TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

<u>TABLE OF CONTENTS</u> (Continued)

TABLE OF CONTENTS (Continued)

ATTACHMENT 1

TABLE OF CONTENTS - TABLES

TABLE OF CONTENTS

TECHNICAL SPECIFICATIONS - TABLES

TABLE OF CONTENTS (ALPHABETICAL ORDER)
Continued

DEFINITIONS

Azimuthal Power Tilt - T_q

Azimuthal Power Tilt shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that axial half (upper or lower) of the core divided by the average power of all quadrants in that axial half (upper or lower) of the core.

Unrodded Planer Radial Peaking Factor - F_{xy}

The Unrodded Planar Radial Peaking Factor is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding azimuthal tilt, T_q. The maximum F_{xy} limit is provided in the Core Operating Limits Report.

Unrodded Integrated Radial Peaking Factor - F_R

The Unrodded Integrated Radial Peaking Factor is the ratio of the peak pin power to the average pin power in an unrodded core, excluding azimuthal tilt, T_q. The maximum F_R limit is provided in the Core Operating Limits Report.

Fire Suppression Water System

The fire suppression water system consists of fire pumps and distribution piping with associated sectionalizing control or isolation valves. Such valves 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.

~~Process Control Program (PCP)~~

~~A manual or set of operating procedures detailing the program of sampling, analysis, and evaluation.~~ Attachment 2 →

Dose Equivalent I-131

That concentration of I-131 (μCi/gm) 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. In other words,

DEFINITIONS

Dose Equivalent I-131 (μCi/gm) = μCi/gm of I-131

+ 0.0361 x μCi/gm of I-132

+ 0.270 x μCi/gm of I-133

+ 0.0169 x μCi/gm of I-134

+ 0.0838 x μCi/gm of I-135

$\bar{E}$ - Average Disintegration Energy

$\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Attachment 3 →

~~Offsite Dose Calculation Manual (ODCM)~~

A manual containing the methodology and parameters to be used in the: 1) calculation of doses in the unrestricted area due to radioactive liquid and gaseous effluents, 2) calculation of liquid and gaseous effluent monitoring instrumentation setpoints, and 3) specific details pertinent to the radiological environmental monitoring program.

Purge-Purging

A means for the removal and replacement of gases within the containment building.

Venting

~~A means for the reduction of pressure greater than atmospheric within the containment structure.~~

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

References

(1) USAR, Section 7.2

(2) USAR, Section 7.3

Attachment 2

Process Control Program (PCP)

The document(s) that contains the current formulas, sampling, analyses, tests, 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 20, 61, 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

Attachment 3

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and Semiannual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area <u>at or beyond the site boundary</u> access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)

2.1.3 Reactor Coolant Radioactivity

Applicability

Applies to the radioactivity of the reactor coolant.

Objective

To ensure that the reactor coolant radioactivity is maintained at a level commensurate with the occupational and public safety.

Specification

(1) The radioactivity of the reactor coolant shall be limited to:

a. $\leq$ 1.0 μCi/gm DOSE EQUIVALENT I-131, and

b. $\leq 100/\bar{E}$ μCi/gm

(2) With the radioactivity of the reactor coolant >1.0 μCi/gm DOSE EQUIVALENT I-131 for more than 100 hours during one continuous time interval or exceeding 60 μCi/gm, be in at least HOT SHUTDOWN with T_{avg} <536°F within 6 hours.

(3) With the radioactivity of the reactor coolant > $100/\bar{E}$ μCi/gm, be in at least HOT SHUTDOWN with T_{avg} < 536°F within 6 hours.

(4) With the radioactivity of the reactor coolant >1.0 μCi/gm DOSE EQUIVALENT I-131, perform the sampling and analysis requirements of items 1.(a)(2)(ii) and 1.(b)(2)(i) of Table 3-4 until the radioactivity of the reactor coolant is restored to within its limits. Data pursuant to Specification 5.9.4.b ~~(4)~~ for the Annual Report shall be compiled as follows:

a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded

b. Purification System flow history starting 48 hours prior to the first sample in which the limit was exceeded.

c. The time duration when the radioactivity of the reactor coolant exceeded 1.0 μCi/gm DOSE EQUIVALENT I-131.

d. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should contain the date and time of sampling and the radioiodine concentrations.

2.0 LIMITING CONDITIONS FOR OPERATION

2.8 Refueling Operations

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

Specifications

The following conditions shall be satisfied during any refueling operations:

gaseous radiation monitor

(1) The equipment hatch and one door in the air lock shall be properly closed. In addition, all automatic containment isolation valves shall be operable or at least one valve in each line shall be closed.

One / one Auxiliary Building Exhaust Stack gaseous

(2) ~~The five~~ containment atmosphere and ~~plant ventilation duct~~ radiation monitor~~s~~ that initiate closure of the containment pressure relief, air sample, and purge system valves shall be tested and verified to be operable immediately prior to refueling operations. The ~~five~~ two monitors shall employ one-out-of-~~five~~ two logic from separate contact outputs for VIAS.

(3) Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

(4) Whenever core geometry is being changed, neutron flux shall be continuously monitored by at least two source range neutron monitors, with each monitor providing continuous visual indication in the control room. When core geometry is not being changed, at least one source range neutron monitor shall be in service.

(5) At least one shutdown cooling pump and heat exchanger shall be in operation. However, the pump and heat exchanger may be removed from operation for up to one hour per 8 hour period during the performance of core alterations in the vicinity of the reactor coolant hot leg loops or during manipulation of a source.

(6) Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

(7) When irradiated fuel is being handled in the auxiliary building, the exhaust ventilation from the spent fuel pool area will be diverted through the charcoal filter.

(8) Prior to initial core loading and prior to refueling operations, a complete check out, including a load test, shall be conducted on fuel handling cranes that will be required during the refueling operation to handle spent fuel assemblies.

(9) A minimum of 23 feet of water above the top of the core shall be maintained whenever irradiated fuel is being handled.

(10) Storage in Region 1 and Region 2 of the spent fuel racks shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.0 weight percent of U-235.

(11) Storage in Region 2 of the spent fuel racks shall be restricted to those assemblies whose parameters fall within the "acceptable" region of Figure 2-10.

If any of the above conditions are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions, and no operations that may change the reactivity of the core shall be made. ~~However, refueling operations may commence and continue with less than 5 containment atmosphere and plant ventilation duct radiation monitors provided that gross, particulate and iodine monitors are monitoring the stack effluent. These three plant ventilation duct radiation monitors will initi te closure of the containment pressure relief, air sample and purge system valves and shall employ a one-out-of three logic for the initiation of VIAS.~~

A ~~The~~ spent fuel assembly may be transferred directly from the reactor core to the spent fuel pool Region 2 provided the independent verification of assembly burnups ~~as defined in Special Procedure SP-BURNUP-1~~ has been completed and the assembly burnup meets the acceptance criteria identified in Technical Specification Figure 2-10.

from the reactor core

Movement of ~~I~~rradiated fuel ~~movement~~ shall not be initiated before the reactor core has ~~decayed~~ for a minimum of 72 hours if the reactor has been operated at power levels in excess of 2% rated power.

been subcritical

Basis

The equipment and general procedures to be utilized during refueling operations are discussed in the USAR. Detailed instructions, the above specifications, and the design of the fuel handling equipment incorporating built-in interlocks and safety features provide assurance that no

Replace with Attachment 4

2.0 LIMITING CONDITIONS FOR OPERATIONS
2.9 Radioactive Effluents
2.9.1 Liquid and Gaseous Effluents

Applicability

Applies to the controlled release of radioactive materials in liquid and gaseous effluents from the facility. The provisions of Technical Specification 2.0.1 for Limiting Condition for Operation are not applicable.

Objective

To define the limits and conditions for the controlled release of radioactive materials in liquid and gaseous effluents to the environs to ensure that these releases are as low as is reasonably achievable in conformance with 10 CFR Part 50.34a and 50.36a, and to ensure that these releases result in concentrations of radioactive materials in liquid and gaseous effluents released to unrestricted areas that are within the limits specified in 10 CFR Part 20.

To ensure that the releases of radioactive materials above background to unrestricted areas are as low as is reasonably achievable, the following design objectives apply.

A. Liquid Effluents

(1) The dose or dose commitment to a member of the public during any calendar year should not exceed 3 millirems to the total body.

(2) The dose or dose commitment to a member of the public during any calendar year should not exceed 10 millirems to any organ.

B. Gaseous Effluents

(1) The calculated annual air dose due to gamma radiation at any location which could be occupied by individuals in unrestricted areas should not exceed 10 millirads;

(2) The calculated annual air dose due to beta radiation at any location which could be occupied by individuals in unrestricted areas should not exceed 20 millirads; and

(3) The calculated annual total quantity of iodine-131, tritium, and all radioactive material in particulate form with half-lives greater than 8 days should not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 millirems.

Attachment 4

2.0 LIMITING CONDITIONS FOR OPERATION

2.9 Radioactive Waste Disposal System

Applicability

Applies to the transfer of waste gases to the waste gas decay tanks. The provisions of Technical Specification 2.0.1 for Limiting Condition for Operation are not applicable.

Objective

To ensure compliance with General Design Criterion 60 of Appendix A to 10 CFR 50.

Specification

(1) The concentration of hydrogen and oxygen in the waste gas decay tanks shall be limited to below flammability concentrations. With hydrogen and oxygen concentrations above flammability concentrations, restore the concentrations to below flammability limits within 48 hours.

(2) The hydrogen and oxygen monitors shall be monitoring the inservice gas decay tank during the transfer of waste gases to the waste gas decay tank. Whenever the monitors are inoperable, transfer of waste gases to a gas decay tank may continue provided grab samples are taken from the gas decay tank and analyzed:

a. Every eight hours during degassing operations, and
b. Daily during other operations.

Basis

Specification 2.9 ensures that the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits with a measurement program 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.

(Next Page is 2-48)

2.9.1 Liquid and Gaseous Effluents (Continued)

(1) Specifications for Liquid Waste Effluents

a. (i) The release rate of radioactive material in liquid effluents shall be controlled such that the instantaneous concentrations for radionuclides, other than dissolved or entrained noble gases, do not exceed the values specified in 10 CFR Part 20, Appendix B, for unrestricted areas. For dissolved or entrained noble gases, the concentration shall be limited to 2.0 E-04 microCi/ml total activity.

(ii) With the concentration of radioactive material released to unrestricted areas exceeding the above limits, appropriate corrective actions shall be taken immediately to restore concentrations within the above limits.

b. The cumulative dose contributions from radioactive materials in liquid effluents released to unrestricted areas shall be determined, in accordance with the ODCM, on a quarterly basis. If the dose contributions, due to the cumulative release of liquid effluents averaged over a calendar quarter, exceed one-half of the design objectives, the following course of actions shall be taken:

(i) Make an investigation to identify the causes for such releases.

(ii) Define and initiate a program of action to reduce such releases to the design levels.

(iii) Submit a special report, pursuant to Specification 5.9.3, within 30 days from the end of the quarter during which release occurred, identifying the causes and describing the proposed program of action to reduce such release to the design levels.

c. The equipment or subsystem(s) of the liquid radwaste treatment system as identified in the ODCM shall be operated prior to the discharge of radioactive materials in liquid wastes. If the radioactive liquid wastes were discharged without treatment by one or more of the pieces of equipment or subsystem(s) identified in the ODCM and it appears that one-half of the annual objective will be exceeded during the calendar quarter, a special report, pursuant to Specification 5.9.3, shall be prepared and submitted to the Commission within 30 days. This report shall include the following information:

(i) Identification of equipment or subsystems not operable and reasons for inoperability.

(ii) Action(s) taken to restore the inoperable equipment to status.

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

d. During release of radioactive liquid waste excluding releases from the steam generators, the following conditions shall be met:

(i) At least one circulating water pump shall be in operation to provide a dilution flow of approximately 120,000 gpm in the discharge tunnel.

(ii) The overboard header effluent radiation monitor shall be set in accordance with the ODCM to alarm and automatically close the discharge valve prior to exceeding the limits specified in 2.9.1(1)a.(i) above.

(iii) The gross liquid waste activity and flow rate shall be continuously monitored and recorded during the release. If the effluent radiation monitor is inoperable, effluent releases may continue provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Specification 3.12.1(1).

2. At least two qualified individuals independently verify the release rate calculations.

If the flow rate indicator is inoperable, effluent releases may continue provided the flow rate is determined at least once per 4 hours during actual release.

If the radioactivity cannot be recorded automatically, effluent releases may continue provided the gross radioactivity level is recorded manually at least once per 4 hours during actual release.

e. Whenever steam generator liquid is being released to the discharge tunnel 1) the steam generator blowdown radiation monitors shall be set to alarm and automatically close the blowdown isolation valves prior to exceeding the limits specified in 2.9.1(1)a(i) above, and 2) the gross activity for each blowdown line shall be monitored and recorded by the blowdown radiation monitors. If one of the two radiation monitors is inoperable, the activity for both blowdown lines shall be monitored by the operable radiation monitor. If both radiation monitors are inoperable, steam generator liquid release may continue provided appropriate grab samples are analyzed for principal gamma emitters at a sensitivity of 5.0E-07 μCi/ml and recorded at least daily when the specific activity of the sample is less than or equal to 0.01 μCi/gram dose equivalent I-131 and at least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 μCi/gram dose equivalent I-131. If the radioactivity cannot be recorded automatically, effluent releases may continue provided the gross radioactivity level is recorded manually at least once per four hours during actual release.

(2) Specifications for Gaseous Waste Effluents

a. (i) The release rate of radioactive materials in gaseous effluents shall be controlled such that the instantaneous concentrations of radionuclides do not exceed the values specified in 10 CFR Part 20, Appendix B, Table 2 for unrestricted areas. Unrestricted area concentrations shall be calculated based on the annual average Chi/Q.

(ii) With the concentration of radioactive material released to unrestricted areas exceeding the above limits, appropriate corrective actions shall be taken immediately to restore concentration within the above limits.

b. The radiation dose contributions from radioactive materials in gaseous effluents shall be determined, in accordance with the ODCM, on a quarterly basis. If the dose contributions, due to the cumulative release of gaseous effluents averaged over a calendar quarter exceed one-half of the design objectives, the following course of actions shall be taken:

(i) Make an investigation to identify the cause for such release rates.

(ii) Define and initiate a program of action to reduce such releases to design levels.

(iii) Submit a special report, pursuant to Specification 5.9.3, within 30 days from the end of the quarter during which release occurred, identifying the causes and describing the proposed program of action to reduce dose contributions.

c. The equipment of subsystem(s) of the gaseous radwaste treatment system as identified in the ODCM shall be operated prior to the discharge of radioactive materials in gaseous wastes. If the radioactive gaseous wastes were discharged without treatment by one or more of the equipment or subsystem(s) identified in the ODCM, a special report, pursuant to Specification 5.9.3, shall be prepared and submitted to the Commission within 30 days. This report shall include the following information:

(i) Identification of equipment or subsystem(s) not operable and reason for inoperability.

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

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

d. The hydrogen and oxygen monitors shall be monitoring the in-service gas decay tank during the transfer of waste gases to the gas decay tank and the concentration of hydrogen and oxygen shall be limited to below flammability concentrations. Whenever the monitors are inoperable, transfer of waste gases to a gas decay tank may continue provided grab samples are taken from the gas decay tank and analyzed: (1) every 8 hours during degassing operations, and (2) daily during other operations.

e. The Auxiliary Building Exhaust Stack gaseous, particulate, and iodine activity monitors may be inoperable provided that 1) releases from a gas decay tank, containment pressure relief line, and the containment purge line are secured, and 2) whenever the Auxiliary Building Exhaust Stack gas or particulate activity monitor is inoperable, appropriate grab samples will be taken and analyzed once per eight (8) hours.

f. During power operation, the condenser air ejector discharge shall be monitored for gross radioactivity. If this monitor is inoperable, grab samples shall be taken and analyzed daily for principal gamma emitters.

g. During release of gaseous radioactive wastes from the gaseous waste discharge header or during containment venting to the Auxiliary Building Exhaust Stack, the following conditions shall be met:

(i) The gas, iodine, and particulate monitors shall be monitoring the Auxiliary Building Exhaust Stack.

(ii) At least one exhaust fan shall be in operation.

(iii) The effluent control radiation monitors shall be set in accordance with the ODCM to alarm and automatically terminate the releases prior to exceeding the limits specified in 2.9.1(2)a(i) above.

(iv) The activity shall be monitored and recorded. The flow rate shall be monitored and recorded, or determined by calculation.

(v) During the release of gaseous wastes from the containment purge line, a containment gas monitor and a particulate monitor shall monitor the containment, in addition to conforming with (i) through (iv) above.

h. During releases from the Laboratory and Radioactive Waste Processing Building Exhaust Stack, the following conditions shall be met:

(i) The Laboratory and Radioactive Waste Processing Building (LRWPB) Exhaust Stack gas, iodine, and particulate monitors shall be monitoring the LRWPB Exhaust Stack. The effluent control radiation monitors shall be set in accordance with the ODCM to alarm prior to exceeding the limits specified in 2.9.1(2)a(i) above. The gas activity monitor may be inoperable provided that appropriate grab samples be taken and analyzed once per 24 hours. The particulate and iodine activity monitors may be inoperable provided that samples are continuously collected as required in Table 3-12.

(ii) The effluent flow rate shall be monitored and recorded, or determined by calculation.

Basis

Releases of radioactivity in liquid wastes within the design objective levels provide reasonable assurance that the resulting annual exposure from liquid effluents will not exceed the limits specified in Appendix I to 10 CFR Part 50. These specifications provide reasonable assurance that the resulting exposure will not exceed 3 mrem to total body or 10 mrem to any organ. At the same time, these specifications permit the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20.

The design objectives have been developed based on operating experience, calculation procedures based on models and data set forth in Regulatory Guide 1.109, and the evaluation of Fort Calhoun facility in accordance with Appendix I of 10 CFR Part 50 dose design objectives. The design objectives take into account a combination of variables including fuel failures, primary system leakage, primary-to-secondary system leakage and the performance of various radioactive waste treatment systems.

Specification 2.9.1(1)a requires the licensee to limit the concentration of radioactive materials in liquid effluents released from the site to levels specified in 10 CFR Part 20, Appendix B, for unrestricted areas. This specification provides assurance that no member of the general public will be exposed at any time to liquid

Basis (Continued)

containing radioactive materials in excess of limits considered permissible under the Commission's Regulations.

Specification 2.9.1(1)b establishes the frequency of dose calculations in accordance with the ODCM. This specification also establishes the reporting requirements in accordance with Section IV.A of Appendix I to 10 CFR Part 50, in addition to the requirements of Section 5.9 of these Technical Specifications.

Specification 2.9.1(1)c requires the operation of the equipment or subsystem(s) of the radioactive liquid waste system, as identified in the ODCM, to reduce the release of radioactive materials in liquid effluents to as low as reasonably achievable, consistent with the requirements of 10 CFR Part 50.36a, and General Design Criterion 60 of Appendix A to 10 CFR Part 50. Normal use of the equipment or subsystem(s) in the radioactive liquid waste system provides reasonable assurance that the quantity released will not exceed the design objectives.

Basis (Continued)

Specification 2.9.1(1)d, consistent with the requirements of General Design Criteria 60 and 64 of 10 CFR Part , Appendix A, requires operation of suitable equipment to dilute, control, and monitor the releases of radioactive materials in liquid wastes, other than steam generator liquid, from the overboard header during any period when releases are taking place.

Specification 2.9.1(1)e requires the monitoring of the steam generator liquid when releases are being discharged to the environment. Inoperability of one radiation monitor will not affect the monitoring capabilities as the other radiation monitor would serve the intended purpose. If both radiation monitors are found inoperable and if steam generator liquid is being released to the environment, the specified sampling frequency provides assurance that no major activity is released during a limited period of time when repairs are being made.

The release of radioactive materials in gaseous waste effluents to unrestricted areas will not result in concentrations that exceed limits specified in 10 CFR Part 20 at any time and should be as low as is reasonably achievable in accordance with the requirements of 10 CFR Parts 50.34a and 50.36a. These specifications provide reasonable assurance that the resulting annual air dose due to gamma radiation will not exceed 10 mrad and that the resulting annual air dose to beta radiation will not exceed 20 mrad from the gaseous waste effluents from the plant. These specifications also provide reasonable assurance that no individual in an unrestric area will receive an annual dose to the total body greater than 5 mrem or an annual dose to the skin greater than 15 mrem from these gaseous effluents, and that the annual dose to any organ of an individual from radioiodines and radioactive material in particulate form will not exceed 15 mrem.

At the same time, these specifications permit the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided with a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such numerical guides for design objectives but still within levels that assure that the average population exposure is equivalent to small fractions of doses from natural background radiation.

Specification 2.9.1(2)a requires the licensee to limit the concentration of radioactive materials in gaseous effluents from the station to levels specified in 10 CFR Part 20, Appendix B, for unrestricted areas. This specification provides assurance that no member of the general public will be exposed at any time to gases containing radioactive materials in excess of limits specified in the Commission's regulations.

Specification 2.9.1(2)b establishes the frequency of dose calculations in accordance with the ODCM. This specification also establishes the reporting

2.0 LIMITING CONDITIONS FOR OPERATIONS
2.9 Radioactive Effluents (Continued)
2.9.1 Liquid and Gaseous Effluents (Continued)

Basis (Continued)

requirements in accordance with Section IV.A of Appendix I to 10 CFR Part 50, in addition to the requirements of Section 5.9 of these Technical Specifications.

Specification 2.9.1(2)c requires the operation of equipment or subsystem(s) of the radioactive gaseous waste system, as identified in the ODCM, to reduce the release of radioactive materials in gaseous effluents to as low as reasonably achievable, consistent with the requirements of 10 CFR Part 50.36a, and General Design Criterion 60 of Appendix A to 10 CFR Part 50. Normal use of the equipment or subsystem(s) in the radioactive gaseous waste system provides reasonable assurance that the quantity released will not exceed the design objectives.

Specification 2.9.1(2)d ensures that the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits with a measurement program 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.

Specification 2.9.1(2)e provides assurance that releases from gas decay tank, Auxiliary Building Exhaust Stack, containment pressure relief line, and containment purge line are not made whenever the stack gas, particulate and iodine monitors are inoperable.

Specification 2.9.1(2)f assures that gross radioactivity, during power operation, is monitored from the condenser air ejector discharge.

Specification 2.9.1(2)g requires operation of suitable equipment to dilute, control, and monitor in order to provide assurance that radioactive materials released in the gaseous effluents are properly controlled and monitored in accordance with the requirements of General Design Criteria 60 and 64 of 10 CFR Part 50, Appendix A.

Specification 2.9.1(2)h provides for releases from the Laboratory and Radioactive Waste Processing Building (LRWPB) whenever the LRWPB Exhaust Stack gas, particulate or iodine activity monitors are inoperable.

2.0 LIMITING CONDITIONS FOR OPERATIONS
2.9 Radioactive Effluents (Continued)
2.9.2 Solid Radioactive Waste

Applicability

This specification applies to the processing and packaging of solid and compacted radwaste.

Objective

To ensure conformance with 10 CFR Part 20 and 10 CFR Part 71 prior to shipment of solidified radwaste from the facility. The provisions of Technical Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Specification

The equipment or subsystem(s) of the solid radwaste system, as identified in the Process Control Program (PCP), shall be operated to provide for the solidification of wet solid wastes and the compaction of compressible wastes. Waste solidification will be verified by requirements specified in the PCP. If solidified radwaste fails to meet the above "objective" regulations or the acceptance criteria of the PCP, no offsite shipments shall be made of the non-conforming materials.

Basis

The solid radwaste system is generally operated on a batch basis, and is available to perform abnormal or emergency functions. The proper operation of the solid radwaste system ensures that the pertinent requirements of 10 CFR Part 20 and 10 CFR Part 71 will be implemented. This specification also complies with the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The operating procedures, process parameters and the acceptance criteria, included in the Process Control Program, will provide compliance with these requirements.

2.0 LIMITING CONDITIONS FOR OPERATION

2.14 Engineered Safety Features System Initiation Instrumentation Settings

Applicability

Applies to the engineered safety features system initiation instrumentation settings.

Objective

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded.

Specifications

The engineered safety features system initiation instrumentation setting limits shall be as stated in Table 2-1.

Basis

(1) High Containment Pressure

The basis for the 5 psig set point for the high pressure signal is to establish a setting which would be exceeded quickly in the event of a DBA, cover a spectrum of break sizes, and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

High containment pressure initiates the steam generator isolation signal which will close the main steam isolation and bypass valves and the main feedwater isolation and bypass valves.

(2) Pressurizer Low Pressure

The pressurizer low pressure safety injection signal is a diverse signal to the high containment pressure safety injection signal. The 1600 psia setting includes an uncertainty of $\pm$ 22 psia and is the setting used in the safety analysis.(1)

(3) Containment High Radiation (Air Monitoring)

~~The containment air monitoring system comprises a moving paper filter particle monitor (channel RM-050) and a sample chamber gas monitor (channel RM-051) installed in a common housing.(2)~~

~~Optionally, the sampling point for channels RM-050 and RM-051 can be switched from the containment to the ventilation discharge duct.~~

~~The ventilation discharge monitoring system consists of a moving paper filter particle monitor (RM-061) and a sample chamber gas monitor (RM-062) installed in a common housing. An iodine monitor for I-131 (RM-060) also monitors these releases.~~

The containment radiation high signal can be initiated by a containment atmosphere gaseous radiation monitor or an Auxiliary Building Exhaust Stack gaseous radiation monitor.(2)

(3) Containment High Radiation (Air Monitoring) (Continued)

The setpoints for the isolation function will be calculated in accordance with the ODCM.

~~Each channel in supplied from a separate instrument A.C. bus and each auxiliary relay requires power to operate. On failure of a single A.C. supply, the A and B matrices will assume a one-out-of-two logic.~~

(4) Low Steam Generator Pressure

A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a $\pm$22 psi uncertainty and was the setting used in the safety analysis.[3]

Closure of the MSIVs (and the bypass valves, along with main feedwater isolation and bypass valves) is accomplished by the steam generator isolation signal which is a logical combination of low steam generator pressure or high containment pressure.

As part of the AFW actuation logic, a separate signal is provided to terminate flow to a steam generator upon sensing a low pressure in that steam generator if the other steam generator pressure is greater than the pressure setting. This is done to minimize the temperature reduction in the reactor coolant system in the event of a main steam-line break. The setting of 466.7 psia includes a +31.7 psi uncertainty; therefore, a setting of 435 psia was used in the safety analysis.

(5) SIRW Tank Low Level

Level switches are provided on the SIRW tank to actuate the valves in the safety injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of approximately 24 minutes following a safety injection signal. The switch-over point of 16 inches above tank bottom is set to prevent the pumps from running dry during the 10 seconds required to stroke the valves and to hold in reserve approximately 28,000 gallons of water of at least the refueling boron concentration. The FSAR loss of coolant accident analysis[4] assumed the recirculation started when the minimum usable volume of 283,000 gallons had been pumped from the tank.

2.0 LIMITING CONDITIONS FOR OPERATIONS

2.14 Engineered Safety Features System Initiation Instrumentation Settings (Continued)

(6) Low Steam Generator Water Level

As part of the AFW actuation logic, a signal is provided to initiate AFW flow to one or two steam generators upon sensing a low water level in the steam generator(s) if the absolute steam generator pressure criteria are satisfied. This function ensures adequate steam generator water level is maintained in the event of a failure to deliver main feedwater to either steam generator. The setting of 28.2% of wide range tap span includes a +13.2% uncertainty; therefore, a setting of 15% of wide range tap span was used in the safety analysis.

(7) High Steam Generator Delta Pressure

As part of the AFW logic, a high steam generator differential pressure signal is generated to provide AFW to the higher pressure steam generator with a concurrent low level signal if both steam generator pressures are less than 466.7 psia. If the differential pressure between steam generators is less than the setting, neither steam generator is supplied with AFW in the presence of a low level signal. The setting of 119.7 psid includes a -15.3 psi uncertainty; therefore, a setting of 135 psid was used in the AFW safety analysis.

References

(1) USAR, Section 14.1.3

(2) USAR, Section ~~11.2.3.2~~ 7.3.2

(3) USAR, Section 14.12

(4) USAR, Section 14.15

(5) USAR, Section 7.4.6

(6) USAR, Section 7.5.2.5

(7) USAR, Section 14.4.1

(5) In the event that any of the following Emergency Auxiliary Feedwater Panel instrumentation or control circuits become inoperable, either restore the inoperable component(s) to operable status within seven days, or be in hot shutdown within the next twelve hours. This specification is applicable in Modes 1 and 2.

Steam Generator Level, Wide Range (AI-179)

Steam Generator Level, Narrow Range (AI-179)

Steam Generator Pressure (AI-179)

Pressurizer Pressure (AI-179)

Basis

During plant operation, the complete instrumentation systems will normally be in service. Reactor safety is provided by the reactor protection system, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels are out of service.

All reactor protection and almost all engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in engineered safeguards control system.

When one of the four channels is taken out of service for maintenance, the protective system logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),(1) which results in a one-out-of-three channel logic. If in the 2 of 4 logic system of the reactor protective system one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1 of 2. At rated power, the minimum operable high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are operable, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

All engineered safety features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-~~5~~ 2 basis.

The containment radiation high signal isolates the containment pressure relief, air sample and purge system valves.

References

(1) ~~FSAR~~, USAR Section 7.2.7.1

TABLE 2-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test Maintenance and Inoperable Bypass
1	Containment Isolation				
A	Manual	1	None	None	N/A
B	Containment High Pressure A	2[a][e]	1	During Leak Test	(f)
	B	2[a][e]	1		
	Pressurizer Low/Low A	2[a][e]	1	Reactor Coolant Pressure Less Than 1700 psia[b]	(f)
	B	2[a][e]	1		
2	Steam Generator Isolation				
A	Manual	1	None	None	N/A
B	Steam Generator Isolation	1	None	None	N/A
	(i) Steam Generator Low Pressure A	2/Steam Gen[e]	1/Steam Gen	Steam Generator Pressure Less Than 550 psia[c]	(f)
	B	2/Steam Gen[e]	1/Steam Gen		
	(ii) Containment High Pressure A	2[a][e]	1	During Leak Test	(f)
	B	2[a][e]	1		
3	Ventilation Isolation				
A	Manual	1	None	None	N/A
B	Containment High Radiation A	2[d]	None	If Containment ~~Ventilation~~ Relief and Purge ~~Isolation~~ Valves Are Closed	~~(f)~~ N/A
	B	2[d]	None		

a A and B circuits each have 4 channels.

b Auto removal of bypass above 1700 psia.

c Auto removal of bypass above 550 psia.

TABLE 2-4
(Continued)

d A and B circuits are both actuated by ~~any~~ either one of the ~~five VIAS~~ two initiating channels; ~~RM-050, RM-051, RM-060, RM-061, or RM-062; however, only RM-050 and RM-051~~ are required for containment ventilation isolation.

e If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition within eight hours from the time of discovery of loss of operability. The remaining inoperable channel may be bypassed for 48 hours from the time of discovery of loss of operability and, if an inoperable channel is not returned to operable status within this time frame, a unit shutdown must be initiated (see Specification 2.15(2)).

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from the time of discovery of loss of operability. If bypassed and that channel is not returned to operable status within 48 hours from the time of discovery of loss of operability, that channel must be placed in the tripped condition within the following eight hours. (See Specification 2.15(1) and exception associated with maintenance.)

3.0 SURVEILLANCE REQUIREMENTS

BASIS

Specifications 3.0.1 through 3.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal ~~Requirements~~ Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting condition of operation will be met."

Specification 3.0.1 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillance that are not performed during refueling outages. The limitation of Specification 3.0.1 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

The provisions of Specification 3.0.2 define the surveillance intervals for use in the Technical Specifications. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. A few surveillance requirements have uncommon intervals. ~~for example, Table 3-9 requires sampling of fish once per season.~~ In such a case the surveillance interval shall be performed as defined by the individual specifications.

Specification 3.0.3 extends the testing interval required by codes and standards referenced by the Technical Specifications. This clarification is provided to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities. Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the codes and standards referenced therein.

Specification 3.0.4 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, as defined by the provisions of Specifications 3.0.1 and 3.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for the corresponding Limiting Condition for Operation. Under the provisions

TABLE 3-2 (continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
4. Containment Pressure High Signal	a. Calibrate	R	a. Known pressure applied to sensors and CPHS actuation logic verified.
	b. Test	M	b. Pressure switch operation simulated one circuit at a time.
5. Containment Spray Logic	a. Test	M	a. Simulation of PPLS and CHPS 2/4 logic using built-in testing system. Both "standby power" and "no standby power" circuits will be tested for A and B channels. Test will verify functioning of initiation circuits of all equipment normally operated by safety feature actuation signals.
	b. Test	R	b. Complete automatic test initiated sensor operation (Item 1(b) and 4(b)) and including all normal automatic operations.
6. Containment Radiation High Signal(1)	a. Check	D	a. ~~Verify readings are below alarm initiation radiation level.~~ Normal readings observed and internal test signals used to verify instrument operation.

(1) CRHS monitors are the containment atmosphere gaseous radiation monitor and the Auxiliary Building Exhaust Stack gaseous radiation monitor.

TABLE 3-2 (continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF
ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROL

	Channel Description	Surveillance Function	Frequency	Surveillance Method
6.	(continued)	b. ~~Calibrate~~	~~R~~	~~b. Exposure to known external radiation source.~~
		c. ~~Test~~ Attachment 5	~~M~~	~~c. Remote operated integral radiation check source used to verify instrumentation, one channel at a time, and isolation lockout relay functional check.~~
7.	Manual Safety Injection Initiation	a. Test	R	a. Manual initiation.
8.	Manual Containment Isolation Initiation	a. Test	R	a. Manual initiation.
		b. Check	R	b. Observe isolation valves closure.
9.	Manual Initiation Containment Spray	a. Test	R	a. Manual switch operation; pumps and valves tested separately.
10.	Automatic Load Sequencers	a. Test	Q	a. Proper operation will be verified during safety feature actuation test of Item 3(a) above.
11.	Diesel Testing	See Technical Specification 3.7		

Attachment 5

6.	(continued)	b. Test	M	b.	Detector exposed to remote operated radiation check source or test signal to verify instrumentation, one channel at a time, and isolation lockout relay functional check.
		c. Calibrate	R	c.	Secondary and Electronic Calibration performed at refueling frequency. Primary calibration performed with exposure to radioactive sources only when required by the secondary and electronic calibration.

TABLE 3-3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING
OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Primary CEA Position Indication System	a. Check	S	a. Comparison of output data with secondary CEAPIS.
	b. Test	M	b. Test of power dependent insertion limits, deviation, and sequence monitoring systems.
	c. Calibrate	R	c. Physically measured CEDM position used to verify system accuracy. Calibrate CEA position interlocks.
2. Secondary CEA Position Indication System	a. Check	S	a. Comparison of output data with primary CEAPIS.
	b. Test	M	b. Test of power dependent insertion limit, deviation, out-of-sequence, and overlap monitoring systems.
	c. Calibrate	R	c. Calibrate secondary CEA position indication system and CEA interlock alarms.
3. Area, Process, and Post-Accident Radiation Monitors Except Effluent Radiation Monitors(1)	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
	b. Test	M	b. Detector exposed to remote operated radiation check source or test signal.
	c. Calibrate	R	c. RM-063L, M, and H and RM-064 - One time factory calibration is acceptable provided linearity solid sources are used to check the integrity of the detectors. RM-091A and B - In situ calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. In situ calibration for at least one decade below 10 R/hr shall be by means of calibrated radiation source. All other monitors - Exposure to known radiation source.

Attachment 6A

(1) The surveillance requirements for effluent radiation monitors are described under Specification 3.12.1. Effluent radiation monitors are: RM-041, RM-042, RM-043, RM-054A, RM-054B, RM-055, RM-055A, RM-057, RM-060, RM-061, and RM-062. RM-050 and RM-051 are considered effluent radiation monitors when monitoring the Auxiliary Building Exhaust Stack.

TABLE 3-3 (continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING
OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency	Surveillance Method
4.	Emergency Plan Radiation Instruments	a. Calibrate	A	a. Exposure to known radiation source.
		b. Test	M	b. Battery check.
~~5.~~	~~Environmental Monitors~~ Attachment 6B	~~a. Check~~	~~M~~	~~a. Operational Check.~~
		~~b. Calibrate~~	~~A~~	~~b. Verify airflow indicator.~~
6.	Pressurizer Level Instruments	a. Check	S	a. Comparison of independent level readings.
		b. Calibrate	R	b. Known differential pressure applied to sensor.
		c. Test	M	c. Signal to alarm meter relay adjusted with test device to verify setting.
7.	CEA Drive System Interlocks	a. Test	R	a. Verify proper operation of all CEDM system interlocks, using simulated signals where necessary.
		b. Test	P	b. If haven't been checked for three months and plant is shutdown.

Attachment 6A

3.	Area and Post Accident Radiation Monitors[1]	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
		b. Test	M	b. Detector exposed to remote operated radiation check source or test signal.
		c. Calibrate	R	c. Secondary and Electronic calibration performed at refueling frequency. Primary calibration with exposure to radioactive sources only when required by the secondary and electronic calibration. RM-091 A/B - Calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. Calibration for at least one decade below 1- R/hr. shall be by means of calibrated radiation source.

[1]Post Accident Radiation Monitors are: RM-063L/M/H, RM-064, and RM-091A/B. Area Radiation Monitors are: RM-070 thru RM-082, RM-084 thru RM-089, and RM-095 thru RM-098.

Attachment 6B

5.	Primary to Secondary Leak-Rate Detection Radiation Monitors (RM-054A/B, RM-057)	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
		b. Test	M	b. Detector exposed to remote operated radiation check sources or test signal.
		c. Calibrate	R	c. Secondary and Electronic calibration performed at refueling frequency. Primary Calibration performed with exposure to radioactive sources only when required by the secondary and electronic calibration.

TABLE 3-4 (Continued)

MINIMUM FREQUENCIES FOR SAMPLING TEST

	Type of Measurement and Analysis	
1. Reactor Coolant (Continued)		
(c) Cold Shutdown (Operating Mode 4)	(1) Chloride	1 per 3 days
(d) Refueling Shutdown (Operating Mode 5)	(1) Chloride (2) Boron Concentration	1 per 3 days (3) 1 per 3 days (3)
(e) Refueling Operation	(1) Chloride (2) Boron Concentration	1 per 3 days (3) 1 per shift (3)
2. SIRW Tank	Boron Concentration	1 per 31 days
3. Concentrated Boric Acid Tanks	Boron Concentration	1 per 31 days
4. SI Tanks	Boron Concentration	1 per 31 days
5. Spent Fuel Pool	Boron Concentration	1 per 31 days
6. Steam Generator Blowdown (Operating Modes 1 and 2)	Isotopic Analysis for Dose Equivalent I-131	1 per 7 days (4)

(1) Until the radioactivity of the reactor coolant is restored to $\leq 1 \mu$ Ci/gm DOSE EQUIVALENT I-131.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was subcritical for 48 hours or longer.

(3) Boron and Chloride sampling/analyses are not required when the core has been off-loaded. Reinitiate boron and chloride sampling/analyses one shift prior to reloading fuel into the cavity to assure adequate shutdown margin and allowable chloride levels are met.

(4) When Steam Generator Dose Equivalent I-131 exceeds 50 percent of the limits in Specification 2.20, the sampling and analysis frequency shall be increased to a minimum of 5 times per week. When Steam Generator Dose Equivalent I-131 exceeds 75 percent of this limit, the sampling and analysis frequency shall be increased to a minimum of once per day.

3.0 SURVEILLANCE REQUIREMENTS

3.10 Reactor Core Parameters (Continued)

(6) Azimuthal Power Tilt (Tq)

Whenever the core power is above 70% of rated power, the azimuthal power tilt shall be determined to be within its limits by calculating the tilt at least once every day using either:

a. The excore detectors with at least four safety channels operable, or

b. The incore detectors with at least two strings of three rhodium detectors per full core height quadrant operable.

(7) DNB Parameters

a. The cold leg temperature, pressurizer pressure, and axial shape index shall be verified to be within the limits of Section 2.10.4(5) at least once per shift.

b. The reactor vessel coolant total flow rate shall be determined to be within its limit by measurement at least once per month.

(Next Page is 3-69)

3.0 SURVEILLANCE REQUIREMENTS

3.11 Radiological Environmental Monitoring Program

Applicability

Applies to radiological monitoring of plant environs.

Objective

To establish a radiological monitoring program adequate to measure changes in the levels of environmental radioactivity due to plant effluents.

Specifications

(1) The radiological environmental monitoring program shall be conducted according to Table 3.9. Additional details of the radiological environmental monitoring program are in the ODCM. No changes shall be made to the ODCM which might reduce the effectiveness of the program. Analytical results of this program and deviations from the sampling schedule shall be reported to the Commission pursuant to Specification 5.9.4.b.

(2) If the level of radioactivity in an environmental sampling medium exceeds the reporting level specified in the ODCM, a non-routine report shall be prepared and submitted to the Commission pursuant to Specification 5.9.4.b.2.

(3) A land use survey shall be conducted once per 24 months between the dates of June 1 and October 1. This survey shall identify the location of the nearest milk animal and the nearest residence in each of the 16 cardinal sectors within a distance of five miles. The results of the land use survey shall be submitted to the Commission pursuant to Specification 5.9.4.b. The survey shall be conducted under the following conditions:

a. Within a one-mile radius from the plant site, enumeration by door-to-door or equivalent counting technique.

b. Within a five-mile radius, enumeration by using referenced information from county agricultural agents or other reliable sources.

If it is learned from this survey that milk animals are present at a location which yields a calculated thyroid dose greater than from previously sampled animals, the new location shall be added to the monitoring program. The sampling location having the lowest calculated dose may then be dropped from the monitoring program at the end of the grazing season during which the survey was conducted. Also, any location(s) from which milk can no longer be obtained may be dropped and replaced if practicable from the monitoring program and the Commission shall be notified pursuant to Specification 5.9.4.b.

3.0 SURVEILLANCE REQUIREMENTS

3.11 Radiological Environmental Monitoring Program (Continued)

(4) Analyses shall be performed on radioactive materials as part of an Interlaboratory Comparison Program that has been approved by the NRC. The results of these analyses shall be included in the Annual Radiological Environmental Operating Report.

Basis

The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals attributable to the operation of Fort Calhoun Station.

The specification for land use survey is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental media are performed in order to demonstrate the validity of results.

TABLE 3-9

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Collection Site[1]	Types of Analysis[2]	Frequency
1. Direct Radiation	a. Ten TLD indicator stations One (1) control station, total of eleven (11).	Gamma Isotopic	Quarterly
	b. An inner-ring of sixteen (16) stations, one in each meteorological sector in the general area of the site boundary and within 2.5 miles.[3]	Gamma dose during Site Area and General Emergencies only.	Replaced Annually
	c. An outer-ring of sixteen (16) stations, one in each meteorological sector located outside of the inner ring but no more distant than approximately 5 miles.[3]	Gamma dose during Site Area and General Emergencies only.	Replaced Annually
2. Air Monitoring	a. Indicator Stations 1. Three (3) Stations in the general area of the Site Boundary 2. City of Blair b. One (1) background station	1. Filter for Gross Beta[4] 2. Charcoal for I-131 3. Filter for Gamma Isotopic	1. Weekly 2. Weekly 3. Quarterly Composite of weekly filters
3. Water	a. Missouri River at nearest downstream drinking water intake. b. Missouri River downstream near the mixing zone. c. Missouri River upstream of plant intake (background).	Gamma Isotopic, H-3	1. Monthly composite for Gamma Isotopic Analysis 2. Quarterly Composite for H-3 Analysis

TABLE 3-9 (Continued)

Radiological Environmental Monitoring Program (Continued)

Exposure Pathway and/or Sample	Collection Site[1]	Types of Analysis[2]	Frequency
4. Milk[5]	a. Nearest family cow when available, or one (1) Dairy farm within 8 kilometers. b. One (1) Dairy farm between 8 kilometers and 30 kilometers (Background).	Gamma Isotopic and I-131	1. Semimonthly grazing season (May to October)
5. Fish	a. Four fish samples within vicinity of plant discharge b. One (1) background sample upstream of plant discharge.	Gamma Isotopic	Once per season (May to October)
6. Sediment	One sample from downstream area on the Station side of the Missouri River.	Gamma Isotopic	Semi-annually

[1] See Table 10 of the ODCM.

[2] The lower limit of detection (LLD) for analysis is defined in the ODCM in accordance with the wording of NUREG-0472, Rev. 2.

[3] Details of the Emergency TLD stations are contained in Emergency Preparedness Implementing Procedures.

[4] When a gross beta count indicates radioactivity greater than 1E-12 μCi/ml or 1 pCi/m ^{3}, a gamma spectral analysis will be performed.

[5] When milk samples are not available, a broad leaf vegetation sample shall be collected monthly when available.

Attachment 7

3.0 SURVEILLANCE REQUIREMENTS
3.12 Radiological Waste Sampling and Monitoring
3.12.1 Liquid and Gaseous Effluents

Applicability

Applies to the sampling, monitoring, and testing used for liquid and gaseous effluents.

Objective

To ensure that radioactive liquid and gaseous releases from the facility are maintained as low as reasonably achievable and within the limits specified by Specifications 2.9.1(1) and 2.9.1(2).

Specifications

(1) Liquid Effluents

a. Radioactive liquid waste sampling and activity analyses shall be performed in accordance with Table 3-11. The results of these analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is limited to the values in Specification 2.9.1(1)a.

b. Prior to release of each batch of liquid effluent, the batch shall be mixed, sampled, and analyzed for principal gamma emitters. When operational or other limitations preclude specific gamma radionuclide analysis of each batch, gross radioactivity measurements shall be made to estimate the quantity and concentrations of radioactive materials released in the batch, and a weekly sample composited from proportional aliquots from each batch released during the week shall be analyzed for the principal gamma-emitting radionuclides.

c. The overboard header radiation monitor shall have a:

(i) Source check prior to any release of radioactive materials from the monitor or the hotel waste tanks.

(ii) Quarterly channel functional test.

(iii) Channel calibration at refueling frequency.

d. The steam generator blowdown radiation monitors shall have:

(i) Daily channel checks.

(ii) Monthly source checks.

Attachment 7

3.0 SURVEILLANCE REQUIREMENTS

3.12 Radioactive Waste Disposal System

Applicability

Applies to the instrumentation used to determine hydrogen and oxygen concentrations in the waste gas decay tanks.

Objective

To ensure the concentrations of hydrogen and oxygen in the gaseous radioactive waste system are maintained below their flammability concentrations as required by Specification 2.9.

Specifications

The hydrogen and oxygen monitoring system for the waste gas decay tanks shall have a:

a. daily channel check (when in service)
b. monthly cross comparison with a grab sample
c. quarterly channel calibration using a gas mixture with concentrations in the range of interest

Basis

The specification ensures that instrumentation used to determine the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained in an operable condition. Maintaining the instrumentation used to determine hydrogen and oxygen concentration with a surveillance program 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 50.

(Next Page is 3-76)

(iii) Quarterly channel functional tests.

(iv) Channel calibration at refueling frequency.

e. The steam generator blowdown effluent flow rate will be calibrated at refueling frequency and visually determined operable daily.

f. Records shall be maintained of the radioactive concentrations and volume before dilution of each batch of liquid effluent released and of the average dilution flow and length of time over which each discharge occurred. Analytical results shall be submitted to the Commission in accordance with Section 5.9.4.a of these specifications.

(2) Gaseous Effluents

a. Radioactive gaseous waste sampling and activity analyses shall be performed in accordance with Table 3-12. The results of these analyses shall be used with the calculational methods in the ODCM to assure that the concentration of radioactive materials in unrestricted areas is limited to the values in Specification 2.9.1(2)a.

b. (i) An Auxiliary Building Exhaust Stack monitor shall have a source check prior to any release of radioactive materials from a gas decay tank or the containment. A monthly source check will be performed during refueling outages if a purge or gas decay tank release is not done during that month.

(ii) The Auxiliary Building Exhaust Stack gaseous, particulate, and iodine monitors and the Laboratory and Radioactive Waste Processing Building Exhaust Stack gaseous, particulate, and iodine monitors shall have a quarterly channel functional test.

(iii) The Auxiliary Building Exhaust Stack gaseous, particulate, and iodine monitors and the Laboratory and Radioactive Waste Processing Building Exhaust Stack gaseous, particulate, and iodine monitors shall be calibrated at refueling frequency.

(iv) The Auxiliary Building Exhaust and the Laboratory and Radioactive Waste Processing Building Exhaust stack flow rates will be calibrated and functionally tested at refueling frequency. The Auxiliary Building Exhaust and the Laboratory and Radioactive Waste Processing Building Exhaust stack radiation monitors flow rates will be calibrated and functionally tested at refueling frequency. The stack flow rates and radiation monitor flow rates will be determined operable by visual inspection daily.

(v) The Laboratory and Radioactive Waste Processing Building Exhaust Stack gaseous, particulate, and iodine activity monitors shall have a daily channel check and a monthly source check.

c. The condenser air ejector monitor shall have a:

(i) Daily channel check.

(ii) Monthly source check.

(iii) Quarterly channel functional test.

(iv) Channel calibration at refueling frequency.

d. The hydrogen and oxygen monitoring system for the gas decay tanks shall have a:

(i) Daily channel check (when in service).

(ii) Monthly cross comparison with a grab sample.

(iii) Quarterly channel calibration using gas mixtures with concentrations in the range of interest.

e. Records shall be maintained and reports of the sampling and results of analyses shall be submitted to the Commission in accordance with Section 5.9.4.a of these specifications.

Basis

The surveillance requirements given under Specification 3.12.1(2) provide assurance that radioactive gaseous effluents from the station are properly controlled and monitored over the life of the station in conformance with the requirements of General Design Criteria 60 and 64 of 10 CFR Part 50, Appendix A. These surveillance requirements provide the data for the licensee and the Commission to evaluate the performance of the station relative to radioactive gaseous wastes released to the environment. The existing minimum sensitivity of airborne effluent monitor RM-062 is 5E-06 mCi/cc/100 cpm and this minimum sensitivity shall be maintained if the monitor is replaced. Reports on the quantities of the radioactive materials released in gaseous effluents shall be furnished to the Commission on the basis of Section 5.9.4.a of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

The surveillance requirements given under Specification 3.12.1(1) provide assurance that liquid wastes are properly controlled and monitored in conformance with the requirements of General Design Criteria 60 and 64 of 10 CFR Part 50, Appendix A, during any planned release of radioactive materials in liquid effluents. These surveillance requirements provide the data for the licensee and the Commission to evaluate the station's performance relative to radioactive liquid wastes released to the environment. Reports on the quantities of radioactive materials released in liquid effluents shall be furnished to the Commission on the basis of Section 5.9.4.a of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

3.0 SURVEILLANCE REQUIREMENTS

3.12 Radiological Waste Sampling and Monitoring (Continued)

3.12.2 Solid Radioactive Waste

Applicability

Applies to the sampling, testing, and analysis of the wet radioactive waste.

Objective

To ensure that the solid radioactive wastes meet the limits specified in Section 2.9.2 of these Specifications.

Specifications

(1) The Process Control Program (PCP) shall be used to verify the solidification of at least one representative test specimen (drum) from at least every twelfth batch of wet radioactive waste (e.g., evaporator concentrates).

A. If any test specimen fails to verify solidification, the following actions shall be taken:

(i) Verify solidification of all other drums from the batch under test.

(ii) Review the adequacy of the solidification parameters defined in the PCP and develop/verify alternative solidification parameters, if required, in accordance with the PCP.

In the event the solidification parameters are altered:

(a) Select one representative drum from each consecutive batch to verify solidification until at least 3 consecutive drums verify solidification. The surveillance schedule defined in Specification 3.12.2(1), above, may be resumed after 3 consecutive drums verify solidification.

(b) Modify the PCP as required and report the changes to the NRC in accordance with Specification 5.9.4.a.

Basis

This specification was developed to ensure the requirements of 10 CFR Parts 20 and 71 for solid radioactive waste are met. The purpose of placing wet radioactive wastes in a solid, dry form is to limit dispersion of radioactive material to the environs in the event of failure of a disposal container (drum) before, during or after disposal. These requirements provide periodic documentation that solidified wet radioactive waste materials are in suitable form for transportation and disposal.

TABLE 3-11

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS

A. Monitor & Hotel Waste Tanks Releases

Sampling Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (4) (µCi/ml)
Each Batch	Principal Gamma Emitters(2)(5)	5.0 E-07
	I-131(2)	1.0 E-06
Monthly From One Batch	Dissolved Noble Gases (2) (Gamma Emitters)	1.0 E-05
Monthly Composite(1)	H-3	1.0 E-05
	Gross α	1.0 E-07
Quarterly Composite(1)	Sr-89, Sr-90	5.0 E-08

B. Steam Generator Blowdown

Sampling Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (4) (µCi/ml)
Weekly Composite(1)	Principal Gamma Emitters(5)	5.0 E-07
	I-131(6)	1.0 E-06
Weekly (3)	Dose Equivalent I-131	1.0 E-06
Monthly	Dissolved Noble Gases (Gamma Emitters)	1.0 E-05
Monthly Composite(1)	H-3	1.0 E-05
	Gross α	1.0 E-07
Quarterly Composite(1)	Sr-89, Sr-90	5.0 E-08

NOTES:

(1) To be representative of the average quantities and concentrations of radioactive materials in liquid effluents, samples should be collected in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite should be mixed in order for the composite sample to be representative of the average effluent release.

TABLE 3-11

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS (Continued)

NOTES:

(2) Or gross radioactivity as described in Specification 3.12.1(1)b.

(3) When steam generator iodine activity exceeds 50 percent of limits in Specification 2.20, the sampling and analysis frequency shall be increased to a minimum of five times per week. When the steam generator iodine activity exceeds 75 percent of this limit, the sampling and analysis frequency shall be increased to a minimum of once per day.

(4) The lower limit of detection (LLD) is defined in the ODCM based on NUREG 0472, Rev. 3.

(5) 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.

(6) A weekly grab sample and analyses program including gamma isotopic identification will be initiated for the turbine building sump effluent when the steam generator blowdown water composite analysis indicates the I-131 concentration is greater than 1.0 E-06 microcurie/milliliter.

TABLE 3-12

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS

Gaseous Source	Sampling and Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (4) (μCi/ml)
A. Gas Decay Tank Releases	Prior to each release	Principal Gamma(5) Emitters	1.0 E-04(1)
B. Containment Purge Releases or Containment Pressure Relief Line Releases	Prior to each release	Principal Gamma(5) Emitters	1.0 E-04(1)
	Prior to each release	H-3	1.0 E-06
C. Condenser Air Ejector Releases	Monthly (3)	Tritium (H-3)	1.0 E-06
	Monthly	Principal Gamma(5) Emitters	1.0 E-04(1)
D. Continuous(2) Auxiliary Building and Laboratory & Radioactive Waste Processing Building Exhaust Stack Releases	Weekly (Charcoal Sample)	I-131	1.0 E-12
	Weekly (2) (Particulates)	Principal Gamma(5) Emitters I-131 & Particulates with half-lives greater than 8 days	1.0 E-11
	Monthly Composite	Gross α	1.0 E-11
	Quarterly Composite (Particulates)	Sr-89, Sr-90	1.0 E-11

NOTES:

(1) For certain mixtures of gamma emitters, it may not be possible to measure radionuclides at levels near their sensitivity limits when other nuclides are present in the sample at much higher levels. Under these circumstances, it will be more appropriate to calculate the levels of such radionuclides using observed ratios with those radionuclides which are measurable.

TABLE 3-12

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS (Continued)

NOTES:

(2) To be representative of the average quantities and concentrations of radioactive material in particulate form released in gaseous effluents, sample should be collected in proportion to the design flow rate of the effluent stream and the design flow rate will be used in estimating releases.

(3) Required only when steam generator blowdown radioactivity for tritium (Table 3-11, Section B) exceeds 3.0E-03 microcurie/milliliter.

(4) The Lower Limit of Detection (LLD) is defined in the ODCM based on NUREG 0472, Rev. 3.

(5) 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, 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, Ce-144 for particulate emissions.

5.9.3 Special Reports

Special reports shall be submitted to the Regional Administrator of the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.

b. Tendon surveillance, reference 3.5.

c. Containment structural tests, reference 3.5.

d. Special maintenance reports.

e. Containment leak rate tests, reference 3.5.

~~f. Radioactive effluent releases, reference 2.9.~~

f ~~g.~~ Materials radiation surveillance specimens reports, reference 3.3.

g ~~h.~~ Fire protection equipment outage, reference 2.19.

h ~~i.~~ Post-accident monitoring instrumentation, reference 2.21.

5.9.4 Unique Reporting Requirements

a. Radioactive Effluent Release Report

A report covering the operation of the Fort Calhoun Station during the previous six months shall be submitted within 60 days after January 1 and July 1 of each year per the requirements of 10 CFR 50.36a.

Attachment 8

The radioactive effluent release report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant as outlined in Regulatory Guide 1.21, Revision 1.

The radioactive effluent release report shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter as outlined in Regulatory Guide 1.21, Revision 1.

The radioactive effluent release report shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter as outlined in Regulatory Guide 1.21, Revision 1. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The meteorological conditions concurrent with the

5.9.4 Unique Reporting Requirements (Continued)

a. Radioactive Effluent Release Report (Continued)

releases of effluents shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

The radioactive effluent release report shall include any changes[1] to the Process Control Program (PCP) or to the Offsite Dose Calculation Manual (ODCM) made during the reporting period. A level of detail commensurate to the significance of the change will be provided.

b. Radiological Environmental Operating Reports

1. Annual Report

An annual report containing the data taken in the radiological environmental monitoring program, in accordance with the ODCM, for the previous calendar year of operation shall be submitted prior to May 1 of each year. The content of the report shall include:

(a) Summarized and tabulated results of the radiological environmental surveillance activities following the format of Regulatory Guide 4.8, Table 1. In the event that some results are not available, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

(b) Interpretations and statistical evaluation of the results, including an assessment of the observed impacts of the plant operation on the environment.

(c) The results of participation in the Interlaboratory Comparison Program.

(d) The results of land use survey required by Specification 3.11(3).

(e) The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 2.1.3. The following information shall be included:

(1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

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(1) These changes can be initiated either by the licensee (implementation: subject to review by the PRC) or by the Commission (implementation: subject to their applicability to the Fort Calhoun Station design, review by the PRC and followed by a review by the SARC).

5.9.4 Unique Reporting Requirements (Continued)

b. Radiological Environmental Operating Reports (Continued)

1. Annual Report (Continued)

(2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations;

(3) Purification system flow history starting 48 hours prior to the first sample in which the limit was exceeded;

(4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and

(5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

2. Non-Routine Report

If a confirmed measured radionuclide concentration in an environmental sampling medium average over any calendar quarter sampling period exceeds the reporting level referenced in Table 3-9, Footnote 2, and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Commission within 30 days from the end of the quarter.

The report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

Attachment 8

a. Semiannual Radioactive Effluent Release Report

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 III.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

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) Section IV.B.2, IV.B.3, and IV.C of the Appendix I to 10 CFR 50.

5.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the Plant Review Committee and the Safety Audit and Review Committee.

l. Records of Environmental Qualification of Electric Equipment pursuant to 10 CFR 50.49.

m. Records of the service lives of all hydraulic and mechanical snubbers which are covered under the provisions of Section 2.18 of the Technical Specifications, including the date at which the service life commences and associated installation and maintenance records.

n. Records of analyses required by the Radiological Environmental Monitoring Program.

5.10.3 A complete record of the analysis employed in the selection of any fuel assembly to be placed in Region 2 of the spent fuel racks will be retained as long as that bundle remains in Region 2 (reference Technical Specifications 2.8(12) and 4.8.4).

5.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

20.1601(a)(1)(2)(3) — and as an alternative method allowed under 20.1601(c)

5.0 ADMINISTRATIVE CONTROLS

5.11.1 In lieu of the "control device" ~~or "alarm signal"~~ required by paragraph ~~20.203(c)(2)~~ 20.1601(a)(1)(2)(3) of 10 CFR 20, and as an alternative method allowed under 20.1601(c) each high radiation area (as defined in § ~~20.202(b)(3)~~ 20.1601 of 10 CFR 20) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by required issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. Individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Supervisor-Radiation Protection in the Radiation Work Permit.

5.11.2 The requirements of 5.11.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr (~~Very~~ Restricted High Radiation Area). In addition, locked doors shall be provided to prevent unauthorized ~~enter~~ entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Supervisor-Radiation Protection with the following exception:

a. In lieu of the above, for accessible localized ~~Very~~ Restricted High Radiation areas located in large areas such as containment, where no lockable enclosure exists in the immediate vicinity ~~of the Very High Radiation area~~ to control access to the ~~Very~~ Restricted High Radiation area and no such enclosure can be readily constructed, then the ~~Very~~ Restricted High Radiation area shall be:

i. roped off such that an individual at the rope boundary is exposed to 1000 mrem/hr or less.
ii. conspicuously posted, and
iii. a flashing light shall be activated as a warning device.

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*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs

The following programs shall be established, implemented, and maintained.

5.16.1 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 individuals in unrestricted areas 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:

a. Limitations on the operability of radioactive liquid and gaseous radiation monitoring instrumentation including operability tests and setpoint determination in accordance with the methodology in the ODCM.

b. Limitations on the concentration of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR Part 20.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR Part 20 and with the methodology and parameters in the ODCM.

d. Limitations on the annual and quarterly doses or dose commitment to individuals in unrestricted areas from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

e. Determination of cumulative doses from radioactive effluents for the current calendar quarter and current calendar year in accordance with the ODCM on a quarterly basis.

f. 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 in plant effluents.

g. Limitations on the concentration resulting from radioactive material released in gaseous effluents to unrestricted areas conforming to 10 CFR Part 20.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

i. Limitations on the annual and quarterly doses beyond the site boundary from Iodine-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

5.16.2 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:

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

b. 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.

c. Participation in an 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.

5.17 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 5.10.2.o. 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 Part 20, 10 CFR Part 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability or effluent, dose, or setpoint calculations.

b. Shall become effective after review by the Plant Review Committee and the Manager - Fort Calhoun Station.

c. Temporary changes to the ODCM may be made in accordance with Technical Specification 5.8.3.

d. Shall be submitted to the Nuclear Regulatory 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.

5.0 ADMINISTRATIVE CONTROLS
5.18 Process Control Program (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 5.10.2.o. This documentation shall contain:

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

   2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after review and acceptance by the PRC and the approval of the Plant Manager.

c. Temporary changes to the PCP may be made in accordance with Technical Specification 5.8.3.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire PCP 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 PCP 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.

TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

Amendment No. ~~32~~,~~38~~,~~52~~,~~54~~,~~57~~,~~67~~,
~~80~~,~~84~~,86

TABLE OF CONTENTS (Continued)

Amendment No. ~~32,34,43,54,55,57,73,80,86,93,99~~,141

TECHNICAL SPECIFICATIONS - TABLES

TABLE OF CONTENTS

Azimuthal Power Tilt - T_q

Azimuthal Power Tilt shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that axial half (upper or lower) of the core divided by the average power of all quadrants in that axial half (upper or lower) of the core.

Unrodded Planer Radial Peaking Factor - F_{xy}

The Unrodded Planar Radial Peaking Factor is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding azimuthal tilt, T_q. The maximum F_{xy} limit is provided in the Core Operating Limits Report.

Unrodded Integrated Radial Peaking Factor - F_R

The Unrodded Integrated Radial Peaking Factor is the ratio of the peak pin power to the average pin power in an unrodded core, excluding azimuthal tilt, T_q. The maximum F_R limit is provided in the Core Operating Limits Report.

Fire Suppression Water System

The fire suppression water system consists of fire pumps and distribution piping with associated sectionalizing control or isolation valves. Such valves 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.

Process Control Program (PCP)

The document(s) that contains the current formulas, sampling, analyses, tests, 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 20, 61, 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

Dose Equivalent I-131

That concentration of I-131 (μCi/gm) 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. In other words,

DEFINITIONS

Dose Equivalent I-131 (μCi/gm) = μCi/gm of I-131

+ 0.0361 x μCi/gm of I-132

+ 0.270 x μCi/gm of I-133

+ 0.0169 x μCi/gm of I-134

+ 0.0838 x μCi/gm of I-135

$\bar{E}$ - Average Disintegration Energy

$\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (High Alarm) setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and Semiannual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

2.1.3 Reactor Coolant Radioactivity

Applicability

Applies to the radioactivity of the reactor coolant.

Objective

To ensure that the reactor coolant radioactivity is maintained at a level commensurate with the occupational and public safety.

Specification

(1) The radioactivity of the reactor coolant shall be limited to:

a. $\leq$ 1.0 μCi/gm DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gm

(2) With the radioactivity of the reactor coolant > 1.0 μCi/gm DOSE EQUIVALENT I-131 for more than 100 hours during one continuous time interval or exceeding 60 μCi/gm, be in at least HOT SHUTDOWN with T_{avg} < 536°F within 6 hours.

(3) With the radioactivity of the reactor coolant > 100/$\bar{E}$ μCi/gm, be in at least HOT SHUTDOWN with T_{avg} < 536°F within 6 hours.

(4) With the radioactivity of the reactor coolant > 1.0 μCi/gm DOSE EQUIVALENT I-131, perform the sampling and analysis requirements of items 1.(a)(2)(ii) and 1.(b)(2)(i) of Table 3-4 until the radioactivity of the reactor coolant is restored to within its limits. Data pursuant to Specification 5.9.4.b for the Annual Report shall be compiled as follows:

a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded.

b. Purification System flow history starting 48 hours prior to the first sample in which the limit was exceeded.

c. The time duration when the radioactivity of the reactor coolant exceeded 1.0 μCi/gm DOSE EQUIVALENT I-131.

d. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should contain the date and time of sampling and the radioiodine concentrations.

2.0 LIMITING CONDITIONS FOR OPERATION

2.8 Refueling Operations

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

Specifications

The following conditions shall be satisfied during any refueling operations:

(1) The equipment hatch and one door in the air lock shall be properly closed. In addition, all automatic containment isolation valves shall be operable or at least one valve in each line shall be closed.

(2) One containment atmosphere gaseous radiation monitor and one Auxiliary Building Exhaust Stack gaseous radiation monitor that initiate closure of the containment pressure relief, air sample, and purge system valves shall be tested and verified to be operable immediately prior to refueling operations. The two monitors shall employ one-out-of-two logic from separate contact outputs for VIAS.

(3) Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

(4) Whenever core geometry is being changed, neutron flux shall be continuously monitored by at least two source range neutron monitors, with each monitor providing continuous visual indication in the control room. When core geometry is not being changed, at least one source range neutron monitor shall be in service.

(5) At least one shutdown cooling pump and heat exchanger shall be in operation. However, the pump and heat exchanger may be removed from operation for up to one hour per 8 hour period during the performance of core alterations in the vicinity of the reactor coolant hot leg loops or during manipulation of a source.

(6) Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

(7) When irradiated fuel is being handled in the auxiliary building, the exhaust ventilation from the spent fuel pool area will be diverted through the charcoal filter.

(8) Prior to initial core loading and prior to refueling operations, a complete check out, including a load test, shall be conducted on fuel handling cranes that will be required during the refueling operation to handle spent fuel assemblies.

(9) A minimum of 23 feet of water above the top of the reactor core shall be maintained whenever irradiated fuel is being handled.

(10) Storage in Region 1 and Region 2 of the spent fuel racks shall be restricted to fuel assemblies having initial enrichment less or equal to 4.0 weight percent of U-235.

(11) Storage in Region 2 of the spent fuel racks shall be restricted to those assemblies whose parameters fall within the "acceptable" region of Figure 2-10.

If any of the above conditions are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions, and no operations that may change the reactivity of the core shall be made.

A spent fuel assembly may be transferred directly from the reactor core to the spent fuel pool Region 2 provided the independent verification of assembly burnups has been completed and the assembly burnup meets the acceptance criteria identified in Technical Specification Figure 2-10.

Movement of irradiated fuel from the reactor core shall not be initiated before the reactor core has been subcritical for a minimum of 72 hours if the reactor has been operated at power levels in excess of 2% rated power.

Bases

The equipment and general procedures to be utilized during refueling operations are discussed in the USAR. Detailed instructions, the above specifications, and the design of the fuel handling equipment incorporating built-in interlocks and safety features provide assurance that no

2.0 LIMITING CONDITIONS FOR OPERATION

2.9 Radioactive Waste Disposal System

Applicability

Applies to the transfer of waste gases to the waste gas decay tanks. The provisions of Technical Specification 2.0.1 for Limiting Condition for Operation are not applicable.

Objective

To ensure compliance with General Design Criterion 60 of Appendix A to 10 CFR 50.

Specification

(1) The concentration of hydrogen and oxygen in the waste gas decay tanks shall be limited to below flammability concentrations. With hydrogen and oxygen concentrations above flammability concentrations, restore the concentrations to below flammability limits within 48 hours.

(2) The hydrogen and oxygen monitors shall be monitoring the inservice gas decay tank during the transfer of waste gases to the waste gas decay tank. Whenever the monitors are inoperable, transfer of waste gases to a gas decay tank may continue provided grab samples are taken from the gas decay tank and analyzed:

a. Every eight hours during degassing operations, and
b. Daily during other operations.

Basis

Specification 2.9 ensures that the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits with a measurement program 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.

2.14 Engineered Safety Features System Initiation Instrumentation Settings

Applicability

Applies to the engineered safety features system initiation instrumentation settings.

Objective

To provide for automatic initiation of the engineered safety features in the event that principal process variable limits are exceeded.

Specifications

The engineered safety features system initiation instrumentation setting limits shall be as stated in Table 2-1.

Basis

(1) High Containment Pressure

The basis for the 5 psig setpoint for the high pressure signal is to establish a setting which would be exceeded quickly in the event of a DBA, cover a spectrum of break sizes, and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

High containment pressure initiates the steam generator isolation signal which will close the main steam isolation and bypass valves and the main feedwater isolation and bypass valves.

(2) Pressurizer Low Pressure

The pressurizer low pressure safety injection signal is a diverse signal to the high containment pressure safety injection signal. The 1600 psia setting includes an uncertainty of $\pm$ 22 psia and is the setting used in the safety analysis.[1]

(3) Containment High Radiation (Air Monitoring)

The containment radiation high signal can be initiated by a containment atmosphere gaseous radiation monitor or an Auxiliary Building Exhaust Stack gaseous radiation monitor.[2]

(3) Containment High Radiation (Air Monitoring) (Continued)

The setpoints for the isolation function will be calculated in accordance with the ODCM.

(4) Low Steam Generator Pressure

A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a $\pm$22 psi uncertainty and was the setting used in the safety analysis.[3]

Closure of the MSIVs (and the bypass valves, along with main feedwater isolation and bypass valves) is accomplished by the steam generator isolation signal which is a logical combination of low steam generator pressure or high containment pressure.

As part of the AFW actuation logic, a separate signal is provided to terminate flow to a steam generator upon sensing a low pressure in that steam generator if the other steam generator pressure is greater than the pressure setting. This is done to minimize the temperature reduction in the reactor coolant system in the event of a main steam-line break. The setting of 466.7 psia includes a +31.7 psi uncertainty; therefore, a setting of 435 psia was used in the safety analysis.

(5) SIRW Tank Low Level

Level switches are provided on the SIRW tank to actuate the valves in the safety injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of approximately 24 minutes following a safety injection signal. The switch-over point of 16 inches above tank bottom is set to prevent the pumps from running dry during the 10 seconds required to stroke the valves and to hold in reserve approximately 28,000 gallons of water of at least the refueling boron concentration. The FSAR loss of coolant accident analysis[4] assumed the recirculation started when the minimum usable volume of 283,000 gallons had been pumped from the tank.

(6) Low Steam Generator Water Level

As part of the AFW actuation logic, a signal is provided to initiate AFW flow to one or two steam generators upon sensing a low water level in the steam generator(s) if the absolute steam generator pressure criteria are satisfied. This function ensures adequate steam generator water level is maintained in the event of a failure to deliver main feedwater to either steam generator. The setting of 28.2% of wide range tap span includes a +13.2% uncertainty; therefore, a setting of 15% of wide range tap span was used in the safety analysis.

(7) High Steam Generator Delta Pressure

As part of the AFW logic, a high steam generator differential pressure signal is generated to provide AFW to the higher pressure steam generator with a concurrent low level signal if both steam generator pressures are less than 466.7 psia. If the differential pressure between steam generators is less than the setting, neither steam generator is supplied with AFW in the presence of a low level signal. The setting of 119.7 psid includes a -15.3 psi uncertainty; therefore, a setting of 135 psid was used in the AFW safety analysis.

References

(1) USAR, Section 14.1.3

(2) USAR, Section 7.3.2.6

(3) USAR, Section 14.12

(4) USAR, Section 14.15

(5) USAR, Section 7.4.6

(6) USAR, Section 7.5.2.5

(7) USAR, Section 14.4.1

(5) In the event that any of the following Emergency Auxiliary Feedwater Panel instrumentation or control circuits become inoperable, either restore the inoperable component(s) to operable status within seven days, or be in hot shutdown within the next twelve hours. This specification is applicable in Modes 1 and 2.

Steam Generator Level, Wide Range (AI-179)
Steam Generator Level, Narrow Range (AI-179)
Steam Generator Pressure (AI-179)
Pressurizer Pressure (AI-179)

Basis

During plant operation, the complete instrumentation systems will normally be in service. Reactor safety is provided by the reactor protection system, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operating with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels are out of service.

All reactor protection and almost all engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in engineered safeguards control system.

When one of the four channels is taken out of service for maintenance, the protective system logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),[1] which results in a one-out-of-three channel logic. If in the 2 of 4 logic system of the reactor protective system one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1 of 2. At rated power, the minimum operable high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are operable, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

All engineered safety features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-2 basis. The containment radiation high signal isolates the containment pressure relief, air sample and purge system valves.

References

(1) USAR, Section 7.2.7.1

TABLE 2-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Containment Isolation				
A	Manual	1	None	None	N/A
B	Containment High Pressure A	2(a)(e)	1	During Leak Test	(f)
	B	2(a)(e)	1		
	Pressurizer Low/Low A	2(a)(e)	1	Reactor Coolant Pressure Less Than 1700 psia(b)	(f)
	B	2(a)(e)	1		
2	Steam Generator Isolation				
A	Manual	1	None	None	N/A
B	Steam Generator Isolation	1	None	None	N/A
	(i) Steam Generator Low Pressure A	2/Steam Gen(e)	1/Steam Gen	Steam Generator Pressure Less Than 550 psia(c)	(f)
	B	2/Steam Gen(e)	1/Steam Gen		
	(ii) Containment High Pressure A	2(a)(e)	1	During Leak Test	(f)
	B	2(a)(e)	1		
3	Ventilation Isolation				
A	Manual	1	None	None	N/A
B	Containment High Radiation A	2(d)	None	If Containment Relief and Purge Valves Are Closed	N/A
	B	2(d)	None		

a A and B circuits each have 4 channels.
b Auto removal of bypass above 1700 psia.
c Auto removal of bypass above 500 psia.

TABLE 2-4
(Continued)

d A and B circuits are both actuated by either one of the two initiating channels.

e If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition within eight hours from the time of discovery of loss of operability. The remaining inoperable channel may be bypassed for 48 hours from the time of discovery of loss of operability and, if an inoperable channel is not returned to operable status within this time frame, a unit shutdown must be initiated (see Specification 2.15(2)).

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from the time of discovery of loss of operability. If bypassed and that channel is not returned to operable status within 48 hours from the time of discovery of loss of operability, that channel must be placed in the tripped condition within the following eight hours. (See Specification 2.15(1) and exception associated with maintenance.)

3.0 SURVEILLANCE REQUIREMENTS

BASIS

Specifications 3.0.1 through 3. establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting condition of operation will be met."

Specification 3.0.1 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillance that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillance that are not performed during refueling outages. The limitation of Specification 3.0.1 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

The provisions of Specification 3.0.2 define the surveillance intervals for use in the Technical Specifications. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. A few surveillance requirements have uncommon intervals. In such a case the surveillance interval shall be performed as defined by the individual specifications.

Specification 3.0.3 extends the testing interval required by codes and standards referenced by the Technical Specifications. This clarification is provided to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities. Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the codes and standards referenced therein.

Specification 3.0.4 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, as defined by the provisions of Specifications 3.0.1 and 3.0.2, as a condition that constitutes a failure to meet the OPERABILITY req ts for the corresponding Limiting Condition for Operation. Under the provisio

TABLE 3-2 (continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency	Surveillance Method
4.	Containment Pressure High Signal	a. Calibrate	R	a. Known pressure applied to sensors and CPHS actuation logic verified.
		b. Test	M	b. Pressure switch operation simulated one circuit at a time.
5.	Containment Spray Logic	a. Test	M	a. Simulation of FPLS and CHPS 2/4 logic using built-in testing system. Both "standby power" and "no standby power" circuits will be tested for A and B channels. Test will verify functioning of initiation circuits of all equipment normally operated by safety feature actuation signals.
		b. Test	R	b. Complete automatic test initiated sensor operation (Item 1(b) and 4(b)) and including all normal automatic operations.
6.	Containment Radiation High Signal[1]	a. Check	D	a. Normal readings observed and internal test signal used to verify instrument operation.

(1) CRHS monitors are the containment atmosphere gaseous radiation monitor and the Auxiliary Building Exhaust Stack gaseous radiation monitor.

TABLE 3-2 (continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency		Surveillance Method
6.	(continued)	b. Test	M	b.	Detector exposed to remote operated radiation check source or test signal to verify instrumentation, one channel at a time, and isolation lockout relay functional check.
		c. Calibrate	R	c.	Secondary and Electronic Calibration performed at refueling frequency. Primary calibration performed with exposure to radioactive sources only when required by the secondary and electronic calibration.
7.	Manual Safety Injection Initiation	a. Test	R	a.	Manual initiation.
8.	Manual Containment Isolation Initiation	a. Test	R	a.	Manual initiation.
		b. Check	R	b.	Observe isolation valves closure.
9.	Manual Initiation Containment Spray	a. Test	R	a.	Manual switch operation; pumps and valves tested separately.
10.	Automatic Load Sequencers	a. Test	Q	a.	Proper operation will be verified during safety feature actuation test of Item 3(a) above.
11.	Diesel Testing	See Technical Specification 3.7			

TABLE 3-3 (continued)
MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency	Surveillance Method
1.	Primary CEA Position Indication System	a. Check	S	a. Comparison of output data with secondary CEAPIS.
		b. Test	M	b. Test of power dependent insertion limits, deviation, and sequence monitoring systems.
		c. Calibrate	R	c. Physically measured CEDM position used to verify system accuracy. Calibrate CEA position interlocks.
2.	Secondary CEA Position Indication System	a. Check	S	a. Comparison of output data with primary CEAPIS.
		b. Test	M	b. Test of power dependent insertion limit, deviation, out-of-sequence, and overlap monitoring systems.
		c. Calibrate	R	c. Calibrate secondary CEA position indication system and CEA interlock alarms.
3.	Area and Post Accident Radiation Monitors[1]	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
		b. Test	M	b. Detector exposed to remote operated radiation check source or test signal.
		c. Calibrate	R	c. Secondary and Electronic calibration performed at refueling frequency. Primary calibration with exposure to radioactive sources only when required by the secondary and electronic calibration. RM-091 A/B - Calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. Calibration for at least one decade below 1- R/hr. shall be by means of calibrated radiation source.

[1]Post Accident Radiation Monitors are: RM-063L/M/H, RM-064, and RM-091A/B. Area Radiation Monitors are: RM-070 thru RM-082, RM-084 thru RM-089, and RM-095 thru RM-098.

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING
OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency	Surveillance Method
4.	Emergency Plan Radiation Instruments	a. Calibrate	A	a. Exposure to known radiation source.
		b. Test	M	b. Battery check.
5.	Primary to Secondary Leak-Rate Detection Radiation Monitors (RM-054A/B, RM-057)	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
		b. Test	M	b. Detector exposed to remote operated radiation check sources or test signal.
		c. Calibrate	R	c. Secondary and Electronic calibration performed at refueling frequency. Primary Calibration performed with exposure to radioactive sources only when required by the secondary and electronic calibration.
6.	Pressurizer Level Instruments	a. Check	S	a. Comparison of independent level readings.
		b. Calibrate	R	b. Known differential pressure applied to sensor.
		c. Test	M	c. Signal to alarm meter relay adjusted with test device to verify setting.
7.	CEA Drive System Interlocks	a. Test	R	a. Verify proper operation of all CEDM system interlocks, using simulated signals where necessary.

TABLE 3-4 (Continued)

MINIMUM FREQUENCIES FOR SAMPLING TESTS

		Type of Measurement and Analysis	
1.	Reactor Coolant (Continued)		
	(c) Cold Shutdown (Operating Mode 4)	(1) Chloride	1 per 3 days
	(d) Refueling Shutdown (Operating Mode 5)	(1) Chloride (2) Boron Concentration	1 per 3 days(3) 1 per 3 days(3)
	(e) Refueling Operation	(1) Chloride (2) Boron Concentration	1 per 3 days(3) 1 per shift(3)
2.	SIRW Tank	Boron Concentration	1 per 31 days
3.	Concentrated Boric Acid Tanks	Boron Concentration	1 per 31 days
4.	SI Tanks	Boron Concentration	1 per 31 days
5.	Spent Fuel Pool	Boron Concentration	1 per 31 days
6.	Steam Generator Blowdown (Operating Modes 1 and 2)	Isotopic Analysis for Dose Equivalent I-131	1 per 7 days(4)

(1) Until the radioactivity of the reactor coolant is restored to $\leq$ 1 μCi/gm DOSE EQUIVALENT I-131.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was subcritical for 48 hours or longer.

(3) Boron and chloride sampling/analyses are not required when the core has been off-loaded. Reinitiate boron and chloride sampling/analyses one shift prior to reloading fuel into the cavity to assure adequate shutdown margin and allowable chloride levels are met.

(4) When Steam Generator Dose Equivalent I-131 exceeds 50 percent of the limits in Specification 2.20, the sampling and analysis frequency shall be increased to a minimum of 5 times per week. When Steam Generator Dose Equivalent I-131 exceeds 75 percent of this limit, the sampling and analysis frequency shall be increased to a minimum of once per day.

(6) Azimuthal Power Tilt (Tq)

Whenever the core power is above 70% of rated power, the azimuthal power tilt shall be determined to be within its limits by calculating the tilt at least once every day using either:

a. The encore detectors with at least four safety channels operable, or

b. The incore detectors with at least two strings of three rhodium detectors per full core height quadrant operable.

(7) DNB Parameters

a. The cold leg temperature, pressurizer pressure, and axial shape index shall be verified to be within the limits of Section 2.10.4(5) at least once per shift.

b. The reactor vessel coolant total flow rate shall be determined to be within its limit by measurement at least once per month.

3.0 SURVEILLANCE REQUIREMENTS

3.12 Radioactive Waste Disposal System

Applicability

Applies to the instrumentation used to determine hydrogen and oxygen concentrations in the waste gas decay tanks.

Objective

To ensure the concentrations of hydrogen and oxygen in the gaseous radioactive waste system are maintained below their flammability concentrations as required by Specification 2.9.

Specifications

The hydrogen and oxygen monitoring system for the waste gas decay tanks shall have a:

a. daily channel check (when in service)
b. monthly cross comparison with a grab sample
c. quarterly channel calibration using a gas mixture with concentrations in the range of interest

Basis

The specification ensures that instrumentation used to determine the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained in an operable condition. Maintaining the instrumentation used to determine hydrogen and oxygen concentration with a surveillance program 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 50.

(Next Page is 3-76)

5.9.3 Special Reports

Special reports shall be submitted to the Regional Administrator of the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference 3.5.
c. Containment structural tests, reference 3.5.
d. Special maintenance reports.
e. Containment leak rate tests, reference 3.5.
f. Materials radiation surveillance specimens reports, reference 3.3.
g. Fire protection equipment outage, reference 2.19.
h. Post-accident monitoring instrumentation, reference 2.21

5.9.4 Unique Reporting Requirements

a. Semiannual Radioactive Effluent Release Report

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 outlines in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section III.B.1 of Appendix I to 10 CFR Part 50.

b. Annual Radiological Environmental Operating Report

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) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

5.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.
b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
c. Records of facility radiation and contamination surveys.
d. Records of radiation exposure for all individuals entering radiation control areas.
e. Records of gaseous and liquid radioactive material released to the environs.
f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.
g. Records of training and qualification for current members of the plant staff.
h. Records of in-service inspections performed pursuant to these Technical Specifications.
i. Records of Quality Assurance activities required by the QA Manual.
j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
k. Records of meetings of the Plant Review Committee and the Safety Audit and Review Committee.
l. Records of Environmental Qualification of Electric Equipment pursuant to 10 CFR 50.49.
m. Records of the service lives of all hydraulic and mechanical snubbers which are covered under the provisions of Section 2.18 of the Technical Specifications, including the date at which the service life commences and associated installation and maintenance records.
n. Records of analyses required by the Radiological Environmental Monitoring Program.
o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

5.10.3 A complete record of the analysis employed in the selection of any fuel assembly to be placed in Region 2 of the spent fuel racks will be retained as long as that bundle remains in Region 2 (reference Technical Specifications 2.8(12) and 4.8.4).

5.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

5.11.1 In lieu of the "control device" required by paragraph 20.1601(a)(1)(2)(3) of 10 CFR 20, and as an alternative method allowed under 20.1601(c) each high radiation area (as defined in § 20.1601 of 10 CFR 20) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by required issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established end personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Supervisor-Radiation Protection in the Radiation Work Permit.

5.11.2 The requirements of 5.11.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr (Restricted High Radiation Area). In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Supervisor-Radiation Protection with the following exception:

a. In lieu of the above, for accessible localized Restricted High Radiation areas located in large areas such as containment, where no lockable enclosure exists in the immediate vicinity to control access to the Restricted High Radiation area, and no such enclosure can be readily constructed, then the Restricted High Radiation area shall be:

   i. roped off such that an individual at the rope boundary is exposed to 1000 mrem/hr or less,
   ii conspicuously posted, and
   iii a flashing light shall be activated as a warning device.

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*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs

The following programs shall be established, implemented, and maintained.

5.16.1 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 individuals in unrestricted areas 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:

a. Limitations on the operability of radioactive liquid and gaseous radiation monitoring instrumentation including operability tests and setpoint determination in accordance with the methodology in the ODCM.

b. Limitations on the concentration of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR Part 20.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR Part 20 and with the methodology and parameters in the ODCM.

d. Limitations on the annual and quarterly doses or dose commitment to individuals in unrestricted areas from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

e. Determination of cumulative doses from radioactive effluents for the current calendar quarter and current calendar year in accordance with the ODCM on a quarterly basis.

f. 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 in plant effluents.

g. Limitations on the concentration resulting from radioactive material released in gaseous effluents to unrestricted areas conforming to 10 CFR Part 20.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

i. Limitations on the annual and quarterly doses beyond the site boundary from Iodine-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

5.16.2 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:

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

b. 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.

c. Participation in an 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.

5.0 ADMINISTRATIVE CONTROLS

5.17 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 5.10.2.o. 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 Part 20, 10 CFR Part 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability or effluent, dose, or setpoint calculations.

b. Shall become effective after review by the Plant Review Committee and the Manager - Fort Calhoun Station.

c. Temporary changes to the ODCM may be made in accordance with Technical Specification 5.8.3.

d. Shall be submitted to the Nuclear Regulatory 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.

5.18 Process Control Program (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 5.10.2.o. This documentation shall contain:

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

   2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after review and acceptance by the PRC and the approval of the Plant Manager.

c. Temporary changes to the PCP may be made in accordance with Technical Specification 5.8.3.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire PCP 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 PCP 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 B

DISCUSSION, JUSTIFICATION, AND NO SIGNIFICANT HAZARDS CONSIDERATIONS

DISCUSSION AND JUSTIFICATION

The Omaha Public Power District (OPPD) proposes to revise the Fort Calhoun Station Unit No. 1 Technical Specifications to implement Generic Letter 89-01 concerning the Radiological Effluent Technical Specifications (RETS), and to revise the requirements for the Containment Radiation High Signal following the guidance of NUREG-0133.

Proposed changes to RETS

As specified in Generic Letter 89-01, it is proposed to relocated RETS from the Technical Specifications to the Offsite Dose Calculation Manual (ODCM). The following is a description of the proposed changes:

1. The definitions for the ODCM and Process Control Program (PCP) were revised to agree with the definitions in Generic Letter 89-01. Definitions for purge-purging and venting were moved to the ODCM.

2. Technical Specifications 2.9 under Limiting Condition for Operations Section, and 3.11 and 3.12 under Surveillance Requirements Section were relocated from the Technical Specifications and incorporated into the ODCM.

3. Tables 3-2, 3-3, and 3-4 were revised to ensure that LCO requirements, which are currently controlled by Specification 3.12, were retained. The surveillance function and frequency for Containment Radiation High Signal, Area and Post Accident Radiation Monitors, and Primary-to-Secondary leak rate detection radiation monitors in Table 3-2 are updated for consistency. Sampling requirements for steam generator blowdown are being relocated from Table 3-11, which is being deleted, to Table 3-4. Operating modes are being added to the sampling of steam generator blowdown consistent with the actions required by Specification 2.20.

4. Tables 3-9, 3-11, and 3-12 were relocated from the Technical Specifications and incorporated into the ODCM

5. Specifications 2.9 and 3.12 are proposed to retain requirements in the Technical Specifications that are related to explosive gases. This was a specific requirement of the Generic Letter. The grab sample provisions are in agreement with the present Specification 2.9.1(2)d. A 48 hour Limiting Condition for Operation has been added for the time allowed for hydrogen and oxygen concentrations to be out of specification. This is consistent with the Standard RETS.

6. Reporting and records requirements were revised in Section 5.9.4 and Sections 5.16, 5.17, and 5.18 under Administrative Controls were added to the Technical Specifications to define administrative details of the ODCM and PCP. Records retention for RETS will be covered under existing Technical Specification 5.10.1.

Proposed Changes to the Containment Radiation High Signal

Consistent with NUREG-0133, it is proposed to remove the Ventilation Isolation Actuation Signal (VIAS) functions of the particulate and iodine monitors sampling the Auxiliary Building Exhaust Stack and Containment. The proposal would remove the VIAS functions of three effluent monitors. These are VIAS signals for Stack Iodine (RM-060), Stack Particulate (RM-061) and Containment Particulate (RM-050).

The particulate and iodine monitors will continue to sample effluents, but will not initiate VIAS. NUREG-0133 recommends the use of gas monitors for VIAS initiation because it is considered to be impractical to apply instantaneous alarm/trip setpoints to integrating radiation monitors sensitive to radioiodines or radioactive materials other than noble gases.

The gas monitors will continue to isolate releases so that 10 CFR Part 20 instantaneous limits and 10 CFR Part 50 annual limits will be complied with.

The following pages are being revised in the Technical Specifications to update the VIAS functions so that only gas monitors initiate VIAS; pages 2-37, 2-38, 2-61, 2-66a, 2-69, and 2-69a.

Changes to Continuous Sampling Requirements

Technical Specification 2.9.1(2)h and Table 3-12 currently require that continuous samples be taken during continuous releases. It is proposed to revise this requirement to provide 2 hours to initiate the auxiliary sampling required during continuous releases. The problems associated with meeting the requirements to continuously sample were reported to the NRC in LER's 91-028, and 92-001. As described in Regulatory Guide 1.21 Appendix A, the intent of continuous sampling is to identify radionuclides being released and to predict their concentrations. Two hours is proposed as a reasonable amount of time which will allow investigation of trouble alarms on equipment and to complete calibrations, filter changes and testing requirements without significantly affecting the results of the sampling. Current requirements allow periodic grab samples be taken when monitors are inoperable for releases which have a greater impact on the total effluents released from the plant. These changes have been included in the draft ODCM.

Administrative changes

a. The Table of Contents is being revised to reflect proposed changes.

b. Page 2-8 is being revised to correct a reference that was changed as a result of the Generic Letter changes.

c. Page 2-38 is being revised to delete a reference to the specific procedure which implements the independent verification of fuel burnup which is required to move fuel directly to Region 2 of the spent fuel pool. The requirements remain, only the name of the procedure is being deleted. Additionally, wording which reflects the CE Standard Technical Specifications are being added to clarify movement of irradiated fuel.

d. Reference 2 on Page 2-63 is being revised to reflect a Section of the Updated Safety Analysis Report (USAR) which better describes the Containment Radiation High Signal.

e. Table 2-4, Item 3B on page 2-69 is being revised to reflect the requirements of CE Standard Technical Specifications. As described in USAR Section 7.3.2.6 and Specification 2.8(2), the Containment Radiation High Signal (CRHS) isolates the containment pressure relief, air sample, and purge system valves. As currently written, it could be implied that all valves which receive a VIAS are required to be closed in order to place the CRHS channels in bypass. This change clarifies the requirements that only the containment vent and purge valves are required to be closed. A description of these valves is being added to the basis of Specification 2.15 on page 2-66a.

f. Page 2-91 is being revised to reflect a change in Section 5.9.3.

g. Page 3-0b is being revised to delete an example of a surveillance test table that will be relocated to the ODCM.

h. Page 3-63b is being revised to include a statement on the bottom of the page indicating that the next page will be page 3-69.

i. Page 5-19 is being revised to add records retention requirements for the ODCM and PCP.

NO SIGNIFICANT HAZARDS CONSIDERATIONS

The proposed changes do not involve a significant hazards consideration because operation of Fort Calhoun Station Unit No. 1, in accordance with these changes, would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The removal of the present Radiological Effluent Technical Specifications to the Offsite Dose Calculation Manual in accordance with the guidelines presented in Generic Letter 89-01 will not cause any increase in the probability or consequences of an accident. Only the procedural details have been transferred to the ODCM. Programmatic controls have been retained or added to ensure continued compliance with federal requirements.

   The removal of VIAS signals from the particulate and iodine radiation monitors will not cause an increase in the probability or consequences of an accident. Initial indications for radioactive release will occur promptly on the noble gas radiation monitors. VIAS will be initiated as soon as the Auxiliary Building Exhaust Stack or the Containment noble gas monitor reaches its alarm setpoint. The removal of the particulate and iodine monitors will not alter the initiation of VIAS in any postulated accident.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   The relocation of RETS from Technical Specifications to the ODCM is an administrative change. Present tests, calibrations, or inspections necessary to ensure the quality of systems and components will continue to be performed, and this change will not create a new or different kind of accident. The removal of the VIAS input from the particulate and iodine radiation monitors is the result of the recognition, by NRC documents, of the impracticality of applying instantaneous alarm setpoints to integrating radiation monitors. The primary and fastest indications of an actual radioactive release are noble gases. The radiation monitors for noble gases will continue to provide inputs for VIAS. Therefore, no new or different kind of accident been created.

3. Involve a significant reduction in the margin of safety.

   The administrative changes made will not cause a reduction in the margin of safety. The present RETS requirements are retained in the ODCM with Programmatic Controls in the Technical Specifications. This is consistent with the guidance of Generic Letter 89-01.

The removal of VIAS inputs from the particulate and iodine radiation monitors will not cause a significant reduction in the margin of safety. The primary indicators for radioactive releases, noble gases, will still be monitored and will initiate VIAS upon reaching their alarm setpoints. Spurious alarms, due to the incorrect application of instantaneous limits to integrating monitors will be eliminated, reducing the functional requirements that the Engineered Safeguard Features must comply with and enhance the reliability of VIAS. Noble gas concentrations are approximately 10,000 times greater than the particulate and iodine concentrations, therefore the margin of safety in detecting radioactive releases will be retained.

Therefore based on the above considerations, it is OPPD's position that this proposed amendment does not involve a significant hazards consideration as defined by 10 CFR 50.92, and the proposed changes will not result in a condition which significantly alters the impact of the station on the environment. The proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(e)(g) and, pursuant to 10 CFR 51.22(b) no environmental assessment need be prepared.

ATTACHMENT C

COMPARISON OF PARAGRAPH NUMBERS FROM TECH. SPECS TO ODCM

Tech. Spec.	ODCM
2.9.1 Objective	1.0
2.9.1.A.(1)	1.1.2
2.9.1.A.(2)	1.1.3
2.9.1.B.(1)	1.2.2
2.9.1.B.(2)	1.2.3
2.9.1.B.(3)	1.2.4
2.9.1.(1)a.(i)	1.1.1
2.9.1.(1)a.(ii)	1.1
2.9.1.(1)b.	4.1.1
2.9.1.(1)b.(i)	4.1.1.A
2.9.1.(1)b.(ii)	4.1.1.B
2.9.1.(1)b.(iii)	4.1.1.C
2.9.1.(1)c.	4.2.1
2.9.1.(1)c.(i)	4.2.1.A
2.9.1.(1)c.(ii)	4.2.1.B
2.9.1.(1)c.(iii)	4.2.1.C
.9.1.(1)d.	2.1.1
..9.1.(1)d.(i)	2.1.1.1
2.9.1.(1)d.(ii)	2.1.1.2
2.9.1.(1)d.(iii)	2.1.1.4
2.9.1.(1)d.(iii)1.	2.1.1.4.A
2.9.1.(1)d.(iii)2.	2.1.1.4.B
2.9.1.(1)d.(iii)(Notes)	2.1.1.6
	2.1.1.5
2.9.1.(1)e.1)	2.1.2.2
2.9.1.(1)e.2)	2.1.2.3
2.9.1.(1)e.	2.1.2.3.A
	2.1.2.3.B
	2.1.2.4
2.9.1.(2)a.(i)	1.2.1
2.9.1.(2)a.(ii)	1.2
2.9.1.(2)b.	4.2.1
2.9.1.(2)b.(i)	4.2.1.A
2.9.1.(2)b.(ii)	4.2.1.B
2.9.1.(2)b.(iii)	4.2.1.C
2.9.1.(2)c.	4.2.2
2.9.1.(2)c.(i)	4.2.2.A
2.9.1.(2)c.(ii)	4.2.2.B
2.9.1.(2)c.(iii)	4.2.2.C
2.9.1.(2)d.	Retained in Tech. Specs.
2.9.1.(2)e.	2.2.1.4
2.9.1.(2)e.1)	2.2.1.4.A
2.9.1.(2)e.2)	2.2.1.4.B
2.9.1.(2)f.	2.2.2.1
2.9.1.(2)g.	2.2.1.1
2.9.1.(2)g.(i)	2.2.1.1.A
2.9.1.(2)g.(ii)	2.2.1.1.C
2.9.1.(2)g.(iii)	2.2.1.1.B
2.9.1.(2)g.(iv)	2.2.1.1 D
	2.2.1.1.E

COMPARISON OF PARAGRAPH NUMBERS FROM TECH. SPECS TO ODCM
Page Two

Tech. Spec.	ODCM
2.9.1.(2)g.(v)	2.2.1.2.A/B
2.9.1.(2)h.(i)	2.2.3.1
2. ..(2)h.(i)	2.2.3.1.A
	2.2.3.1.B
2.9.1.(2)h.(ii)	2.2.3.1.C
3.11 Objective	5.0
3.11(1)	5.1.1
3.11(2)	5.1.3
3.11(3)	5.1.4
3.11(3)a.	5.1.4.A
3.11(3)b.	5.1.4.B
3.11(3)b.(Note)	5.1.4.B
3.11(4)	5.1.5
3.12.1 Objective	3.0
3.12.1(1)a.	3.1.1
3.12.1(1)b.	3.1.2
	3.1.2.1
	3.1.2.2
3.12.1(1)c.(i)(ii)(iii)	Table 3, Part A
3.12.1(1)d.(i)(ii)(iii)(iv)	Table 3, Part A
3.12.1(1)e.	Table 3, Note #4
3.12.1(1)f.	3.1.3
3.12.1(2)a.	3.2.1
3.12.1(2)b.(i)(ii)(iii)(iv)	Table 3, Part D
3.12.1(2)b.(v)	Table 3, Part B
3.12.1(2)c.(i)(ii)	Table 3, Part B
Table 3-11, Note (4)	3.3
Table 3-12, Note (4)	3.3
5.9.3.a.	4.3
5.9.4.b.1.	4.4
5.9.4.b.1.a.	4.4.1
5.9.4.b.1.b.	4.4.2
5.9.4.b.1.c.	4.4.3
5.9.4.b.1.d.	4.4.4
5.9.4.b.1.e.	4.4.5
5.9.4.b.1.e.(1)	4.4.5.1
5.9.4.b.1.(2)	4.4.5.1.A
5.9.4.b.1.(3)	4.4.5.1.B
5.9.4.b.1.(4)	4.4.5.1.C
5.9.4.b.1.(5)	4.4.5.1.D
5.9.4.b.2.	4.5