ATTACHMENT A

AMENDED TECHNICAL SPECIFICATIONS

TECHNICAL SPECIFICATIONS
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DEFINITIONS

PROTECTIVE SYSTEMS (Continued)

Engineered Safety Feature Logic[2]

The system which utilizes relay contact outputs from individual instrument channels to provide a dual channel signal to independently initiate the actuation of the engineered safety feature equipment. Two logic subsystems, termed A and B, are provided; each subsystem is composed of four channels wired to provide independent safety feature initiation signals on a 2-out-of-4 basis.

Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

INSTRUMENTATION SURVEILLANCE

Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during normal plant operation. This determination shall where feasible, include comparison of the channel with other independent channels measuring the same variable.

Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating action.

Channel Calibration

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

Source Check

Verification of channel response when the channel sensor is exposed to a radioactive source.

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 Planar 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.

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.

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.

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,

$$
\begin{aligned}
\text{Dose Equivalent I-131 } (\mu\text{Ci/gm}) = {} & \mu\text{Ci/gm of I-131} \\
& + 0.0361 \times \mu\text{Ci/gm of I-132} \\
& + 0.270 \times \mu\text{Ci/gm of I-133} \\
& + 0.0169 \times \mu\text{Ci/gm of I-134} \\
& + 0.0838 \times \mu\text{Ci/gm of I-135}
\end{aligned}
$$

$\bar{E}$ - 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)

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.

References

(1) USAR, Section 7.2

(2) USAR, Section 7.3

Amendment No. 67

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 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.

(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 mCi/ml total activity.

(ii) With the concentration of radioactive material released to unrestricted areas exceeding the abo e 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 reason 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 one 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, for unrestricted areas.

(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 cumulative dose contributions to each of the 16 cardinal sectors, 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 or 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 inservice 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. (i) The stack monitors for gaseous, particulate and iodine activities 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 ventilation stack gas or particulate monitor is inoperable, appropriate grab samples will be taken and analyzed once per eight (8) hours.

(ii) 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.

f. During release of gaseous radioactive wastes from the gaseous waste discharge header or during containment venting to the ventilation stack, the following conditions shall be met:

(i) The gas, iodine, and particulate monitors shall be monitoring the vent 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.

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

(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.

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, calculational 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 area. This specification provides assurance that no member of the general public will be exposed at any time to liquid 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.

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

Basis

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.

Specification 2.9.1(1)d, consistent with the requirements of General Design Criteria 60 and 64 of 10 CFR Part 50, 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 unrestricted 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 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 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, containment pressure relief line, and containment purge line are not made whenever the ventilation stack monitors are inoperable. This specification also assures that the gross radioactivity, during power operation, is monitored from the condenser air ejector discharge.

Specification 2.9.1(2)f 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.

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.

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

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

Each channel is 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 ±22 psi uncertainty and was the setting used in the safety analysis.[3]

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 steamline 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 switchover 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 at least 1700 ppm borated water. 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

TABLE 2-1

Engineered Safety Features System Initiation Instrument Setting Limits

Functional Unit	Channel	Setting Limit	
1. High Containment Pressure	a. Safety Injection b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode	≤ 5 psig	
2. Pressurizer Low/Low Pressure	a. Safety b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode	≥ 1600 psia (1)	
3. Containment High Radiation	Containment Ventilation Isolation	In accordance with the Offsite Dose Calculational Manual	
4. Low Steam Generator Pressure	a. Steam Line Isolation b. Auxiliary Feedwater Actuation	≥ 500 psia (2) ≥ 466.7 psia	
5. SIRW Low Level Switches	Recirculation Actuation	16 inches +0, -2 in. above tank bottom	
6. 4.16 KV Emergency Bus Low Voltage	a. Loss of Voltage	(2995.2 + 104) volts 20.8 ≤ 5.9(4) seconds	Trip
	b. Degraded Voltage i) Bus 1A3 Side	> 3825.52 volts (4.8 ± .5) seconds	Trip

TABLE 2-1 (Continued)

Engineered Safety Features System Initiation Instrument Setting Limits

Functional Unit	Channel	Setting Limit	
6. (Continued)	b. (Continued)		
	ii) Bus 1A4 Side	≥ 3724.08 volts (4.8 ± .5) seconds	Trip
7. Low Steam Generator Water Level	Auxiliary Feedwater Actuation	≥ 28.2% of wide range tap span	
8. High Steam Generator Delta Pressure	Auxiliary Feedwater Actuation	≤ 119.7 psid	

(1) May be bypassed below 1700 psia and is automatically reinstated above 1700 psia.
(2) May be bypassed below 550 psia and is automatically reinstated above 550 psia.
(3) Simultaneous high containment pressure and pressurizer low/low pressure.
(4) Applicable for bus voltage ≤ 2995.2 - 20.8 volts only. (For voltage ≥ (2995.2 - 20.8) volts, time delay shall be > 5.9 seconds.)

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

Channel Description	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	N	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 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.

(1) The surveillance requirements for effluent radiation monitors are described under Specification 3.12.1. Effluent radiation monitors are: 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 ventilation stack.

TABLE 3-4 (Continued)

MINIMUM FREQUENCIES FOR SAMPLING TEST

	Type of Measurement and Analysis	Sample and Analysis Frequency
1. Reactor Coolant (Continued)		
(c) Cold Shutdown	(1) Chloride	1 per 3 days
(d) Refueling Operation	(1) Chloride	1 per 3 days
	(2) Boron Concentration	1 per 3 days
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

(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.

3.0 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.

(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. Filter for Gross Beta [4]	1. Weekly
	1. Three (3) Stations in the general area of the Site Boundary	2. Charcoal for I-131	2. Weekly
		3. Filter for Gamma Isotopic	3. Quarterly composite of weekly filters
	2. City of Blair		
	b. One (1) background Station		
3. Water	a. Missouri River at nearest downstream drinking water intake.	Gamma Isotopic, H-3	1. Monthly Composite for Gamma Isotopic Analysis
	b. Missouri River downstream near the mixing zone.		2. Quarterly Composite for H-3 Analysis
	c. Missouri River upstream of plant intake (background).		

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.	Gamma Isotopic and I-131	1. Semimonthly grazing season (May to October)
	b. One (1) Dairy farm between 8 kilometers and 30 kilometers (background).		
5. Fish	a. Four fish samples within vicinity of plant discharge.	Gamma Isotopic	Once per season (May to October)
	b. One (1) background sample upstream of plant discharge.		
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.

(DELETED)

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. The specified frequencies may be adjusted to accommodate operation schedules except that variance should not exceed 1.25 times the specified interval.

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 Specification 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 gammer 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 "R" frequency (every 18 months).

d. The steam generator blowdown radiation monitors shall have:

(i) Daily channel checks.

(ii) Monthly source checks.

(iii) Quarterly channel functional tests.

(iv) Channel calibration at "R" frequency (every 18 months).

e. The steam generator blowdown effluent flow rate will be calibrated at "R" frequency (every 18 months) 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) A ventilation stack radiation 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) Each ventilation stack monitor shall have a quarterly channel functional test.

(iii) Each ventilation stack monitor shall be calibrated at "R" frequency (every 18 months).

(iv) The ventilation stack flow rate will be calibrated and functionally tested at "R" frequency (every 18 months). The stack radiation monitor flow rate will be calibrated and functionally tested at "R" frequency (every 18 months). Both will be determined operable by visual inspection daily.

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 "R" frequency (every 18 months).

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

(i) Daily channel check.

(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.

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

Basis

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)

3.12.1 Liquid and Gaseous Effluents (Continued)

3.12.2 Solid Radioactive Waste(1)

Applicability(1)

Objective(1)

Specification(1)

Basis(1)

(1) The surveillance requirements for this section will be incorporated under a supplement to this Facility License Change after the Process Control Program (PCP) has been issued.

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)(7)	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.

(7) 1 per 7 days.

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 Emitters (5)	1.0 E-04 (1)
B. Containment Purge Releases or Containment Pressure Relief Line Releases	Prior to each release	Principal Gamma Emitters (5)	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 Emitters (5)	1.0 E-04 (1)
D. Continuous (2) 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 materials 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.

5.5.2.8 e. The Fort Calhoun Station Emergency Plan and implementing procedures at least once every 12 months.

f. The Site Security Plan and implementing procedures at least once every 12 months.

g. The Safeguards Contingency Plan and implementing procedures at least once every 12 months.

h. The Radiological Effluent Program including the Radiological Environmental Monitoring Program and the results thereof, the Offsite Dose Calculation Manual and implementing procedures, and the Process Control Program for the solidification of radioactive wastes at lease once per 2 years.

i. Any other area of facility operation considered appropriate by the Safety Audit and Review Committee or the Assistant General Manager - Nuclear Production, Production Operations, Fuels, and Quality Assurance & Regulatory Affairs.

Authority

5.5.2.9 The Safety Audit and Review Committee shall report to and advise the Assistant General Manager - Nuclear Production, Production Operations, Fuels, and Quality Assurance & Regulatory Affairs on those areas of responsibility specified in Sections 5.5.2.7 and 5.5.2.8.

Records

5.5.2.10 Records of Safety Audit and Review Committee activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each Safety Audit and Review Committee meeting shall be prepared, approved, and forwarded to the Assistant General Manager - Nuclear Production, Production Operations, Fuels, and Quality Assurance & Regulatory Affairs within 14 days following each meeting.

b. Reports of reviews encompassed by Section 5.5.2.7 e, f, g, and h, above shall be prepared, approved, and forwarded to the Assistant General Manager - Nuclear Production, Production Operations, Fuels, and Quality Assurance & Regulatory Affairs within 14 days following completion of the review.

c. Audit reports encompassed by Section 5.5.2.8 above shall be forwarded to the Assistant General Manager - Nuclear Production, Production Operations, Fuels, and Quality Assurance & Regulatory Affairs and to the responsible management positions designated by the Safety Audit and review Committee within 30 days after completion of the audit.

5.5.3 Fire Protection Inspection

a. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm. The audit and inspection program responsibility shall rest with the Safety Audit and Review Committee.

b. An inspection and audit of the fire protection and loss prevention program by an outside qualified fire consultant shall be performed at intervals no greater than three years.

5.9.3 Special Reports

Special reports shall be submitted to the Director of Regulatory Operations 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, reference 3.4.

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.

g. Materials radiation surveillance specimens reports.

h. Fuel performance following each refueling outage.

i. Fire protection equipment outage, reference 2.19.

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.

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

5.9.4 (Continued)

The radioactive effluent release report shall include any changes[1] to the Process Control Program (PCP) or to the Offsite Dose Calculational 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).

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.

(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 (Continued)

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5.0 ADMINISTRATIVE CONTROLS

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 inservice 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 which are covered under the provisions of Section 5.12 of these Technical Specifications.

m. Records of the service lives of all hydraulic and mechanical snubbers listed on Table 2-6 (a) and (b) 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.

5.11 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.

ATTACHMENT B

DISCUSSION, JUSTIFICATION, AND SIGNIFICANT HAZARDS CONSIDERATION

DISCUSSION, JUSTIFICATION, AND SIGNIFICANT HAZARDS CONSIDERATION

The proposed revisions to the Fort Calhoun Station Unit No. 1 Technical Specifications are summarized and discussed in the following paragraphs. The proposed Technical Specifications are intended to provide the following exclusive functions:

1. Response to the Commission's request, dated November 15, 1978, in order to submit changes to the radiological effluent Technical Specifications.

2. Assurance to provide compliance to the requirements specified in Section 20.106 of 10 CFR Part 20.

3. Assurance to conform with dose design objectives of Appendix I to 10 CFR Part 50.

4. Clarification in other areas of the Technical Specifications with regard to effluents monitoring and surveillance requirements.

Section 2.9, "Radioactive Effluents" and Section 3.12, "Radioactive Waste Sampling and Monitoring"

The design of Fort Calhoun facility was evaluated in accordance with dose design objectives set forth in Appendix I to 10 CFR Part 50; consequently, Sections 2.9 and 3.12 of the Technical Specifications are revised appropriately. The proposed Technical Specification requirements for limiting conditions of operation and surveillance testing are developed for the purpose of keeping releases of radioactive materials in unrestricted areas during normal operation, including expected operational occurrences, as low as reasonably achievable.

Specification 2.9.1(2)d describes Hydrogen and Oxygen detectors for the Waste Gas Disposal System to be used to monitor the concentration of Hydrogen and Oxygen to prevent the existence of a flammable mixture of these gases in the Gas Decay Tanks. The detectors are currently not operable and the grab sample method also described in proposed Specification 2.9.1(2)d(i) is being utilized. These monitors will be returned to service as soon as reasonable on a schedule consistent with implementation of the approved specifications.

The proposed Technical Specifications provide reasonable assurance that the resulting radiation exposures will not exceed the dose design objectives set out in Appendix I to 10 CFR Part 50. At the same time, these Technical Specifications permit the operating flexibility, compatible with considerations of health and safety of the public, by taking into account unusual conditions of operation which may, on a temporary basis, result in exposures higher than the few percent of natural background radiation, but still within the limits specified in 10 CFR Part 20. As described in the USAR, radiation monitors are provided for the detection of leakage of radioactive fluids through the component cooling system into the raw water system. These monitors are installed in the return lines before they enter the circulating water discharge tunnel. Monitors on steam generator blowdown prevent the introduction of radioactive materials into the raw water discharge header that would exceed 10 CFR 20 limits at the river.

Section 3.11 "Radiological Environmental Monitoring Program"

Section 3.11 has been revised in order to measure changes in the levels of environmental radioactivity due to plant effluents. The proposed monitoring program will provide measurements of radiation and of radioactive materials in those exposure pathways; i.e., drinking water, fish, milk, direct radiation, etc., which can have significant impact on the dose contributions to an individual. This program has been developed, specifically for Fort Calhoun Station's environs, on the basis of 11 years of operational experience and in conformance with the Section IV.B of Appendix I to 10 CFR Part 50.

Table 2-1 of Section 2.14 "Engineered Safety Features System Initiation Instrument Setting Limits"

Item 3 of Table 2-1 provides the setpoint limits for various particulate, gaseous, and iodines effluent process radiation monitors that would initiate the containment Ventilation Isolation Actuation Signal (VIAS). A VIAS is initiated by, among other things, a Containment Atmosphere Radiation High Signal (CRHS) by deriving signal(s) from the high alarm setpoints of process radiation monitors. Further discussion on the modes and logic of system operation is provided in Section 7 and Section 11 of the USAR.

As the present setpoints of process radiation monitors for monitoring the gaseous waste effluents are based on the release rate limits specified in Section 2.9(2) of the existing Technical Specifications and the existing Technical Specifications have been revised to provide compliance with 10 CFR Part 20 and especially with 10 CFR Part 50, Appendix I, it was therefore considered imperative to revise the setpoints in accordance with the provisions of Specifications 2.9.1(2)a of the proposed Technical Specifications. Operation of the effluent process radiation monitors and VIAS system provides assurance that no member of the general public will be exposed at any time to radioactive effluents in excess of limits specified in 10 CFR Part 20.

The setpoints for the isolation function have been established in the Offsite Dose Calculational Manual (ODCM).

Table 3-3 of Section 3.2 "Instrumentation and Control"

Item 3 (Area and Process Monitors) of Table 3-3 specifies the minimum surveillance frequencies for channel checks, channel calibrations, and source checks for all the area process and post accident radiation monitors installed at Fort Calhoun Station. As the surveillance requirements for effluent process monitors (liquid and gaseous) are explicitly specified in Specification 3.12 in accordance with the Standard Technical Specifications, it is appropriate to exclude these radiation monitors from Item 3 of Table 3-3. The intent of the proposed change to Item 3 is to provide clarification.

Table 3-4 of Section 3.2 "Equipment and Sampling Tests"

Item 2 (Steam Generator Coolant) of Table 3-4 deals with a radioactivity analysis for a liquid effluent. As the minimum sampling frequencies for conducting analyses are specified more explicitly in Table 3-11 (Radioactive Liquid Waste Sampling and Analysis) of Section 3.12 of the proposed Technical Specifications, it was considered inappropriate to list the same requirement under Section 3.2, which is considered to be more general with respect to in-plant radioactivity analyses for the radioactive system.

Significant Hazards Considerations:

1) This proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Operating equipment is not altered in any way by this proposal; hence, the conclusions of previous evaluations remain valid. This proposal improves the ability to monitor and control the release of radiological effluents. The increased surveillance requirements of the proposal provide greater assurance that improper radiological effluent releases will not occur and that the consequences of any accident may be diminished by these Technical Specifications.

2) The proposed amendment will not create the probability of a new or different kind of accident from any previously evaluated. Neither plant operating equipment nor safeguards equipment are altered, removed, or replaced by this proposal. Rather, the method for control of radiological effluents from the site has been standardized in accordance with peer recommendations and experience. The changes provided improved methods for surveillance within the plant and in the adjacent environment. The surveillance of all effluents, whether planned or unplanned, is improved by these Technical Specifications.

3) The proposed amendment does not involve a reduction in a margin of safety. As stated previously, the proposed changes provide assurance that radiological effluents are properly monitored prior to, during, and after release from the station. The margin of safety within direct operation of plant systems is unaffected; i.e., neither diminished nor increased. However, the status and content of radiological effluents in liquid, gas, and solid form is closely administered through surveillance and control procedures encompassed within these Technical Specifications.

COMPARISON OF STANDARD RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS
FOR PRESSURIZED WATER REACTORS (NUREG-0472)
AND PROPOSED TECHNICAL SPECIFICATIONS
FOR FORT CALHOUN STATION UNIT NO. 1

Section or Subsection of STS	Section or Subsection of Fort Calhoun Tech. Specifications	Remarks
1.0	DEFINITIONS	Appropriate/applicable definitions have been incorporated.
3.3.3.10	2.9.1(1), 3.12.1(1) and Section 2 of ODCM	All liquid effluent radiation monitoring instrumentation specifically applicable to the design of Fort Calhoun Station has been incorporated.
3.3.3.11	2.9.1(2), 3.12.1(2), and Section 2 of ODCM	--
3.11.1.1	2.9.1(1)a	The ACTION statement under Specification 2.9.1(1)a(ii) is slightly different than the one specified under 3.11.1.1. This is because, with the concentration of radioactive material released from the site to unrestricted areas exceeding the limits specified under Table II, Appendix B of 10 CFR Part 20, the following course of actions would alleviate the problem by: 1) decreasing the release rate of radioactive materials beings discharged to the environment and/or 2) increasing the dilution flow rate to restore concentrations within the required limits. If the concentrations of radioactive materials released to the unrestricted areas (when averaged over a period of 24 hours) were to exceed 500 or 5000 times the limits specified for such materials, the licensee is required to notify the Commission pursuant to Section 20.403 of 10 CFR Part 20.
3.11.1.2.a	2.9.1(1)b	The contents of ACTION statement are in conformance with Appendix I to 10 CFR Part 50.
3.11.1.2.b	Design Objective "A" of 2.9.1	The requirements are in conformance with Paragraph "A" and B.1, Section II, Appendix I, 10 CFR Part 50.

Section or Subsection of STS	Section or Subsection of Fort Calhoun Tech. Specifications	Remarks
3.11.1.3	2.9.1(1)c	The dose calculation requirement is not considered appropriate especially when the equipment available in the liquid radwaste treatment system is used prior to any releases to the environs in order to comply with the ALARA philosophy. See also Section 2 of the ODCM for further explanation.
3.11.1.4	--	As Fort Calhoun Station does not have outside storage tanks, a 10 curie limit for outside liquid radioactive effluent tanks has been placed in the ODCM.
3.11.2.1	2.9.1(2)a	Based on the operating history of the plant and based on the evaluation of Fort Calhoun Station in accordance with Appendix I dose design objectives, the proposed Technical Specification provides sufficient assurance to conform with 10 CFR Part 20 limits. The ACTION statement under Specification 2.9.1(2)a(ii) is slightly different than the one specified under 3.11.2.1. For this explanation see comments in response to item 3.11.1.1 and 4.11.1.1.
3.11.2.2.a	2.9.1(2)b	--
3.11.2.2.b	Design Objective "B" of Specification 2.9.1	--
3.11.2.3.a	--	Reference response to items 3.11.2.2.a, 3.11.2.2.b, and 4.11.2.2.
3.11.2.3.b	--	Reference response to items 3.11.2.2.a, 3.11.2.2.b, and 4.11.2.2.
3.11.2.4	2.9.1(2)c	The gaseous waste treatment system at Fort Calhoun primarily consists of a holdup system; i.e., gas decay tanks. These tanks are used to take credit for

COMPARISON OF STANDARD RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS
FOR PRESSURIZED WATER REACTORS (NUREG-0472)
AND PROPOSED TECHNICAL SPECIFICATIONS
FOR FORT CALHOUN STATION UNIT NO. 1
(Continued)

Section or Subsection of STS	Section or Subsection of Fort Calhoun Tech. Specifications	Remarks
		the decay of short half-life radionuclides (e.g., Xe-133, I-131, etc). Further discussion on the holdup time is provided in Section 8 of the ODCM. The proposed Technical Specification provides assurance that conformance with 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and Appendix I dose design objectives is achieved.
3.11.2.5	2.9.1(2)d	Based on the plant operating experience and following the definitions of flammable concentrations for hydrogen and oxygen, the proposed Technical Specification 219.1(2)d provides an appropriate method.
3.11.2.6a&b	--	The maximum activity contained in each gas decay tank has been calculated equivalent to 2.56E+04 curies. This calculated quantity is based upon 5% χ/Q, an average gross energy of 0.19 Mev per disintegration (for Xe-133), and an offsite dose limit equivalent to 500 mrem. As this calculated activity is higher than one computed for 1% failed fuel design basis (Section 11.1.3 and Table 11.1.19 of the USAR), the applicability of this specification to the Fort Calhoun Station is considered inappropriate.
3.11.3	2.9.2	--
3.11.3.a	2.9.2	--
3.11.4	--	These specifications are not considered appropriate due to: 1) Fort Calhoun Station design was evaluated in accordance with Appendix I dose design objective in June 1976, and the results verified that doses to the maximum exposed individual are well within the Appendix I dose design limits; 2) the proposed Technical Specifications are considered very restrictive when compared with the doses specified

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		under item 3.11.2.5 so the licensee will always be in compliance with the limits specified under 40 CFR Part 190; 3) a site like Fort Calhoun Station does not fall under the entire definition and scope of "Uranium Fuel Cycle."
3.12.1	3.11	--
3.12.1.a	5.9.4.b	--
3.12.1.b	3.11	LLD specifics are in the ODCM.
3.12.1.c	3.11 5.9.4.a & b	Table 3.9 --
3.12.2.a & b	3.11.3 5.9.4.b	Based on the land use survey conducted in 1976 and then in 1978, the special reporting and surveillance requirements are considered too restrictive and unjustifiable. The results of the land use survey shall be submitted in the Annual Radiological Environmental Operating Report, whenever applicable.
3.12.3	3.11.4	--
4.3.3.10	3.12.1(1)c, 3.12.1(1)d, 3.12.1(1)e, and 3.12.1(1)f	--
4.3.3.11	3.12.1(2)b, 3.12.1(2)c, 3.12.1(2)e, and 3.12.1(2)f	--
4.11.1.1.1	3.12.1(1)a	--
4.11.1.1.2	2.9.1(1)a, 2.9.1(1)f, and Section 3, ODCM	--
4.11.1.2	2.9.1(1)b	The dose contributions shall be determined on a quarterly basis instead of on a monthly basis. This is considered appropriate as the Fort Calhoun Station has been in

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		commercial operation since August 1973, and during the past 11 years of operation the annual releases of radionuclides, via liquid and gaseous effluents, to the unrestricted areas and the resulting does have been a fraction of the appendix dose design objectives. The Fort Calhoun Station design was evaluated in June 1976, in accordance with Appendix I to 10 CFR Part 50, and the resultant doses were well within the specified limits. the plant was licensed in accordance with Section 50.34a of 10 CFR Part 50, and the unit has always operated within the intent and scope of Section 50.36a of 10 CFR part 50 in order to comply with the as low as reasonably achievable philosophy. Therefore increasing the frequency, for the determination of doses, from quarterly to once per 31 days would not serve any meaningful purpose especially when it is clear, based on previous operating data, that the dose to an individual on a quarterly basis will be less than or equal to one-half of the annual dose objectives and on an annual basis will be less than the dose design objectives as per Appendix I to 10 CFR Part 50.
4.11.1.3	--	Reference comments in response to items 3.11.1.3 and 4.11.1.2 above. This requirement is not considered appropriate as the equipment available in liquid waste treatment system is used prior to all effluent releases and the average frequency of operation has always been more than the specified value; i.e., once per 92 days.
4.11.1.4	--	Not appropriate per comment in response to item 3.11.1.4.
4.11.2.1.1	--	The release rate shall be determined to conform with 10 CFR Part 20.

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4.11.2.1.2	3.12.1(2)a and 2.9.1(2)f	The release rates or concentrations of radionuclides shall be determined to conform with the 10 CFR Part 20 limits.
4.11.2.2	2.9.1(2)b	The dose contributions shall be determined on a quarterly basis instead of once per 31 days. See comments in response to item 4.11.1.2 above.
4.11.2.3	--	Reference response to items 3.11.2.2.a, 3.11.2.2.b, and 4.11.2.2 above.
4.11.2.4.1	--	Reference comments in response to items 3.11.2.4 and 4.11.2.2 above.
4.11.2.5	2.9.1(2)d	Reference comments in response to item 3.11.2.5 above.
4.11.2.6	--	The maximum activity contained in each gas tank has been calculated equivalent to 2.56E+04 curies. This calculated quantity is based upon 5% χ/Q, an average gross energy of 0.19 Mev per disintegration (for Xe-133), and an offsite dose limit equivalent to 500 mrem. As this calculated activity is higher than one computed for 1% failed fuel design basis (Section 11.1.3 and Table 11.1.19 of the USAR), the applicability of this specification to the Fort Calhoun Station is considered inappropriate.
4.11.3.a & b	--	These specifications will be proposed after the new solid waste treatment system and process control program are fully operational at Fort Calhoun Station.
4.12.1	3.11	The specific requirements regarding the program defined under Table 3.12-1 have been incorporated in the OCDM.
4.12.2	3.11.3 5.9.4.b	Reference comments in response to item 3.12.2.a and b above.

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4.12.3	5.9.4.b	--
6.5.1.6.k	5.5.1.6.a	This responsibility is covered under the current Technical Specification referenced.
6.5.1.6.1	5.5.1.6.e	This responsibility is covered under the current Technical Specification referenced.
6.5.2.8	5.5.2.8	The appropriate sections have been added.
6.8.1	5.8.1	Provided by present Technical Specification.
6.9.1.9	5.9.2.b	Provided by present Technical Specification.
6.9.1.11	5.9.4.b	--
6.9.1.12	5.9.4	--
6.9.2	5.9.3	Provided by present Technical Specification.
6.10.2	5.10.2	--
6.13.2	5.9.4.a	--
6.14	5.9.4.a (footnote)	The first draft of the Process Control Program (PCP) will be reviewed by the Commission and any changes following the first submittal would not significantly change the safety related intent of the PCP. Also, these changes will be properly reviewed by the licensee before any implementation. Conformance to appropriate guides or regulations will be assured. Therefore, a special section on the PCP is not considered appropriate.
6.14.2	5.9.4.a	--
6.15	5.9.4.a (footnote)	Similar justification as above for the PCP.
6.15.1	5.9.4.a	--

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6.16	--	This specification is considered inappropriate and unjustifiable especially when the Fort Calhoun Station was designed in accordance with Section 50.34a of 10 CFR Part 50 and was evaluated in June 1975 in accordance with Appendix I to 10 CFR Part 50. This section, in fact, applies to only those plants which have not been issued an operating license. Major changes to radioactive waste systems (liquid, gaseous, and solid', initiated by the licensee are always reviewed by the Plant Review Committee and, if appropriate, the Safety Audit and Review Committee in accordance with 10 CFR Part 50.59. Any changes are always submitted in the Monthly Operating Report.