2
(b) Iodine Sampler	1 (Note 1)	2
(c) Particulate Sampler	1 (Note 1)	2
(d) Flow Rate Monitor	1 (Note 1)	1
(e) Sampler Flow Rate Monitor	1 (Note 1)	1

Note 1: One per vent stack

ATTACHMENT 15

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NORTH ANNA RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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 the flow rate is estimated at least once per 4 hours.

ACTION 2: With the number of channels OEPRABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

ACTION 3 With the number of channels OPERABLE less than required by the minimum channels OEPRABLE requirement, effluent releases via this pathway may continue provided the frequency of the grab samples required by Technical Specification requirement 4.4.6.3.b is increased to at least once per 4 hours and these samples are analyzed for gross activity or gamma isotopic activity within 8 hours.

ACTION 4: With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the Waste Gas Decay Tanks may be released to the environment provided that prior to initiate the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Station Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of Waste Gas Decay Tank effluents.

ACTION 5: With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases from the Waste Gas Decay Tanks may continue provided samples are continuously collected with auxiliary sampling equipment as required in Attachment 11.

ATTACHMENT 16

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SURRY RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. PROCESS VENT SYSTEM				
(a) Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	M, *	R	Q
(b) Iodine Sampler	W	N.A.	N.A.	N.A.
(c) Particulate Sampler	W	N.A.	N.A.	N.A.
(d) Process Vent Flow Rate Monitor	D	N.A.	R	N.A.
(e) Sampler Flow Rate Measuring Device	D	N.A.	SA	N.A.
2. 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.
3. 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.	SA	N.A.

* Prior to each Waste Gas Decay Tank release

ATTACHMENT 17

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NORTH ANNA RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. PROCESS VENT SYSTEM				
(a) Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	P	R	Q (Note 1)
(b) Iodine Sampler	W	N.A.	N.A.	N.A.
(c) Particulate Sampler	W	N.A.	N.A.	N.A.
(d) Process Vent Flow Rate Measuring Device	D	N.A.	R	Q
(e) Sampler Flow Rate Monitor	D (Note 3)	N.A.	R	N.A.
2. CONDENSER AIR EJECTOR SYSTEM				
(a) Noble Gas Activity Monitor	D	M	R	Q (Note 2)
(b) Flow Rate Monitor	D	N.A.	R	Q
3. VENTILATION VENT SYSTEM (Shared with Unit 2)				
(a) Noble Gas Activity Monitor	D	M	R	Q (Note 2)
(b) Iodine Sampler	W	N.A.	N.A.	N.A.
(c) Particulate Sampler	W	N.A.	N.A.	N.A.
(d) Flow Rate Monitor	D	N.A.	R	Q
(e) Sampler Flow Rate Monitor	D Note (3)	N.A.	R	N.A.

ATTACHMENT 17

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NORTH ANNA RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTE 1: The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm/trip setpoint.

b. Instrument controls not set in operate mode.

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

a. Instrument indicates measured levels above the alarm setpoint.

b. Instrument controls not set in operate mode.

NOTE 3: CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

ATTACHMENT 18

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CRITICAL ORGAN AND INHALATION DOSE FACTORS FOR SURRY

(Critical Pathway Dose Factors)

Ventilation Vent D/Q = 9.0E-10 m^{-2} at 5150 meters S Direction

Process Vent D/Q = 4.3E-10 m^{-2} at 5150 meters S Direction

Radionuclide	RM_{ivv} $\frac{mrem/yr}{Curie/sec}$	RM_{ipv} $\frac{mrem/yr}{Curie/sec}$
H-3	7.20E+02	3.12E+02
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-89	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	8.06E+04	3.85E+04
Te-129m	1.25E+05	5.98E+04
I-131	6.21E+08	2.97E+08
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND

ND - No data for dose factor according to Reg. Guide 1.109, Rev. 1.

ATTACHMENT 18

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CRITICAL ORGAN AND INHALATION DOSE FACTORS FOR SURRY

(Inhalation Pathway Dose Factors)

Ventilation Vent χ/Q = 3.0E-07 sec/m^3 at 5150 meters S Direction

Process Vent χ/Q = 1.3E-07 sec/m^3 at 5150 meters S Direction

Radionuclide	RI_{ivv} mrem/yr / Curie/sec	RIipv mrem/yr / Curie/sec
H-3	1.94E+02	8.41E+01
Cr-51	1.73E+01	7.48E+00
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-89	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	1.46E+03	6.33E+02
Te-129m	1.64E+03	7.12E+02
I-131	4.45E+06	1.93E+06
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND

ND - No data for dose factor according to Reg. Guide 1.109, Rev. 1.

ATTACHMENT 19

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CRITICAL ORGAN AND INHALATION DOSE FACTORS FOR NORTH ANNA

(Critical Pathway Dose Factors)

Ventilation Vent D/Q = 2.4E-09 m^{-2} at 3250 meters N Direction

Process Vent D/Q = 1.1E-09 m^{-2} at 3250 meters N Direction

Radionuclide	R_{ivv} $\frac{mrem/yr}{Curie/sec}$	Ripv $\frac{mrem/yr}{Curie/sec}$
H-3	1.73E+03	9.36E+02
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-89	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	1.97E+05	9.04E+04
Te-129m	2.95E+05	1.35E+05
I-131	1.45E+09	6.72E+08
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND

ND - No data for dose factor according to Reg. Guide 1.109, Rev. 1.

ATTACHMENT 20

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SURRY'S RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample and Sample Location	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
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 ground level 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.	<u>Radioiodine Cannister</u> I-131 Analysis Weekly <u>Particulate Sampler</u> Gross beta radioactivity analysis following filter change; Gamma isotopic analysis of composite (by location) quarterly

ATTACHMENT 20

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SURRY'S RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample and Sample Location	Collection Frequency	Type and Frequency of Analysis
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
4. INGESTION			
a) Milk	a) 4 samples from milking animals in the vicinity of Station. b) 1 sample from milking animals at a control location (15-30 km distant)	Monthly	Gamma isotopic and I-131 analysis monthly
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 b) 1 sample soybean c) 1 sample peanuts	Annually	Gamma isotopic on edible portion

ATTACHMENT 21

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NORTH ANNA'S RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM(Note 1)

Exposure Pathway and/or Sample	Number of Sample and Sample Location(Note 2)	Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION (Note 3)	36 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously to be placed as follows: 1) An inner ring of stations, one in each meteorological sector within the site boundary. 2) An outer ring of stations, one in each meteorological sector within 8 km range from the site 3) The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	GAMMA DOSE Quarterly

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NORTH ANNA'S RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample and Sample Location (Note 2)	Collection Frequency	Type and Frequency of Analysis
2. AIRBORNE			
Radioiodines and Particulates	Samples from 5 locations: a) 3 samples from close to the 3 site boundary locations (in different sectors) of the highest calculated historical annual average ground level D/Q. b) 1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q. c) 1 sample from a control location 15-40 km distant and in the least prevalent wind direction	Continuous sampler (2/3 running time cycle), operation with sample collection weekly	Radioiodine Cannister I-131 analysis, weekly Particulate Sampler Gross beta radioactivity analysis following filter change; (Note 4) Gamma isotopic analysis of composite (by location) quarterly (Note 5)
3. WATERBORNE			
a) Surface	1 sample circulating water discharge	Sample off upstream, downstream and cooling lagoon. Grab Monthly	Gamma isotopic analysis monthly; (Note 5) Composite for tritium analysis quarterly.
b) Ground	Sample from 1 or 2 sources only if likely to be affected.	Grab Quarterly	Gamma isotopic and tritium analysis quarterly (Note 5)
c) Sediment	1 sample from downstream area with existing or potential recreational value	Semi-Annually	Gamma isotopic analysis semi-annually (Note 5)

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NORTH ANNA'S RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample and Sample Location (Note 2)	Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a) Milk (Note 7)	a) Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr. (Note 6) b) 1 sample from milking animals at a control location (15-30 km distant) and in the least prevalent wind direction).	Monthly at all times.	Gamma isotopic (Note 5) and I-131 analysis monthly.
b. Fish and Invertebrates	a) 1 sample of commercially and recreationally important species (bass, sunfish, catfish) in vicinity of plant discharge area. b) 1 sample of same species in areas not influenced by plant discharge	Semiannually	Gamma isotopic on edible portions.
c) Food Products	a) Samples of an edible broad leaf vegetation grown nearest each of two different offsite locations of highest predicted historical annual average ground level D/Q if milk sampling is not performed. b) 1 sample of broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed	Monthly if available, or at harvest	Gamma isotopic (Note 5) and I-131 analysis.

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NORTH ANNA'S RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Note 1: The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and be included in the sampling program.

Note 2: For each and every sample location in Attachment 21, specific parameters of distance and direction sector from the centerline of the reactor, and additional description where pertinent, shall be provided in Attachment 23. Refer to Radiological Assessment Branch Technical Positions and to NUREG-0133, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plant. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to subsection 6.6.1. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a licensee Event Report and pursuant to subsection 6.6.2, identify the cause of the unavailability of samples for that pathway and identify the new locations for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report revised figures and tables for the ODCM reflecting the new locations.

Note 3: One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations, e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

Note 4: Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

Note 5: Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

Note 6: The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

Note 7: If milk sampling cannot be performed, use item 4.c (Pg. 3 of 4, Attachment 21)

ATTACHMENT 22

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SURRY'S ENVIRONMENTAL SAMPLING LOCATIONS

SAMPLE MEDIA	LOCATION	DISTANCE (MILES)	DIRECTION	REMARKS
Air Charcoal and Particulate	Surry Station (SS)	0.37	NNE	Site Boundary Location at Sector with Highest D/Q
	Hog Island Reserve (HIR)	2.0	NNE	
	Bacons Castle (BC)	4.5	SSW	
	Alliance (ALL)	5.1	WSW	
	Colonial Parkway (CP)	3.7	NNW	
	Dow Chemical (DOW)	5.1	ENE	
	Fort Eustis (FE)	4.8	ESE	
	Newport News (NN)	16.5	ESE	Control Location
Environmental TLDs	Control (00)			Onsite **
	West North West (02)	0.17	WNW	Site Boundary
	Surry Station Discharge (03)	0.6	NW	Site Boundary
	North North West (04)	0.4	NNW	Site Boundary
	North (05)	0.33	N	Site Boundary
	North North East (06)	0.28	NNE	Site Boundary
	North East (07)	0.31	NE	Site Boundary
	East North East (08)	0.43	ENE	Site Boundary
	East (Exclusion) (09)	0.31	E	Onsite
	West (10)	0.40	W	Site Boundary
	West South West (11)	0.45	WSW	Site Boundary
	South West (12)	0.30	SW	Site Boundary
	South South West (13)	0.43	SSW	Site Boundary
	South (14)	0.48	S	Site Boundary
	South South East (15)	0.74	SSE	Site Boundary
	South East (16)	1.00	SE	Site Boundary
	East (17)	0.57	E	Site Boundary
	Station Intake (18)	1.23	ESE	Site Boundary
	Hog Island Reserve (19)	1.94	NNE	Near Resident

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SURRY'S ENVIRONMENTAL SAMPLING LOCATIONS

SAMPLE MEDIA	LOCATION		DISTANCE (MILES)	DIRECTION	REMARKS
Environmental	Bacons Castle	(20)	4.45	SSW	Approx. 5 miles
TLDs	Route 633	(21)	3.5	SW	Approx. 5 miles
	Alliance	(22)	5.1	WSW	Approx. 5 miles
	Surry	(23)	8.0	WSW	Population Center
	Route 636 and 637	(24)	4.0	W	Approx. 5 miles
	Scotland Wharf	(25)	5.0	WNW	Approx. 5 miles
	Jamestown	(26)	6.3	NW	Approx. 5 miles
	Colonial Parkway	(27)	3.7	NNW	Approx. 5 miles
	Route 617 and 618	(28)	5.2	NNW	Approx. 5 miles
	Kingsmill	(29)	4.8	N	Approx. 5 miles
	Williamsburg	(30)	7.8	N	Population Center
	Kingsmill North	(31)	5.6	NNE	Approx. 5 miles
	Budweiser	(32)	5.7	NNE	Population Center
	Water Plant	(33)	4.8	NE	Approx. 5 miles
	Dow	(34)	5.1	ENE	Approx. 5 miles
	Lee Hall	(35)	7.1	ENE	Population Center
	Goose Island	(36)	5.0	E	Approx. 5 miles
	Fort Eustis	(37)	4.8	ESE	Approx. 5 miles
	Newport News	(38)	16.5	ESE	Population Center
	James River Bridge	(39)	14.8	SSE	Control
	Benn's Church	(40)	14.5	S	Control
	Smithfield	(41)	11.5	S	Control
	Rushmere	(42)	5.2	SSE	Approx. 5 miles
	Route 628	(43)	5.0	S	Approx. 5 miles
Milk	Lee Hall		7.1	ENE	
	Epp's		4.8	SSW	
	Colonial Parkway		3.7	NNW	
	Judkin's		6.2	SSW	
	William's		22.5	S	Control Location

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SURRY'S ENVIRONMENTAL SAMPLING LOCATIONS

SAMPLE MEDIA	LOCATION	DISTANCE (MILES)	DIRECTION	REMARKS
Well Water	Surry Station			Onsite***
	Hog Island Reserve	2.0	NNE	
	Bacons Castle	4.5	SSW	
	Jamestown	6.3	NW	
Crops (Corn, Peanuts, Soybeans)	Slade's Farm	2.4	S	State Split
	Brock's Farm	3.8	S	State Split
Crops (Cabbage, Kale)	Poole's Garden	2.3	S	State Split
	Carter's Grove Garden	4.8	NE	State Split
	Ryan's Garden			Control Location (Chester, Va.)
River Water (Bi-monthly)	Surry Station Intake	1.9	ESE	
	Hog Island Point	2.4	NE	
	Newport News	12.0	SE	
	Chicahominy River	11.2	WNW	Control Location
	Surry Station Discharge	0.17	NW	
River Water (Monthly)	Surry Discharge	0.17	NW	
	Scotland Wharf	5.0	WNW	Control Location
Sediment (Silt)	Chicahominy River	11.2	WNW	Control Location
	Surry Station Intake	1.9	ESE	
	Surry Station Discharge	1.0	NNW	
	Hog Island Point	2.4	NE	
	Point of Shoals	6.4	SSE	
	Newport News	12.0	SE	

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SURRY'S ENVIRONMENTAL SAMPLING LOCATIONS

SAMPLE MEDIA	LOCATION	DISTANCE (MILES)	DIRECTION	REMARKS
Clams	Chicahominy River	11.2	WNW	Control Location
	Surry Station Discharge	1.3	NNW	
	Hog Island Point	2.4	NE	
	Jamestown	5.1	WNW	
	Lawne's Creek	2.4	SE	
Oysters	Deep Water Shoals	3.9	ESE	
	Point of Shoals	6.4	SSE	
	Newport News	12.0	SE	
Crabs	Surry Station Discharge	0.6	NW	
Fish	Surry Station Discharge	0.6	NW	
Shoreline Sediment	Hog Island Reserve	0.8	N	
	Burwell's Bay	7.76	SSE	

** Onsite Location - in Lead Shield

*** Onsite sample of Well Water - taken from tap-water at Surry Environmental Building.

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NORTH ANNA'S ENVIRONMENTAL SAMPLING LOCATIONS

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Environmental TLDs	NAPS Sewage Treatment Plant	01	0.20	NE	Quarterly & Annually	On-Site
	Frederick's Hall	02	5.30	SSW	Quarterly & Annually	
	Mineral,VA	03	7.10	WSW	Quarterly & Annually	
	Wares Crossroads	04	5.10	WSW	Quarterly & Annually	
	Route 752	05	4.20	NNE	Quarterly & Annually	
	Sturgeon's Creek Marina	05A	3.20	N	Quarterly & Annually	
	Levy, VA	06	4.70	ESE	Quarterly & Annually	
	Bumpass, VA	07	7.30	SSE	Quarterly & Annually	
	End of Route 685	21	1.00	WNW	Quarterly & Annually	Exclusion Boundary
	Route 700	22	1.00	WSW	Quarterly & Annually	Exclusion Boundary
	"Aspen Hills"	23	0.93	SSE	Quarterly & Annually	Exclusion Boundary
	Orange, VA	24	22.00	NW	Quarterly & Annually	Control
	Bearing Cooling Tower	N-1/33	0.06	N	Quarterly	On-Site
	Sturgeon's Creek Marina	N-2/34	3.20	N	Quarterly	
	Parking Lot "C"	NNE-3/35	0.25	NNE	Quarterly	On-Site
	Good Hope Church	NNE-4/36	4.96	NNE	Quarterly	
	Parking Lot "B"	NE-5/37	0.20	NE	Quarterly	On-Site
	Lake Anna Marina	NE-6/38	1.49	NE	Quarterly	
	Weather Tower Fence	ENE-7/39	0.36	ENE	Quarterly	On-Site
	Route 689	ENE-8/40	2.43	ENE	Quarterly	
	Near Training Facility	E-9/41	0.30	E	Quarterly	On-Site

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NORTH ANNA'S ENVIRONMENTAL SAMPLING LOCATIONS

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Environmental TLDs (cont.)	"Morning Glory Hill"	E-10/42	2.85	E	Quarterly	
	Island Dike	ESE-11/43	0.12	ESE	Quarterly	On-Site
	Route 622	ESE-12/44	4.70	ESE	Quarterly	
	Biology Lab	SE-13/45	0.75	SE	Quarterly	On-Site
	Route 701 (Dam Entrance)	SE-14/46	5.88	SE	Quarterly	
	"Aspen Hills"	SSE-15/47	0.93	SSE	Quarterly	Exclusion Boundary
	Elk Creek	SSE-15/47	0.93	SSE	Quarterly	
	Warehouse Compound Gate	S-17/49	0.22	S	Quarterly	On-Site
	Elk Creek Church	S-18/50	1.55	S	Quarterly	
	NAPS Access Road	SSW-19/51	0.36	SSW	Quarterly	On-Site
	Route 700	SW-22/54	4.36	SW	Quarterly	
	500 KV Tower	WSW-23/55	0.40	WSW	Quarterly	On-Site
	Route 700	WSW-24/56	1.00	WSW	Quarterly	Exclusion Boundary
	NAPS Radio Tower	W-25/27	0.31	W	Quarterly	On-Site
	Route 685	W-26/58	1.55	W	Quarterly	
	End of Route 685	WNW-27/59	1.00	WNW	Quarterly	Exclusion Boundary
	H. Purcell's Private Road	WNW-27/59	1.52	WNW	Quarterly	
	End of #1/#2 Intake	NW-29/61	0.15	NW	Quarterly	On-Site
	Lake Anna Campground	NW-30/62	2.54	NW	Quarterly	
	#1/#2 Intake	NNW-31/63	0.07	NNW	Quarterly	On-Site
	Route 208	NNW-32/64	3.43	NNW	Quarterly	
	Bumpass Post Office	C-1/2	7.30	SSE	Quarterly	Control
	Orange, VA	C-3/4	22.00	NW	Quarterly	Control
	Mineral, VA	C-5/6	7.10	WSW	Quarterly	Control
	Louisa, VA	C-7/8	11.54	WSW	Quarterly	Control

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NORTH ANNA'S ENVIRONMENTAL SAMPLING LOCATIONS

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Airborne Particulate and Radioiodine	NAPS Sewage Treatment Plant	01	0.20	NE	Weekly	On-Site
	Frederick's Hall	02	5.30	SSW	Weekly	
	Mineral, VA	03	7.10	WSW	Weekly	
	Wares Crossroads	04	5.10	WNW	Weekly	
	Route 752	05	4.20	NNE	Weekly	
	Sturgeon's Creek Marina	05A	3.20	N	Weekly	
	Levy, VA	06	4.70	ESE	Weekly	
	Bumpass, VA	07	7.30	SSE	Weekly	
	End of Route 685	21	1.00	WNW	Weekly	Exclusion Boundary
	Route 700	22	1.00	WSW	Weekly	Exclusion Boundary
	"Aspen Hills"	23	0.93	SSE	Weekly	Exclusion Boundary
	Orange, VA	24	22.00	NW	Weekly	Control
Surface Water	Waste Heat Treatment Facility (Second Cooling Lagoon)	08	1.10	SSE	Monthly	
	Lake Anna (upstream)	09	2.20	NW	Monthly	Control
River Water	North Anna River (downstream)	11	5.80	SE	Quarterly	
Ground Water (well water)	Biology Lab	01A	0.75	SE	Quarterly	
Aquatic Sediment	Waste Heat Treatment Facility (Second Cooling Lagoon)	08	1.10	SSE	Semi-Annually	
	Lake Anna (upstream)	09	2.20	NW	Semi-Annually	Control
	North Anna River (downstream)	11	5.80	SE	Semi-Annually	
Shoreline Soil	Lake Anna (upstream)	09	2.20	NW	Semi-Annually	
Soil	NAPS Sewage Treatment Plant	01	0.20	NE	Once per 3 yrs	On-Site
	Mineral, VA	03	7.10	WSW	Once per 3 yrs	
	Wares Crossroads	04	5.10	WNW	Once per 3 yrs	
	Route 752	05	4.20	NNE	Once per 3 yrs	

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NORTH ANNA'S ENVIRONMENTAL SAMPLING LOCATIONS

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Soil (cont.)	Levy, VA	06	4.70	ESE	Once per 3 yrs	
	Bumpass, VA	07	7.30	SSE	Once per 3 yrs	
	End of Route 685	21	1.00	WNW	Once per 3 yrs	Exclusion Boundary
	Route 700	22	1.00	WSW	Once per 3 yrs	Exclusion Boundary
	"Aspen Hills"	23	0.93	SSE	Once per 3 yrs	Exclusion Boundary
	Orange, VA	24	22.00	NW	Once per 3 yrs	Control
Milk	Holladay Dairy (R.C. Goodwin)	12	8.30	NW	Monthly	
	Terrell's Dairy (Frederick's Hall)	13	5.60	SSE	Monthly	
Fish	Waste Heat Treatment Facility (Second Cooling Lagoon)	08	1.10	SSE	Quarterly	
	Lake Anna (upstream)	09	2.20	NW	Quarterly	Control
Food Products (Broad Leaf vegetation)	Route 713	14	varies	NE	Monthly if available, or at harvest	
	Route 614	15	varies	SE		
	Route 629/522	16	varies	NW		Control
	Route 685	21	varies	WNW		
	"Aspen Hills" Area	23	varies	SSE		

ATTACHMENT 24

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DETECTION CAPABILITIES FOR SURRY STATION ENVIRONMENTAL SAMPLE ANALYSIS(Note 1)

LOWER LIMIT OF DETECTION (LLD)(Note 4)

Analysis (Note 2)	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg) (wet)	Milk (pCi/l)	Food Products (pCi/kg) (wet)	Sediment (pCi/kg) (wet)
Gross beta	4	0.01				
H-3	2,000					
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	(Note 3) 1	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 1: Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

Note 2: 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.

Note 3: LLD for the Ground (drinking) Water Samples. The LLD for the Surface (non-drinking Water Samples is 10 pCi/l.

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DETECTION CAPABILITIES FOR SURRY STATION ENVIRONMENTAL SAMPLE ANALYSIS(Note 1)

LOWER LIMIT OF DETECTION (LLD)(Note 4)

Note 4: Acceptable detection capabilities for radioactive materials in environmental samples are tabulated in terms of the lower limits of detection (LLDs). LLD is defined, for purposes of this requirement, as the smallest concentration of radioactive material in a sample 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):

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

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume).

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

E = the counting efficiency (as counts per disintegration).

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

2.22×10^6 = the number of disintegrations per minute (dpm) per microcurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

Δt = the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

Typical values of E, V, Y and Δt 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.

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DETECTION CAPABILITIES FOR NORTH ANNA STATION ENVIRONMENTAL SAMPLE ANALYSIS(Note 1)

LOWER LIMIT OF DETECTION (LLD)(Note 3)

Analysis (Note 2)	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg) (wet)	Milk (pCi/l)	Food Products (pCi/kg) (wet)	Sediment (pCi/kg) (wet)
Gross beta	4	0.01				
H-3	2,000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	(Note 3) 1	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

Note 1: This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

Note 2: This LLD value is for drinking water samples.

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DETECTION CAPABILITIES FOR NORTH ANNA STATION ENVIRONMENTAL SAMPLE ANALYSIS(Note 1)

LOWER LIMIT OF DETECTION (LLD)(Note 3)

Note 3: Acceptable detection capabilities for radioactive materials in environmental samples are tabulated in terms of the lower limits of detection (LLDs). LLD is defined, for purposes of this requirement, as the smallest concentration of radioactive material in a sample 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):

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

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume).

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

E = the counting efficiency (as counts per disintegration).

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

2.22×10^6 = the number of disintegrations per minute (dpm) per microcurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

Δt = the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

Typical values of E, V, Y and Δt 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.

ATTACHMENT 26

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REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES AT SURRY STATION

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	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	(Note 1) 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	

Note 1: Reporting Level for the Ground (drinking) Water Samples required by Attachment 20. The Reporting Level for the Surface (non-drinking) Water Samples required by Attachment 20 is 20 pCi/l.

ATTACHMENT 27

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REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES AT NORTH ANNA STATION

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	20,000(1)				
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	

Note 1: For drinking water samples.

ATTACHMENT 28

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

1.0 METEOROLOGICAL ANALYSIS

1.1 Purpose

The purpose of the meteorological analysis was to determine the annual average χ/Q and D/Q values at critical locations around the Station for ventilation vent (ground level) and process vent (mixed mode) releases. The annual average χ/Q and D/Q values were used in performing a dose pathway analysis to determine both the maximum exposed individual at SITE BOUNDARY and MEMBER OF THE PUBLIC. The χ/Q and D/Q values resulting in the maximum exposures were incorporated into the dose factors in Attachments 12 and 18.

1.2 Meteorological Data, Parameters, and Methodology

Onsite meteorological data for the period January 1, 1979, through December 31, 1981, was used in calculations. This data included wind speed, wind direction, and differential temperature for the purpose of determining joint frequency distributions for those releases characterized as ground level (i.e., ventilation vent), and those characterized as mixed mode (i.e., process vent). The portions of release characterized as ground level were based on $\Delta T_{158.9ft-28.2ft}$ and 28.2 foot wind data, and the portions characterized as mixed mode were based on $\Delta T_{158.9ft-28.2ft}$ and 158.9 ft wind data.

χ/Q's and D/Q's were calculated using the NRC computer code "XOQDOQ - Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations", September, 1977. The code is based upon a straight line airflow model implementing the assumptions outlined in Section C (excluding C1a and C1b) of Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light - Water - Cooled Reactors".

The open terrain adjustment factors were applied to the χ/Q values as recommended in Regulatory Guide 1.111. The site region is characterized flat terrain such that open terrain correction factors are considered appropriate. The ground level ventilation vent release calculations included a building wake correction based on a 1516 m^2 containment minimum cross-sectional area. The effective release height used in mixed mode release calculations was based on a process vent release height of 131 ft, and plume rise due to momentum for a vent diameter of 3 in. with plume exit velocity of 100 ft/sec.

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Ventilation vent, and vent releases other than from the process vent, are considered ground level as specified in Regulatory Guide 1.111 for release points less than the height of adjacent solid structures, terrain elevations were obtained from Surry Power Station Units 1 and 2 Virginia Electric and Power Company Updated Final Safety Analysis Report Table 11A-11.

χ/Q and D/Q values were calculated for the nearest SITE BOUNDARY, resident, milk cow, and vegetable garden by sector for process vent and ventilation vent releases. χ/Q values were also calculated for the nearest discharge canal bank for process and ventilation vent releases.

According to the definition for short term in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Stations", October, 1978, some gaseous releases may fit this category, primarily waste gas decay tank releases and containment purges. However, these releases are considered long term for dose calculations as past releases were both random in time of day and duration as evidenced by reviewing past release reports. Therefore, the use of annual average concentrations is appropriate according to NUREG-0133.

1.3 Results

The χ/Q value that resulted in the maximum total body, skin and inhalation exposure for ventilation vent releases was 6.0E-05 sec/m^3 at a SITE BOUNDARY location 499 meters N sector. For process vent releases, the SITE BOUNDARY χ/Q value was 1.0E-06 sec/m^3 at a location 644 meters S sector. The discharge canal bank χ/Q value that resulted in the maximum inhalation exposure for ventilation vent releases was 7.8E-05 sec/m^3 at a location 290 meters NW sector. The discharge canal bank χ/Q value for process vent was 1.6E-06 sec/m^3 at a location 290 meters NW sector.

Pathway analysis indicated that the maximum exposure from I-131, and from all radionuclides in particulate form with half-lives greater than 8 days was through the grass-cow-milk pathway. The D/Q value from ventilation vent releases resulting in the maximum exposure was 9.0E-10 per m^2 at a location 5150 meters S sector. For process vent releases, the D/Q value was 4.3E-10 per m^2 at a location 5150 meters S sector. For tritium, the χ/Q value from ventilation vent releases resulting in the maximum exposure for the milk pathway was 3.0E-07 sec/m^3, and 1.3E-07 sec/m^3 for process vent releases at a location 5150 meters S sector. The inhalation pathway is the only other pathway existing at this location. Therefore, the χ/Q values given for tritium also apply for the inhalation pathway.

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

2.0 LIQUID PATHWAY ANALYSIS

2.1 Purpose

The purpose of the liquid pathway analysis was to determine the maximum exposed MEMBER OF THE PUBLIC in UNRESTRICTED AREAS as a result of radioactive liquid effluent releases. The analysis includes a determination of most restrictive liquid pathway, most restrictive age group, and critical organ. This analysis is required for subsection 6.2, Liquid Radioactive Waste Effluents.

2.2 Data, Parameters, and Methodology

Radioactive liquid effluent release data for the years 1976, 1977, 1978, 1979, 1980, and 1981 was compiled from the Surry Power Station effluent release reports. The data for each year, along with appropriate site specific parameters and default selected parameters, was entered into the NRC computer code LADTAP as described in NUREG-0133.

Liquid radioactive effluents from both units are released to the James River via the discharge canal. Possible pathways of exposure for release from the Station include ingestion of fish and invertebrates and shoreline activities. The irrigated food pathway and potable water pathway do not exist at this location. Access to the discharge canal by the general public is gained two ways: access for bank fishing is controlled by the Station and is limited to Virginia Power employees or guests of employees, and boating access is open to the public as far upstream as the inshore end of the discharge canal groin. It has been estimated that boat sport fishing would be performed a maximum of 800 hours per year, and that bank fishing would be performed a maximum of 160 hours per year.

For an individual fishing in the discharge canal, no river dilution was assumed for the fish pathway. For an individual located beyond the discharge canal groins, a river dilution factor of 5 was assumed as appropriate according to Regulatory Guide 1.109, Rev. 1, and the fish, invertebrate, and shoreline pathways were considered to exist. Dose factors, bioaccumulation factors, and shore width factors given in Regulatory Guide 1.109, Rev. 1, and in LADTAP were used, as were usage terms for shoreline activities and ingestion of fish and invertebrates. Dose to an individual fishing on the discharge bank was determined by multiplying the annual dose calculated with LADTAP by the fractional year the individual spent fishing in the canal.

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

2.3 RESULTS

For the years 1976, 1977, 1979, 1980, and 1981, the invertebrate pathway resulted in the largest dose. In 1978 the fish pathway resulted in the largest dose. The maximum exposed MEMBER OF THE PUBLIC was determined to utilize the James River. The critical age group was the adult and the critical organ was either the thyroid or GI-LLI. The ingestion dose factor, A_i, in subsection 6.2.3, Liquid Effluent Dose Limit, includes the fish and invertebrate pathways. A_i dose factors were calculated for the total body, thyroid, and GI-LLI organs.

3.0 GASEOUS PATHWAY ANALYSIS

3.1 Purpose

A gaseous effluent pathway analysis was performed to determine the location that would result in the maximum doses due to noble gases for use in demonstrating compliance with subsections 6.3.1.a and 6.3.3.a. The analysis also included a determination of the location, pathway, and critical organ, of the maximum exposed MEMBER OF THE PUBLIC, as a result of the release of I-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for use in demonstrating compliance with subsection 6.3.4.a. In addition, the analysis includes the determination of the critical organ, maximum age group, and sector location of an exposed individual through the inhalation pathway from I-131, tritium, and particulates for use in demonstrating compliance with subsection 6.3.1.a.

3.2 Data, Parameters, and Methodology

Annual average χ/Q values were calculated, as described in subsection 1 of this attachment, for the nearest SITE BOUNDARY in each directional sector and at other critical locations accessible to the public inside SITE BOUNDARY. The largest χ/Q value was determined to be 6.0E-05 sec/m^3 at SITE BOUNDARY for ventilation vent releases at a location 499 meters N direction, and 1.0E-06 sec/m^3 at SITE BOUNDARY for process vent releases at a location 644 meters S direction. The maximum doses to total body and skin, and air doses for gamma and beta radiation due to noble gases would be at these SITE BOUNDARY locations. The doses from both release points are summed in calculations to calculate total maximum dose.

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Step 6.3.1.a.2 dose limits apply specifically to the inhalation pathway. therefore, the locations and χ/Q values determined for maximum noble gas doses can be used to determine the maximum dose form I-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for the inhalation pathway.

The NRC computer code GASPAR, "Evaluation of Atmospheric Releases", Revised 8/19/77, was run using 1976, 1977, 1978, 1979, 1980 and 1981 Surry Power Station gaseous effluent release report data. Doses from I-131, tritium, and particulates for the inhalation pathway were calculated using the 6.0E-05 sec/m^3 SITE BOUNDARY χ/Q. except for the source term data and the χ/Q value, computer code default parameters were used. Results for each year indicated that the critical age group was the child and the critical organ was the thyroid for the inhalation pathway. In 1979, the teen was the critical age group. However, the dose calculated for the teen was only slightly greater than for the child and the doses could be considered equivalent.

The gamma and beta dose factors K_{ivv}, L_{ivv}, M_{ivv}, and N_{ivv} in Attachment 12 were obtained by performing a units conversion of the appropriate dose factors from Table B-1, Regulatory Guide 1.109, Rev. 1, to mrem/yr per Ci/m^3 or mrad/yr per Ci/m^3, and multiplying by the ventilation vent SITE BOUNDARY χ/Q value of 6.0E-05 sec/m^3. The same approach was used in calculating the gamma and beta dose factors K_{ipv}, L_{ipv}, M_{ipv}, and N_{ipv} in Attachment 12 using the process vent SITE BOUNDARY χ/Q value of 1.0E-06 sec/m^3.

Inhalation pathway dose factors P_{ivv} and P_{ipv} in Attachment 12 were calculated using the following equation:

$$P_i = K' \text{ (BR) } DFA_i \text{ } (\chi/Q \text{ (mrem/yr per Curie/sec)}$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

BR = the breathing rate of the child age group, 3700 m3/yr, from Table E-5, Regulatory Guide 1.109, Rev.1

DFA_i = the thyroid organ inhalation dose factor for child age group for the ith radionuclide, in mrem/pCi, from Table E-9, Regulatory Guide 1.109, Rev. 1

χ/Q = the ventilation vent SITE BOUNDARY χ/Q, 6.0E-5 sec/m^3, or the process vent SITE BOUNDARY χ/Q, 1.0E-06 sec/m^3 as appropriate.

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Subsection 6.3.4.a, requires that the dose to the maximum exposed MEMBER OF THE PUBLIC from I-131, tritium, and from all radionuclides in particulate form with half-lives greater than 8 days be less than or equal to the specified limits. Dose calculations were performed for an exposed MEMBER OF THE PUBLIC within SITE BOUNDARY UNRESTRICTED AREAS, discharge canal bank, and to an exposed MEMBER OF THE PUBLIC beyond SITE BOUNDARY at real residences with the largest χ/Q values using the NRC computer code GASPAR. Doses to MEMBERS OF THE PUBLIC were also calculated for the vegetable garden, meat animal, and milk-cow pathways with the largest D/Q values using the NRC computer code GASPAR.

It was determined that the MEMBER OF THE PUBLIC within SITE BOUNDARY would be using the discharge canal bank for fishing a maximum of 160 hours per year. The maximum annual χ/Q at this location was determined to be 7.8E-05 sec/m^3 at 290 meters NW direction. After applying a correction for the fractional part of year an individual would be fishing at this location, the dose was calculated to be less than an individual would receive at SITE BOUNDARY.

The MEMBER OF THE PUBLIC receiving the largest dose beyond SITE BOUNDARY was determined to be located 5150 meters S sector. The critical pathway was the grass-cow-milk, the maximum age group was the infant, and the critical organ the thyroid. For each year 1976, 1977, 1978, 1979, 1980 and 1981 the dose to the infant from the grass-cow-mild pathway was greater than the dose to the MEMBER OF THE PUBLIC within SITE BOUNDARY, nearest residence, vegetable or meat pathways. Therefore, the maximum exposed MEMBER OF THE PUBLIC was determined to be the infant, exposed through the grass-cow-milk pathway, critical organ thyroid, at a location 5150 meters S sector. The only other pathway existing at this location for the infant is the inhalation.

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

The RM_{ivv} and RM_{ipv} dose factors, except for tritium, in Attachment 18 were calculated by multiplying the appropriate D/Q value with the following equation:

$$RM_i = K' \frac{Q_F\ (U_{ap})}{\lambda_i + \lambda_w} F_m\ (r)\ (DFL_i) \left[\frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s)\ e^{-\lambda_i t_h}}{Y_s}\right] e^{-\lambda_i t_f}$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

Q_F = cow's consumption rate, 50, in Kg/day (wet weight)

U_{ap} = infant milk consumption rate, 330, liters/yr

Y_p = agricultural productivity by unit area of pasture feed grass, 0.7 Kg/m^2

Y_s = agricultural productivity by unit area of stored feed, 2.0, in Kg/m^2

F_m = stable element transfer coefficients, from Table E-1, Regulatory Guide 1.109, Rev. 1

r = fraction of deposited activity retained on cow's feed grass, 1.0 for radioiodine, and 0.2 for particulates

DFL_i = thyroid ingestion dose factor for the ith radionuclide for the infant, in mrem/pCi, from Table E-14, Regulatory Guide 1.109, Rev.1

λ_i = decay constant for the ith radionuclide, in sec^{-1}

λ_w = decay constant for removal of activity of leaf and plant surfaces by weathering, 5.73E-07 sec^{-1} (corresponding to a 14 day half-life)

t_f = transport time from pasture to cow, to milk, to receptor, 1.73+05, in seconds

t_h = transport time from pasture, to harvest, to cow, to milk, to receptor, 7.78E+06, in seconds

f_p = fraction of year that cow is on pasture, 0.67 (dimensionless), 7.78E+06 in seconds

f_s = fraction of cow feed that is pasture grass while cow is on pasture, 1.0, dimensionless

Parameters used in the above equation were obtained from NUREG-0133 and Regulatory Guide 1.109, Rev.1.

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SURRY METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Since the concentration of tritium in milk is based on the airborne concentration rather than the deposition, the following equation is used:

$$R_{H\text{-}3} = K' \, K''' \, F_m \, Q_F \, U_{ap} \, (DFL_{H\text{-}3}) \, [0.75 \, (0.5/H)] \times \chi/Q$$

where:

K''' = a constant of unit conversion 1E+03 gm/kg

H = absolute humidity of the atmosphere, 8.0, gm/m^3

0.75 = the fraction of total feed that is water

0.5 = the ratio of the specific activity of the feed grass to the atmospheric water

χ/Q = the annual average concentration at a location 5150 meters S sector, 3.0E-07 sec/m^3 for ventilation vent releases, and 1.3E-07 sec/m^3 for the process vent releases

Other parameters have been previously defined.

The inhalation pathway dose factors RI_{ivv} and RI_{ipv} in Attachment 18 were calculated using the following equation:

$$RI_i = K' \, (BR) \, DFA_i \, (\chi/Q) \text{ (mrem/yr per Curie/sec)}$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

BR = breathing rate of the infant age group, 1400 m^3/yr, from Table E-5, Regulatory Guide 1.109, Rev.1

DFA_i = thyriod organ inhalation dose factor for infant age group for the ith radionuclide, in mrem/pCi, from Table E-10, Regulatory Guide 1.109, Rev.1

χ/Q = ventilation vent χ/Q, 3.0E-07 sec/m^3, or the process vent SITE BOUNDARY χ/Q, 1.3E-07 sec/m^3, at a location 5150 meters S sector.

The GASPAR computer runs using 1976, 1977, 1978, 1979, 1980 and 1981 Surry effluent release data were reviewed to determine the percent of total dose from the cow milk and inhalation pathways for I-133. I-133 contributed less than 1% of the total dose to an infant's thyroid except for the year 1977 when the percent I-133 was 1.77. The calculations indicate that I-133 is a negligible dose contributor and it's inclusion in a sampling and analysis program, and dose calculation is unnecessary.

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NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

1.0 METEOROLOGICAL ANALYSIS

1.1 Purpose

The purpose of the meteorological analysis was to determine the annual average χ/Q and D/Q values at critical locations around the Station for ventilation vent (ground level) and process vent (mixed mode) releases. The annual average χ/Q and D/Q values were used in performing a dose pathway analysis to determine both the maximum exposed individual at SITE BOUNDARY and MEMBER OF THE PUBLIC. The χ/Q and D/Q values resulting in the maximum exposures were incorporated into the dose factors in Attachments 13 and 19.

1.2 Meteorological Data, Parameters, and Methodology

Onsite meteorological data for the period January 1, 1981, through December 31, 1981, was used in calculations. This data included wind speed, wind direction, and differential temperature for the purpose of determining joint frequency distributions for those releases characterized as ground level (e.g., ventilation vent), and those characterized as mixed mode (i.e., process vent). The portions of release characterized as ground level were based on $\Delta T_{158.9ft\text{-}28.2ft}$ and 28.2 foot wind data, and the portions characterized as mixed mode were based on $\Delta T_{158.9ft\text{-}28.2ft}$ and 158.9 ft wind data.

χ/Q's and D/Q's were calculated using the NRC computer code "XOQDOQ - Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations", September, 1977. The code is based upon a straight line airflow model implementing the assumptions outlined in Section C (excluding C1a and C1b) of Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light - Water - Cooled Reactors".

The open terrain adjustment factors were applied to the χ/Q values as recommended in Regulatory Guide 1.111. The site region is characterized by gently rolling terrain such that open terrain correction factors are considered appropriate. The ground level ventilation vent release calculations included a building wake correction based on a 1516 m^2 containment minimum cross-sectional area.

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NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

The effective release height used in mixed mode release calculations was based on a process vent release height of 157.5 ft, and plume rise due to momentum for a vent diameter of 3 in. with plume exit velocity of 100 ft/sec. Ventilation vent, and vent releases other than from the process vent, are considered ground level as specified in Regulatory Guide 1.111 for release points less than the height of adjacent solid structures, terrain elevations were obtained from North Anna Power Station Units 1 and 2 Virginia Electric and Power Company Final Safety Analysis Report Table 11C.2-8.

χ/Q and D/Q values were calculated for the nearest SITE BOUNDARY, resident, milk cow, and vegetable garden by sector for process vent and ventilation vent releases at distances specified from North Anna Power Station Annual Environmental Survey Data for 1981. χ/Q values were also calculated for the nearest lake shoreline by sector for the process vent and ventilation vent releases.

According to the definition for short term in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Stations", October, 1978, some gaseous releases may fit this category, primarily waste gas decay tank releases and containment purges. However, these releases are considered long term for dose calculations as past releases were both random in time of day and duration as evidenced by reviewing past release reports. Therefore, the use of annual average concentrations is appropriate according to NUREG-0133.

The χ/Q and D/Q values calculated from 1981 meteorological data are comparable to the values presented in the North Anna Power Station UFSAR.

1.3 Results

The χ/Q value that resulted in the maximum total body, skin and inhalation exposure for ventilation vent releases was 9.3E-06 sec/m^3 at a SITE BOUNDARY location 1416 meters SE sector. For process vent releases, the SITE BOUNDARY χ/Q value was 1.2E-06 sec/m^3 at a location 1513 meters S sector. The shoreline χ/Q value that resulted in the maximum inhalation exposure for ventilation vent releases was 1.0E-04 sec/m^3 at a location 241 meters NNE sector. The shoreline χ/Q value for process vent was 3.7E-06 sec/m^3 at a location 241 meters NNE sector.

ATTACHMENT 29

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NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Pathway analysis indicated that the maximum exposure from I-131, and from all radionuclides in particulate form with half-lives greater than 8 days was through the grass-cow-milk pathway. The D/Q value from ventilation vent releases resulting in the maximum exposure was 2.4E-09 per m^2 at a location 3250 meters N sector. For process vent releases, the D/Q value was 1.1E-09 per m^2 at a location 3250 meters N sector. For tritium, the χ/Q value from ventilation vent releases resulting in the maximum exposure for the milk pathway was 7.2E-07 sec/m^3, and 3.9E-07 sec/m^3 for process vent releases at a location 3250 meters N sector.

2.0 LIQUID PATHWAY ANALYSIS

2.1 Purpose

The purpose of the liquid pathway analysis was to determine the maximum exposed MEMBER OF THE PUBLIC in UNRESTRICTED AREAS as a result of radioactive liquid effluent releases. The analysis includes a determination of most restrictive liquid pathway, most restrictive age group, and critical organ. This analysis is required for subsection 6.2, Liquid Radioactive Waste Effluents.

2.2 Data, Parameters, and Methodology

Radioactive liquid effluent release data for the years 1979, 1980, and 1981 was compiled from the North Anna Power Station semi-annual effluent release reports. The data for each year, along with appropriate site specific parameters and default selected parameters, was entered into the NRC computer code LADTAP as described in NUREG-0133.

Reconcentration of effluents using the small lake connected to larger water body model was selected with the appropriate parameters determined from Table 3.5.3.5, Design Data for Reservoir and Waste Heat Treatment Facility from Virginia Electric and Power Company, Applicant's Environmental Report Supplement, North Anna Power Station, Units 1 and 2, March 15, 1972. Dilution factors for aquatic foods, shoreline, and drinking water were set to one. Transit time calculations were based on average flow rates. All other parameters were defaults selected by the LADTAP computer code.

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NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

2.3 RESULTS

For each year, the fish pathway resulted in the largest dose. The critical organ each year was the liver, and the adult and teenage age groups received the same organ dose. However, since the adult total body dose was greater than the teen total body dose for each year, the adult was selected as the most restrictive age group. Dose factors in Attachment 7 are for the maximum exposed MEMBER OF THE PUBLIC, an adult, with the critical organ being the liver.

3.0 GASEOUS PATHWAY ANALYSIS

3.1 Purpose

A gaseous effluent pathway analysis was performed to determine the location that would result in the maximum doses due to noble gases for use in demonstrating compliance with subsections 6.3.1.a and 6.3.3.a. The analysis also included a determination of the critical pathway, location of maximum exposed MEMBER OF THE PUBLIC, and the critical organ for the maximum dose due to I-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for use in demonstrating compliance with requirements in step 6.3.1.a.1 and subsection 6.3.3.a. The Analysis also included a determination of the critical pathway, location of maximum exposed MEMBER OF THE PUBLIC, and the critical organ for the maximum dose due to I-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for use in demonstrating compliance with step 6.3.1.a.2 and subsection 6.3.4.a.

3.2 Data, Parameters, and Methodology

Annual average χ/Q values were calculated, as described in subsection 1 of this attachment, for the nearest SITE BOUNDARY in each directional sector and at other critical locations beyond the SITE BOUNDARY. The largest χ/Q value was determined to be 9.3E-06 sec/m^3 at SITE BOUNDARY for ventilation vent releases at a location 1416 meters SE direction, and 1.2E-06 sec/m^3 at SITE BOUNDARY for process vent releases at a location 1513 meters S direction. The maximum doses to total body and skin, and air doses for gamma and beta radiation due to noble gases would be at these SITE BOUNDARY locations. The doses from both release points are summed in calculations to calculate total maximum dose.

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NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Step 6.3.1.a.2 dose limits apply specifically to the inhalation pathway. therefore, the locations and χ/Q values determined for maximum noble gas doses can be used to determine the maximum dose form I-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for the inhalation pathway.

The NRC computer code GASPAR, "Evaluation of Atmospheric Releases", Revised 8/19/77, was run using 1979, 1980 and 1981 North Anna Power Station Gaseous Effluent Release Report data. Doses from I-131, tritium, and particulates for the inhalation pathway were calculated using the 9.3E-06 sec/m^3 SITE BOUNDARY χ/Q. Except for the source term data and the χ/Q value, computer code default parameters were used. Results for each year indicated that the critical age group was the child and the critical organ was the thyroid for the inhalation pathway.

The gamma and beta dose factors K_{ivv}, L_{ivv}, M_{ivv}, and N_{ivv} in Attachment 12 were obtained by performing a units conversion of the appropriate dose factors from Table B-1, Regulatory Guide 1.109, Rev. 1, to mrem/yr per Ci/m^3 or mrad/yr per Ci/m^3, and multiplying by the ventilation vent SITE BOUNDARY χ/Q value of 9.3E-06 sec/m^3. The same approach was used in calculating the gamma and beta dose factors K_{ipv}, L_{ipv}, M_{ipv}, and N_{ipv} in Attachment 13 using the process vent SITE BOUNDARY χ/Q value of 1.2E-06 sec/m^3.

The inhalation pathway dose factors P_{ivv} and P_{ipv} in Attachment 13 were calculated using the following equation:

$$P_i = K' \, (BR) \, DFA_i \, (\chi/Q) \text{ (mrem/yr per Curie/sec)}$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

BR = the breathing rate of the child age group, 3700 m^3/yr, from Table E-5, Regulatory Guide 1.109, Rev.1

DFA_i = the thyroid organ inhalation dose factor for child age group for the ith radionuclide, in mrem/pCi, from Table E-9, Regulatory Guide 1.109, Rev. 1

χ/Q = the ventilation vent SITE BOUNDARY χ/Q, 9.3E-06 sec/m^3, or the process vent SITE BOUNDARY χ/Q, 1.2E-06 sec/m^3 as appropriate.

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NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Subsection 6.3.4.a, requires that the dose to the maximum exposed MEMBER OF THE PUBLIC from I-131, tritium, and from all radionuclides in particulate form with half-lives greater than 8 days be less than or equal to the specified limits. Dose calculations were performed for an exposed MEMBER OF THE PUBLIC within SITE BOUNDARY UNRESTRICTED AREAS, and to an exposed MEMBER OF THE PUBLIC beyond SITE BOUNDARY at locations identified in the North Anna Power Station Annual Environmental Survey Data for 1981.

It was determined that the MEMBER OF THE PUBLIC within SITE BOUNDARY would be using Lake Anna for recreational purposes a maximum of 2232 hours per year. It is assumed that this MEMBER OF THE PUBLIC would be located the entire 2232 hours at the lake shoreline with the largest annual χ/Q of 1.0E-04 at a location 241 meters NNE sector. The NRC computer code GASPAR was run to calculate the inhalation dose to this individual. The GASPAR results were corrected for the fractional year the MEMBER OF THE PUBLIC would be using the lake.

Using the NRC computer code GASPAR and annual average χ/Q and D/Q values obtained as described in subsection 1 of this attachment the MEMBER OF THE PUBLIC receiving the largest dose beyond SITE BOUNDARY was determined to be located 3250 meters N sector. The critical pathway was the grass-cow-milk, the maximum age group was the infant, and the critical organ the thyroid.

For each year 1979, 1980 and 1981 the dose to the infant from the grass-cow-milk pathway was greater than the dose to the MEMBER OF THE PUBLIC within SITE BOUNDARY. Therefore, the maximum exposed MEMBER OF THE PUBLIC was determined to be the infant, exposed through the grass-cow-milk pathway, critical organ thyroid, at a location 3250 meters N sector.

ATTACHMENT 29

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NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

The R_{ivv} and R_{ipv} dose factors, except for tritium, in Attachment 19 were calculated by multiplying the appropriate D/Q value with the following equation:

$$R_i = K' \frac{Q_F\ (U_{ap})}{\lambda_i + \lambda_w} F_m\ (r)\ (DFL_i) \left[\frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s)\ e^{-\lambda_i t_h}}{Y_s}\right] e^{-\lambda_i t_f}$$

where:

- K' = a constant of unit conversion, 1E+12 pCi/Ci
- Q_F = cow's consumption rate, 50, in Kg/day (wet weight)
- U_{ap} = infant milk consumption rate, 330, liters/yr
- Y_p = agricultural productivity by unit area of pasture feed grass, 0.7 Kg/m^2
- Y_s = agricultural productivity by unit area of stored feed, 2.0, in Kg/m^2
- F_m = stable element transfer coefficients, from Table E-1, Regulatory Guide 1.109, Rev. 1
- r = fraction of deposited activity retained on cow's feed grass, 1.0 for radioiodine, and 0.2 for particulates
- DFL_i = thyroid ingestion dose factor for the ith radionuclide for the infant, in mrem/pCi, from Table E-14, Regulatory Guide 1.109, Rev.1
- λ_i = decay constant for the ith radionuclide, in sec^{-1}
- λ_w = decay constant for removal of activity of leaf and plant surfaces by weathering, 5.73E-07 sec^{-1} (corresponding to a 14 day half-life)
- t_f = transport time from pasture to cow, to milk, to receptor, 1.73E+05, in seconds
- t_h = transport time from pasture, to harvest, to cow, to milk, to receptor, 7.78E+06, in seconds
- f_p = fraction of year that cow is on pasture, 0.58 (dimensionless), 7 months per year from NUREG-0597
- f_s = fraction of cow feed that is pasture grass while cow is on pasture, 1.0, dimensionless

Parameters used in the above equation were obtained from NUREG-0133 and Regulatory Guide 1.109, Rev.1.

ATTACHMENT 29

(Page 8 of 8)

NORTH ANNA METEOROLOGICAL, LIQUID AND GASEOUS PATHWAY ANALYSIS

Since the concentration of tritium in milk is based on the airborne concentration rather than the deposition, the following equation is used:

$$R_{H-3} = K' \; K''' \; F_m \; Q_F \; U_{ap} \; (DFL_{H-3}) \; [0.75 \, (0.5/H)] \; x \; \chi/Q$$

where:

K''' = a constant of unit conversion 1E+03 gm/kg

H = absolute humidity of the atmosphere, 8.0, gm/m^3

0.75 = the fraction of total feed that is water

0.5 = the ratio of the specific activity of the feed grass to the atmospheric water

χ/Q = the annual average concentration at a location 3250 meters N sector, 7.2E-07 sec/m^3 for ventilation vent releases, and 3.9E-07 sec/m^3 for the process vent releases

Other parameters have been previously defined.

Attachment 5

Process Control Program

Virginia Electric and Power Company

VIRGINIA POWER

Station Administrative Procedure

Title: Radioactive Waste Process Control Program (PCP)

Lead Department: Radiological Protection

Procedure Number:	Revision Number:	Effective Date:
VPAP-2104	0	05/31/90

Surry Power Station		North Anna Power Station	
Approved by:		Approved by:	
SNSOC Chairman	3.23.90 Date	SNSOC Chairman	3/22/90 Date
Approved by:		Approved by:	
Station Manager	4/3/90 Date	Station Manager	4/19/90 Date

Approved by: ASST Vice President - Nuclear Operations 4-27-90 Date

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TABLE OF CONTENTS

1.0 PURPOSE

This procedure establishes Virginia Power's PROCESS CONTROL PROGRAM (PCP) including associated requirements and responsibilities. The PCP provides instructions for processing and packaging of wet radioactive wastes to assure compliance with applicable Federal and State regulations for disposal of solid radioactive waste.

2.0 SCOPE

This procedure is applicable to the processing and packaging of wet radioactive waste performed at or by the Station. Systems and procedures used for implementing the PCP, including vendor supplied systems and procedures, shall be considered a part of the PCP.

3.0 REFERENCES/COMMITMENT DOCUMENTS

3.1 References

3.1.1 10 CFR 20, Standards for Protection Against Radiation

3.1.2 10 CFR 50, Domestic Licensing of Production and Utilization Facilities

3.1.3 10 CFR 61, Licensing Requirements for Land Disposal of Radioactive Waste

3.1.4 10 CFR 71, Packaging and Transportation of Radioactive Material

3.1.5 49 CFR Parts 170 to 189, Department of Transportation Regulations for Transportation of Hazardous Materials

3.1.6 USNRC Low-Level Waste Licensing, Branch Technical Position on Radioactive Waste Classification and Technical Position on Waste Form, May 1983, Rev 0

3.1.7 INPO 88-010, Guidelines for Radiological Protection at Nuclear Power Stations

3.1.8 NUREG-0800, USNRC, Standard Review Plan 11.4, Solid Waste Management Systems, Rev 2, July 1981

3.1.9 NRC Generic Letter 89-01, Implementation of Programmatic Controls for Radiological Effluent Technical Specifications (RETS) in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the PROCESS CONTROL PROGRAM

3.1.10 Surry and North Anna Technical Specifications

3.1.11 NODS-HP-01, Radiation Protection Plan

3.1.12 VPAP-0102, Station Nuclear Safety and Operating Committee

3.1.13 VPAP-2101, Radiation Protection Plan

3.1.14 VPAP-2103, Offsite Dose Calculation Manual (ODCM)

3.1.15 VPAP-3001, Safety Evaluations *(when issued)*

3.1.16 Chem-Nuclear Systems, Inc. Letter (concerning limitation of package void space), October 6, 1989, GAR-196-89, [4605g]

3.2 Commitment Documents

None

4.0 DEFINITIONS

NOTE: Terms which are defined in Surry and North Anna Technical Specifications appear as all capitalized letters in the text of this procedure for identification.

4.1 Batch

A quantity of waste that is or may be mixed to produce a homogeneous mixture for the purposes of sampling, testing, and processing. Different samples of a homogeneous mixture are expected to exhibit similar chemical and physical properties.

4.2 Composite

A mixture of samples, proportional by volume to the individual transfers making up a batch, that creates a test specimen representative of the batch.

4.3 Free Liquid

Free liquid is the liquid still visible after solidification or dewatering is complete, or is drainable from the low point of a punctured container (NRC SRP 11.4, ETSB 11-3).

4.4 High Integrity Container

A container designed to provide long-term structural stability to contained waste during the required disposal period. May be used as an alternative to waste solidification. See section C.4 of NRC BTP (Waste Form) for more details. High integrity containers must be approved by the appropriate agency.

SUPPLEMENTAL REFERENCE PAGE

This Supplemental Reference Page is provided to aid the procedure user in determining the appropriate procedures to use until such time that procedures referenced in the References Section, which reflect "When Issued", are approved and issued.

a. Upgraded Procedure Reference

VPAP-3001, Safety Evaluations *(When Issued)*

The following existing procedures shall be used with respect to Safety Evaluations until such time that the new referenced procedure is approved and issued:

a. **Surry**

1. SUADM-LR-12, Safety Analysis/10CFR50.59/10CFR72.48 Safety Evaluations and Justifications for Continued Operations

b. **North Anna**

1. ADM-3.9, 10CFR 50.59 Safety Evaluation and JCOs (North Anna)
2. ADM-3.15, Tracking of Justifications for Continued Operation (JCO)

NOTE: This Supplemental Reference Page shall be removed and processed as directed upon notification from Records Management.

4.5 Non-Corrosive Liquid

In lieu of specific tests, a liquid may be considered non-corrosive if it has a pH between 4 and 11 (based on section C.2.h of NRC BTP (Waste Form)).

4.6 Process Control Program

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

4.7 Site Boundary

The SITE BOUNDARY is that line beyond which the land is not owned, leased, or otherwise controlled by Virginia Power.

4.8 Solidification

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

4.9 Spent Ion Exchange Material

Organic resins and other ion exchange material are considered spent when decontamination factors decrease significantly or when activity levels reach a pre-determined level.

4.10 Stabilization or Stability

A structurally stable waste form will generally maintain its physical dimensions and its form under the expected disposal conditions. Structural stability can be provided by the waste form itself, processing the waste to a stable form (e.g, solidify), or placing the waste in a disposal container or structure that provides stability after disposal (10 CFR 61.56(b)).

4.11 Test Specimen

A sample obtained from a batch of waste to be processed (solidified or absorbed), or a simulated sample of similar chemical and physical characteristics, on which a test can be performed to verify the intended process will perform satisfactory.

4.12 Unrestricted Area

UNRESTRICTED AREA is defined as any area at or beyond the SITE BOUNDARY where access is not controlled by Virginia Power for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional or recreational purposes.

4.13 Wet versus Dry Wastes (from NRC SRP 11.4, BTP ETSB 11-3)

Radioactive waste is generated in the forms of "wet" and "dry" wastes. Wet wastes, including spent ion exchange material, filter sludge, evaporator concentrates, and spent cartridge filter elements, normally are byproducts from liquid processing systems. Dry wastes, including activated charcoal, HEPA filters, rags, paper, and clothing, normally are byproducts from ventilation air and gaseous waste processing systems, and maintenance and refueling operations.

5.0 RESPONSIBILITIES

5.1 Health Physics

Health Physics (HP) is responsible for:

5.1.1 Implementing the PROCESS CONTROL PROGRAM as a part of the Radiation Protection Program.

5.1.2 Ensuring that vendors brought on site by Health Physics to perform waste processing are cognizant of responsibilities in accordance with this procedure.

5.1.3 Maintaining procedures necessary for implementing the PCP.

5.2 Operations Department

The Operations Department is responsible for:

5.2.1 Implementing the PROCESS CONTROL PROGRAM as part of normal Station operations.

5.2.2 Ensuring that vendors brought on site by Operations to perform waste processing are cognizant of responsibilities in accordance with this procedure.

5.2.3 Maintaining procedures necessary for implementing the PCP.

6.0 INSTRUCTIONS

6.1 General Descriptions and Requirements

6.1.1 Types of Wet Radioactive Waste

Wet radioactive wastes produced at the Station which must be processed for disposal include:

- Resin
- Filter elements
- Waste oil
- Liquid waste

6.1.2 Waste Sources

a. Station systems which normally process radioactive liquids with the subsequent generation of spent radioactive ion exchange bead resin and/or filter elements which must be processed for disposal are:

- Primary Coolant System
- Boron Recovery System
- Spent Fuel Pit Purification System
- Vent and Drain System
- Liquid Waste Processing System

b. If primary to secondary leakage occurs while the Condensate Polishing System is processing secondary condensate, resin and filter elements used in the system may become radioactive. If so, they shall be processed for disposal.

c. If lubricating/cooling oil becomes contaminated with radioactive material, and if the oil is to be disposed of as radioactive waste in a licensed land disposal facility, the oil shall be considered and processed as wet radioactive waste.

d. If liquid wet waste is produced which must be disposed of (e.g., evaporator bottoms or decontamination solutions) it shall be treated as wet radioactive waste.

6.1.3 Requirements for Processing Wet Radioactive Waste

a. Liquids which are to be processed as radioactive waste shall be processed by solidification.

b. Resins shall be processed by dewatering and/or solidification.

c. Filter elements shall be processed by dewatering or encapsulation in a solidification binder.

d. Waste oil shall be processed by solidification or transferred to a licensed waste processor for disposal.

e. Class B and Class C waste shall be stabilized prior to disposal (10 CFR 61).

f. Certain categories of Class A waste shall be stabilized prior to disposal as required by the disposal site and/or the disposal site license conditions.

6.1.4 Process Control Program Implementing Procedures

a. Health Physics shall maintain procedures necessary to implement the PCP. Procedures shall include acceptable methods for:

1. Radioactive waste sampling, analysis and waste classification. Waste classification shall be performed per 10 CFR 61.55, Waste Classification, and methods set forth in NRC BTP on Radioactive Waste Classification.
2. Radioactive waste processing including waste solidification and stabilization. Acceptance criteria shall meet criteria set forth in:
   - 10 CFR 61.56, Waste Characteristics
   - NRC BTP on Waste Form
   - Disposal site criteria

3. Radioactive waste packaging and shipping. Acceptance criteria shall meet requirements set forth in:

- 10 CFR 20.311, Transfer for Disposal and Manifests
- 10 CFR 71, Packaging and Transporting of Radioactive Material
- 49 CFR 170 - 189, Transportation of Hazardous Materials

b. Operations Department shall maintain procedures necessary to implement the PCP. Procedures shall include acceptable methods for dewatering ion exchange resin.

6.1.5 Requirements For Use of Contractor Services

The following actions shall be taken before a contractor-supplied waste processing system is used on site:

a. Obtain the following, as a minimum, for review and evaluation:

- A detailed system description, which may be included in a topical report or equivalent documentation
- System operating procedures, which include process control parameters
- A list of required physical interfaces and Station materials/services
- A list of chemicals to be brought on site, quantity to be used and material safety data sheets for each chemical
- A list of expected utility/contractor responsibilities including disposal of unused and contaminated chemicals
- Vendor's document control procedures/manual to ensure controls are in place which prohibits use of procedures not approved by Station Nuclear Safety Operating Committee (SNSOC)

b. Compare the system description and operating procedures to the requirements provided in Subsection 6.2, Solidification of Wet Waste. Ensure that the system can be operated within requirements.

c. Submit system operating procedures to SNSOC for review and approval in accordance with VPAP-0102, Station Nuclear Safety and Operating Committee. Processing of radwaste shall not be performed without approved operating procedures.

d. Ensure the contractor provides a system as proposed, described, and approved for use at the Station.

6.2 Solidification of Wet Waste

Procedures used for wet waste solidification shall incorporate the following requirements:

6.2.1 Solidification Parameters

a. As appropriate, parameters used when performing solidification may include, but are not limited to:

- Waste type
- Waste pH
- Ratios of waste/liquid to solidification agent/catalyst
- Waste oil content
- Waste principal chemical constituents
- Mixing and curing times

b. Once established, solidification parameters shall provide boundary conditions to ensure that:

- Solidification is complete
- Requirements for waste form stability are met
- There are no detectable free standing liquids

6.2.2 Adverse Chemical Reactions During Solidification

Adverse chemical reactions between waste contaminants and solidification agents may not be noticeable during specimen tests performed to develop solidification parameters. To preclude such adverse chemical reactions, the following shall be performed prior to initial solidification of wet radioactive waste :

NOTE: Performance of this subsection is not required if solidification is to be performed by a vendor and results of such testing performed by the vendor was included in a technical report describing the proposed solidification methodology.

a. Prepare large volume (e.g., 1 or 2 gallons) non-radioactive mixtures of the waste stream chemicals potentially present (e.g., resin beads, boric acid, acids, bases, detergents, decontamination solutions).

b. Solidify the mixture.

1. The mixture shall be solidified using solidification procedure and parameters prepared for specified waste stream.
2. The solidification shall be performed within an insulated container to simulate the restricted heat removable capability of larger containers.

c. Ensure the mixture solidifies without generating excessive temperatures or gases.

6.2.3 Sampling, Analysis, and Process Surveillance

Wet radioactive waste shall be processed strictly in accordance with the approved solidification procedure for the specific waste stream substances to be solidified. Waste shall be sampled, analyzed, and compared to solidification parameters.

a. Results of sample analysis shall be recorded on waste processing data sheets.

b. A representative test specimen from at least every tenth batch of each type of waste to be solidified shall be used to verify solidification. If any test specimen fails to solidify:

1. Solidification of the batch under test shall be suspended until such time as:
    - Additional samples can be obtained
    - Alternative solidification parameters can be determined
    - Subsequent tests verify solidification
2. Solidification of the batch may then be resumed using the alternative solidification parameters determined.
3. A representative test specimen shall be obtained from each subsequent batch of the same type of waste and test solidification performed.
4. Collection and testing of representative test specimens from each consecutive batch shall continue until three consecutive initial test specimens demonstrate solidification.

c. If necessary, procedures shall be revised to ensure solidification of subsequent batches of waste.

d. If provisions of the PROCESS CONTROL PROGRAM cannot be satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

6.2.4 Solidification Acceptance Criteria

NOTE: The following are general considerations. Specific site disposal criteria must be addressed based on the site to be used.

Procedures for wet radioactive waste solidification shall incorporate the following requirements:

a. Containers for processed waste shall be filled to at least 85% of capacity. If a container is processed to less than 85% of capacity, it shall not be shipped for disposal prior to approval from the disposal site.

b. Solid waste that contains liquid shall have as little free standing liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume. The liquid shall be noncorrosive.

   1. If a high integrity container is not used, the maximum free liquid is 0.5% of the waste volume.
   2. If a high integrity container is used, the maximum free liquid is 1.0% of the waste volume.

6.3 Dewatering and Encapsulation of Filter Elements

NOTE: Filter elements are normally mechanical filters with wound fiber cartridges used for removing particulates from liquid systems. This procedure is only applicable to filter elements which are of the cartridge type.

6.3.1 General Requirements

a. Spent filter elements removed from systems shall be placed in appropriate storage to await processing and shipment.

b. Processing of spent filter elements shall be based on waste classification of filter.

NOTE: The following are general considerations. Specific site disposal criteria must be addressed based on the site to be used.

1. If filter media is classified as Class A waste and does not contain nuclides with half-lives greater than 5 years which have a total specific activity of 1 μCi/cc or greater, it may be disposed of as Class A waste.
2. If filter media is classified as Class B or Class C waste (per 10 CFR 61.55), it shall be encapsulated in a solidification media prior to disposal or disposed of in a high integrity container (NRC BTP, C.5 (Waste Form)).

6.3.2 Filter Elements to be Disposed of as Class A Waste

a. Filters should be allowed to drain dry in such a manner that any liquid trapped in voids is allowed to drain.

b. Filters shall not be compacted unless they are first allowed to dry essentially free of moisture.

c. If moist filters are to be packaged without compaction:

1. There shall be no indication of moisture on the filter in the form of drops or surface wetness.
2. Place filters in a container or plastic bag to which absorbent material has been placed to absorb unintentional and incidental amounts of liquids. The amount of absorbent material should be equal to at least one-fourth the volume of filter.

d. Ensure documentation indication package contents describes the presence of filters.

6.3.3 Filter Elements to be Disposed of as Class B or C Waste

a. If filters are to be solidified by being encapsulated in a solidification media:

1. Place filters in a suitable container such that filters will be completely surrounded by the solidification media when added. A basket type arrangement of thin wire is recommended to hold filters in a fixed geometry.

NOTE: The solidification media, including absence of free liquid, must be tested and documented in a manner required for solidification described in subsection 6.2, Solidification of Wet Waste.

2. Fill container with solidification media until filters are completely covered and container is filled to at least 85% of capacity.
3. Place solidified filter container in container appropriate for shipping and disposal at specified disposal site. A high integrity container is recommended to ensure compliance with all requirements.

b. If an encapsulated filter is to be disposed of in a high integrity container, properly place the container with the encapsulated filter in a high integrity container.

c. If an un-encapsulated filter is to be disposed of in a high integrity container:

1. Place filters in container such that filters will be held in a fixed geometry and such that liquids will not be trapped within filters. A basket type arrangement of thin wire is recommended to hold filters provided container's C of C will not be violated.
2. If resin will be added, proceed with resin addition as appropriate.
3. Dewater the container, as applicable.

6.4 Reporting Requirements

6.4.1 Major Changes to Radioactive Solid Waste Treatment Systems

NOTE: Information required by this subsection to be reported to the NRC may be submitted as part of the annual FSAR update.

Major changes to the radioactive solid waste systems:

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

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

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59. Such evaluations shall be made in accordance with VPAP-3001, Safety Evaluations.
2. Detailed information sufficient to totally support the reason for the change without benefit of additional or supplemental information.
3. A detailed description of equipment, components, and processes involved and interfaces with other plant systems.
4. An evaluation of the change, in quantity of solid waste differing from that previously predicted in the license application and amendments to the application.
5. An evaluation of the change, which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license and amendments.
6. A comparison of the predicted releases of radioactive materials, in solid waste, to the actual releases for the period prior to the changes.
7. An estimate of the exposure to plant operating personnel as a result of the change.
8. Documentation of SNSOC review and approval.

6.4.2 Changes to the Process Control Program (PCP)

Changes to the PCP shall be:

a. Documented; reviews shall be retained as Station records. Documentation shall include:

   1. Information to support the change together with the appropriate analyses or evaluations justifying the changes.
   2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Reviewed and approved by SNSOC and Plant Manager prior to implementation.

7.0 RECORDS

The following individual/packaged documents and related correspondence completed as a result of the performance or implementation of this procedure are records. Records shall be transmitted to Records Management in accordance with VPAP-1701, Records Management.

PROCESS CONTROL PROGRAM records shall include, but are not limited to:

- System description of any contractor's temporary processing system. Such a description may be provided in a topical report or other equivalent documentation
- Approved solidification system operating procedures
- Data sheets used to record solidification data, including test specimen data
- Records of reviews performed for changes made to the PROCESS CONTROL PROGRAM

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

Serial # 89-093

January 31, 1989

Rec'd. FEB 0 9 1989

Nuclear Operations
Licensing Supervisor

TO ALL POWER REACTOR LICENSEES AND APPLICANTS

SUBJECT: IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS IN THE ADMINISTRATIVE CONTROLS SECTION OF THE TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR TO THE PROCESS CONTROL PROGRAM (GENERIC LETTER 89-01)

The NRC staff has examined the contents of the Radiological Effluent Technical Specifications (RETS) in relation to the Commission's Interim Policy Statement on Technical Specification Improvements. The staff has determined that programmatic controls can be implemented in the Administrative Controls section of the Technical Specifications (TS) to satisfy existing regulatory requirements for RETS. At the same time, the procedural details of the current TS on radioactive effluents and radiological environmental monitoring can be relocated to the Offsite Dose Calculation Manual (ODCM). Likewise, the procedural details of the current TS on solid radioactive wastes can be relocated to the Process Control Program (PCP). These actions simplify the RETS, meet the regulatory requirements for radioactive effluents and radiological environmental monitoring, and are provided as a line-item improvement of the TS, consistent with the goals of the Policy Statement.

New programmatic controls for radioactive effluents and radiological environmental monitoring are incorporated in the TS to conform to the regulatory requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50. Existing programmatic requirements for the PCP are being retained in the TS. The procedural details included in licensees' present TS on radioactive effluents, solid radioactive wastes, environmental monitoring, and associated reporting requirements will be relocated to the ODCM or PCP as appropriate. Licensees will handle future changes to these procedural details in the ODCM and the PCP under the administrative controls for changes to the ODCM or PCP. Finally, the definitions of the ODCM and PCP are updated to reflect these changes.

Enclosure 1 provides guidance for the preparation of a license amendment request to implement these alternatives for RETS. Enclosure 2 provides a listing of existing RETS and a description of how each is addressed. Enclosure 3 provides model TS for programmatic controls for RETS and its associated reporting requirements. Finally, Enclosure 4 provides model specifications for retaining existing requirements for explosive gas monitoring instrumentation requirements that apply on a plant-specific basis. Licensees are encouraged to propose changes to TS that are consistent with the guidance provided in the enclosures. Conforming amendment requests will be expeditiously reviewed by

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the NRC Project Manager for the facility. Proposed amendments that deviate from this guidance will require a longer, more detailed review. Please contact the appropriate Project Manager if you have questions on this matter.

Sincerely,

Steven A. Varga
Acting Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1 through 4 as stated

GUIDANCE FOR THE IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RETS IN THE ADMINISTRATIVE CONTROLS SECTION OF TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF CURRENT RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR PROCESS CONTROL PROGRAM

INTRODUCTION

This enclosure provides guidance for the preparation of a license amendment request to implement programmatic controls in Technical Specifications (TS) for radioactive effluents and for radiological environmental monitoring conforming to the applicable regulatory requirements. This will allow the relocation of existing procedural details of the current Radiological Effluent Technical Specifications (RETS) to the Offsite Dose Calculation Manual (ODCM). Procedural details for solid radioactive wastes will be relocated to the Process Control Program (PCP). A proposed amendment will (1) incorporate programmatic controls in the Administrative Controls section of the TS that satisfy the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a. and Appendix I to 10 CFR Part 50, (2) relocate the existing procedural details in current specifications involving radioactive effluent monitoring instrumentation, the control of liquid and gaseous effluents, equipment requirements for liquid and gaseous effluents, radiological environmental monitoring, and radiological reporting details from the TS to the ODCM, (3) relocate the definition of solidification and existing procedural details in the current specification on solid radioactive wastes to the PCP, (4) simplify the associated reporting requirements, (5) simplify the administrative controls for changes to the ODCM and PCP, (6) add record retention requirements for changes to the ODCM and PCP, and (7) update the definitions of the ODCM and PCP consistent with these changes.

The NRC staff's intent in recommending these changes to the TS and the relocation of procedural details of the current RETS to the ODCM and PCP is to fulfill the goal of the Commission Policy Statement for Technical Specification Improvements. It is not the staff's intent to reduce the level of radiological effluent control. Rather, this amendment will provide programmatic controls for RETS consistent with regulatory requirements and allow relocation of the procedural details of current RETS to the ODCM or PCP. Therefore, future changes to these procedural details will be controlled by the controls for changes to the ODCM or PCP included in the Administrative Controls section of the TS. These procedural details are not required to be included in TS by 10 CFR 50.36a.

DISCUSSION

Enclosure 2 to Generic Letter 89-01 provides a summary listing of specifications that are included under the heading of RETS in the Standard Technical Specifications (STS) and their disposition. Most of these specifications will be addressed by programmatic controls in the Administrative Controls section of the TS. Some specifications under the heading of RETS are not covered by the new programmatic controls and will be retained as requirements in the existing plant TS. Examples include requirements for explosive gas monitoring instrumentation, limitations on the quantity of radioactivity in liquid or gaseous holdup or storage tanks or in the condenser exhaust for BWRs, or limitations on explosive gas mixtures in offgas treatment systems and storage tanks.

Licensees with nonstandard TS should follow the guidance provided in Enclosure 2 for the disposition of similar requirements in the format of their TS.

Because solid radioactive wastes are addressed under existing programmatic controls for the Process Control Program, which is a separate program from the new programmatic controls for liquid and gaseous radioactive effluents, the requirements for solid radioactive wastes and associated solid waste reporting requirements in current TS are included as procedural details that will be relocated to the PCP as part of this line-item improvement of TS. Also, the staff has concluded that records of licensee reviews performed for changes made to the ODCM and PCP should be documented and retained for the duration of the unit operating license. This approach is in lieu of the current requirements that the reasons for changes to the ODCM and PCP be addressed in the Semiannual Effluent Release Report.

The following items are to be included in a license amendment request to implement these changes. First, the model specifications in Enclosure 3 to Generic Letter 89-01 should be incorporated into the TS to satisfy the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50. The definitions of the ODCM and PCP should be updated to reflect these changes. The programmatic and reporting requirements are general in nature and do not contain plant-specific details. Therefore, these changes to the Administrative Controls section of the TS are to replace corresponding requirements in plant TS that address these items. They should be proposed for incorporation into the plant's TS without change in substance to replace existing requirements. If necessary, only changes in format should be proposed. If the current TS include requirements for explosive gas monitoring instrumentation as part of the gaseous effluent monitoring instrumentation requirements, these requirements should be retained. Enclosure 4 to Generic Letter 89- 01 provides model specifications for retaining such requirements.

Second, the procedural details covered in the licensee's current RETS, consisting of the limiting conditions for operation, their applicability, remedial actions, surveillance requirements, and the Bases section of the TS for these requirements, are to be relocated to the ODCM or PCP as appropriate and in a manner that ensures that these details are incorporated in plant operating procedures. The NRC staff does not intend to repeat technical reviews of the relocated procedural details because their consistency with the applicable regulatory requirements is a matter of record from past NRC reviews of RETS. If licensees make other than editorial changes in the procedural details being transferred to the ODCM, each change should be identified by markings in the margin and the requirements of new Specification 6.14a.(1) and (2) followed.

Finally, licensees should confirm in the amendment request that changes for relocating the procedural details of current RETS to either the ODCM or PCP have been prepared in accordance with the proposed changes to the Administrative Controls section of the TS so that they may be implemented immediately upon issuance of the proposed amendment. A complete and legible copy of the revised ODCM should be forwarded with the amendment request for NRC use as a reference. The NRC staff will not concur in or approve the revised ODCM.

Licensees should refer to "Generic Letter 89-01" in the Subject line of license amendment requests implementing the guidance of this Generic Letter. This will facilitate the staff's tracking of licensees' responses to this Generic Letter.

SUMMARY

The license amendment request for the line-item improvements of the TS relative to the RETS will entail (1) the incorporation of programmatic controls for radioactive effluents and radiological environmental monitoring in the Administrative Controls section of the TS, (2) incorporatation of the procedural details of the current RETS in the ODCM or PCP as appropriate, and (3) confirmation that the guidance of this Generic Letter has been followed.

DISPOSITION OF SPECIFICATIONS AND ADMINISTRATIVE CONTROLS
INCLUDED UNDER THE HEADING OF RETS IN THE STANDARD TECHNICAL SPECIFICATIONS

SPECIFICATION	TITLE	DISPOSITION OF EXISTING SPECIFICATION
1.17	OFFSITE DOSE CALCULATION MANUAL	Definition is updated to reflect the change in scope of the ODCM.
1.22	PROCESS CONTROL PROGRAM	Definition is updated to reflect the change in scope of the PCP.
1.32	SOLIDIFICATION	Definition is relocated to the PCP.
3/4.3.3.10	RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION	Programmatic controls are included in 6.8.4 g. Item 1). Existing specification procedural details are relocated to the ODCM.
3/4.3.3.11	RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION	Programmatic controls are included in 6.8.4 g. Item 1). Existing specification procedural details are relocated to the ODCM. Existing requirements for explosive gas monitoring instrumentation should be retained. Model specifications for these requirements are provided in Enclosure 4.
3/4.11.1.1	LIQUID EFFLUENTS: CONCENTRATION	Programmatic controls are included in 6.8.4 g. Items 2) and 3). Existing specification procedural details are relocated to the ODCM.
3/4.11.1.2	LIQUID EFFLUENTS: DOSE	Programmatic controls are included in 6.8.4 g. Items 4) and 5). Existing specification procedural details are relocated to the ODCM.
3/4.11.1.3	LIQUID EFFLUENTS: LIQUID RADWASTE TREATMENT SYSTEM	Programmatic controls are included in 6.8.4 g. Item 6). Existing specification procedural details are relocated to the ODCM.
3/4.11.1.4	LIQUID HOLDUP TANKS	Existing specification requirements to be retained.

DISPOSITION OF SPECIFICATIONS AND ADMINISTRATIVE CONTROLS
INCLUDED UNDER THE HEADING OF RETS IN THE STANDARD TECHNICAL SPECIFICATIONS (Cont.)

SPECIFICATION	TITLE	DISPOSITION OF EXISTING SPECIFICATION
3/4.11.2.1	GASEOUS EFFLUENTS: DOSE RATE	Programmatic controls are included in 6.8.4 g. Items 3) and 7). Existing specification procedural details are relocated to the ODCM.
3/4.11.2.2	GASEOUS EFFLUENTS: DOSE-NOBLE GASES	Programmatic controls are included in 6.8.4 g. Items 5) and 8). Existing specification procedural details are relocated to the ODCM.
3/4.11.2.3	GASEOUS EFFLUENTS: DOSE--IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM	Programmatic controls are included in 6.8.4 g. Items 5) and 9). Existing specification procedural details are relocated to the ODCM.
3/4.11.2.4	GASEOUS EFFLUENTS: GASEOUS RADWASTE TREATMENT or VENTILATION EXHAUST TREATMENT SYSTEM	Programmatic controls are included in 6.8.4 g. Item 6). Existing specification procedural details are relocated to the ODCM.
3/4.11.2.5	EXPLOSIVE GAS MIXTURE	Existing specification requirements should be retained.
3/4.11.2.6	GAS STORAGE TANKS	Existing specification requirements should be retained.
3/4.11.2.7	MAIN CONDENSER (BWR)	Existing specification requirements should be retained.
3/4.11.2.8	PURGING AND VENTING (BWR Mark II containments)	Programmatic controls are included in 6.8.4 g. Item 10). Existing specification procedural details are relocated to the ODCM.
3/4.11.3	SOLID RADIOACTIVE WASTES	Existing specification procedural details are relocated to the PCP.
3/4.11.4	RADIOACTIVE EFFLUENTS: TOTAL DOSE	Programmatic controls are included in 6.8.4 g. Item 11). Existing specification procedural details are relocated to the ODCM.

DISPOSITION OF SPECIFICATIONS AND ADMINISTRATIVE CONTROLS INCLUDED UNDER THE HEADING OF RETS IN THE STANDARD TECHNICAL SPECIFICATIONS (Cont.)

SPECIFICATION	TITLE	DISPOSITION OF EXISTING SPECIFICATION
3/4.12.1	RADIOLOGICAL ENVIRONMENTAL MONITORING: MONITORING PROGRAM	Programmatic controls are included in 6.8.4 h. Item 1). Existing specification procedural details are relocated to the ODCM.
3/4.12.2	RADIOLOGICAL ENVIRONMENTAL MONITORING: LAND USE CENSUS	Programmatic controls are included in 6.8.4 h. Item 2). Existing specification procedural details are relocated to the ODCM.
3/4.12.3	RADIOLOGICAL ENVIRONMENTAL MONITORING: INTERLABORATORY COMPARISON PROGRAM	Programmatic controls are included in 6.8.4 h. Item 3). Existing specification procedural details are relocated to the ODCM.
5.1.3	DESIGN FEATURES: SITE - MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS	Existing specification requirements should be retained.
6.9.1.3	REPORTING REQUIREMENTS: ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT	Specification simplified and existing reporting details are relocated to the ODCM.
6.9.1.4	REPORTING REQUIREMENTS: SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT	Specification simplified and existing reporting details are relocated to the ODCM or PCP as appropriate.
6.13	PROCESS CONTROL PROGRAM	Specification requirements are simplified.
6.14	OFFSITE DOSE CALCULATION MANUAL	Specification requirements are simplified.
6.15	MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS	Existing procedural details are relocated to the ODCM or PCP as appropriate.

TECHNICAL SPECIFICATIONS TO BE REVISED

1.17	DEFINITIONS: OFFSITE DOSE CALCULATION MANUAL
1.22	DEFINITIONS: PROCESS CONTROL PROGRAM
6.8.4 g.	PROCEDURES AND PROGRAMS: RADIOACTIVE EFFLUENT CONTROLS
6.8.4 h.	PROCEDURES AND PROGRAMS: RADIOLOGICAL ENVIRONMENTAL MONITORING
6.9.1.3	REPORTING REQUIREMENTS: ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT
6.9.1.4	REPORTING REQUIREMENTS: SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
6.10	RECORD RETENTION
6.13	PROCESS CONTROL PROGRAM (PCP)
6.14	OFFSITE DOSE CALCULATION MANUAL (ODCM)

MODEL TECHNICAL SPECIFICATION REVISIONS
(To supplement or replace existing specifications)

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.3 and 6.9.1.4.

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

6.0 ADMINISTRATIVE CONTROLS

6.8 PROCEDURES AND PROGRAMS

6.8.4 The following programs shall be established, implemented, and maintained:

g. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

ADMINISTRATIVE CONTROLS

6.8.4 g. Radioactive Effluent Controls Program (Cont.)

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on venting and purging of the Mark II containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable (BWRs w/Mark II containments), and

11) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

h. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

ADMINISTRATIVE CONTROLS

6.9 REPORTING REQUIREMENTS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.4 The Semiannual Radioactive Effluent Release Report 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 report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

6.10 RECORD RETENTION

6.10.3 The following records shall be retained for the duration of the unit Operating License:

o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3o. This documentation shall contain:

    1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

*A single submittal may be made for a multi-unit station.

**A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

ADMINISTRATIVE CONTROLS

6.13 PROCESS CONTROL PROGRAM (PCP) (Cont.)

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the [URG] and the approval of the Plant Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3o. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the [URG] and the approval of the Plant Manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

MODIFICATION OF THE SPECIFICATION FOR RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION TO RETAIN REQUIREMENTS FOR EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENTATION

EXPLOSIVE
~~RADIOACTIVE~~ GASE~~OUS-EFFLUENT~~ MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The explosive ~~radioactive~~ gase~~ous-effluent~~ monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification~~s-3.11.2.1-and~~ 3.11.2.5 are not exceeded. ~~The-Alarm/Trip-Setpoints-of-these-channels-meeting-Specification 3.11.2.1-shall-be-determined-and-adjusted-in-accordance-with-the-methodology and-parameters-in-the-ODCM.~~

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With an explosive ~~radioactive~~ gase~~ous-effluent~~ monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification~~,-immediately-suspend-the-release-of-radioactive gaseous-effluents-monitored-by-the-affected-channel,-or~~ declare the channel inoperable and take the ACTION shown in Table 3.3-13.

b. With less than the minimum number of explosive ~~radioactive~~ gase~~ous-effluent~~ monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful ~~explain-in-the-next-Semi-annual-Radioactive-Effluent-Release-Report~~ prepare and submit a Special Report to the Commission pursuant to Specification ~~6.9.1.4~~ 6.9.2 to explain why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each explosive ~~radioactive~~ gase~~ous-effluent~~ monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, ~~SOURCE CHECK,~~ CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

EXPLOSIVE ~~RADIOACTIVE~~ ~~GASEOUS~~ ~~EFFLUENT~~ MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. (Not used)			
2A. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitor (Automatic Control)	1	**	49
b. Hydrogen or Oxygen Monitor (Process)	1	**	49
2B. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems not designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitors (Automatic Control, redundant)	2	**	50, 52
b. Hydrogen or Oxygen Monitors (Process, dual)	2	**	50

TABLE 3.3-13 (Continued)

* (Not used)

** During WASTE GAS HOLDUP SYSTEM operation.

ACTION STATEMENTS

ACTION 45 - (Not used)

ACTION 46 - (Not used)

ACTION 47 - (Not used)

ACTION 48 - (Not used)

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this WASTE GAS HOLDUP SYSTEM may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 50 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

ACTION 51 - (Not used)

ACTION 52 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

TABLE 4.3-9

EXPLOSIVE
~~RADIOACTIVE~~ GAS~~EOUS~~ ~~EFFLUENT~~ MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	~~SOURCE CHECK~~	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. (Not used)					
2A. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)					
a. Hydrogen Monitor (Automatic Control)	D	N.A.	Q(4)	M	**
b. Hydrogen or Oxygen Monitor (Process)	D	N.A.	Q(4) or Q(5)	M	**
2B. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems not designed to withstand the effects of a hydrogen explosion)					
a. Hydrogen Monitors (Automatic Control, redundant)	D	N.A.	Q(4)	M	**
b. Hydrogen or Oxygen Monitors (Process, dual)	D	N.A.	Q(4) or Q(5)	M	**

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* (Not used)

** During WASTE GAS HOLDUP SYSTEM operation.

(1) (Not used)

(2) (Not used)

(3) (Not used)

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

a. One volume percent hydogen, balance nitrogen, and

c. Four volume percent hydrogen, balance nitrogen.

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

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter No.	Subject	Date of Issuance	Issued To
88-20	INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES - 10 CFR 50.54(f)	11/23/88	ALL LICENSEES HOLDING OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTOR FACILITIES
88-19	USE OF DEADLY FORCE BY LICENSEE GUARDS TO PREVENT THEFT OF SPECIAL NUCLEAR MATERIAL	10/28/88	ALL FUEL CYCLE FACILITY LICENSEES WHO POSSESS, USE, IMPORT, EXPORT, OR TRANSPORT FORMULA QUANTITIES OF STRATEGIC SPECIAL NUCLEAR MATERIAL
88-18	PLANT RECORD STORAGE ON OPTICAL DISKS	10/20/88	ALL LICENSEES OF OPERATING REACTORS AND HOLDERS OF CONSTRUCTION PERMITS
88-17	LOSS OF DECAY HEAT REMOVAL 10 CFR 50.54(f)	10/17/88	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR PRESSURIZED WATER REACTORS
88-16	REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS FROM TECHNICAL SPECIFICATIONS	10/04/88	ALL POWER REACTOR LICENSEES AND APPLICANTS
88-15	ELECTRIC POWER SYSTEMS - INADEQUATE CONTROL OVER DESIGN PROCESSES	09/12/88	ALL POWER REACTOR LICENSEES AND APPLICANTS
88-14	INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT	08/08/88	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS
88-13	OPERATOR LICENSING EXAMINATIONS	08/08/88	ALL POWER REACTOR LICENSEES AND APPLICANTS FOR AN OPERATING LICENSE.
88-12	REMOVAL OF FIRE PROTECTION REQUIREMENTS FROM TECHNICAL SPECIFICATIONS	08/02/88	ALL POWER REACTOR LICENSEES AND APPLICANTS