April 3, 1985

Docket No. 50-285

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Mr. R. L. Anrews
Division Manager - Nuclear Production
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Andrews:

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated March 21, 1978 and supplements dated March 30, 1979 and October 18, 1984.

The amendment revises the TS to incorporate the requirements of Appendix I of 10 CFR Part 50 as the Radiological Effluent Technical Specifications (RETS). The RETS provide new TS sections defining limiting conditions of operation and surveillance requirements for radiological liquid and gaseous effluent monitoring; concentration, dose, and treatment of liquid, gaseous and solid wastes; total dose; and radiological environmental monitoring. In addition, some changes are made in administrative controls, specifically dealing with the Process Control Program (PCP) and Offsite Dose Calculation Manual (ODCM).

A copy of the Safety Evaluation with an attached Technical Evaluation Report is also enclosed. The notice of issuance will be included in the Commission's next regular monthly Federal Register notice.

It should be noted that the issuance of this amendment does not totally complete our actions related to this subject. As you know, during our review, minor discrepancies were found in our review of your ODCM. These discrepancies were sent to your staff under separate cover. We are in receipt of your proposed resolutions and we will respond to your resolutions before the RETS are implemented. In addition, the one page specific surveillance requirements for solid radioactive waste, specification 3.12.2, has not been received yet, and this will be the subject of a separate licensing action. You are requested within 60 days of receipt of this letter to submit the page.

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PDR ADOCK 05000285
P PDR

The reporting and/or recordkeeping requirements of this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Original signed by:

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 86 to DPR-40
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
PMKreutzer	ETourigny:dd	JRMiller	M. Karman	GCLainas
3/27/85	3/27/85	3/28/85	3/28/85	4/1/85

Ft. Calhoun

Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, NW
Washington, DC 20036

Mr. Jack Jensen, Chairman
Washington County Board of Supervisors
Blair, Nebraska 68023

Metropolitan Planning Agency
ATTN: Dagnia Prieditis
7000 West Center Road
Omaha, Nebraska 68107

Mr. Larry Yandell
U.S. NRC Resident Inspector
Post Office Box 309
Fort Calhoun, Nebraska 68023

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
C-E Power Systems
7910 Woodmont Avenue
Bethesda, Maryland 20814

Regional Administrator
USNRC, Region IV
Office of Executive Director
for Operations
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

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

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee) dated March 21, 1978 and supplements dated March 30, 1979 and October 18, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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P PDR

2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 86 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within six months of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 3, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Revise Appendix "A" Technical Specifications as indicated below. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

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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 behaviour 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 known 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,

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)

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

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 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 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 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, 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 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 analysed: (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.

(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 areas. 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.

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 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 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

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, 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.

2.9.2 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 (3) b. Containment Spray c. Containment Isolation d. Containment Air Cooler DBA Mode	≤ 5 psig
2. Pressurizer Low/Low Pressure	a. Safety Injection (3) b. Containment Spray 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 ≤ 5.9 (4) 20.8 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	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.

(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-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 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.

3.0 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.

3.11 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.

TABLE 3-10

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 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 "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 shich 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.

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
	Monthly (3)	Tritium (H-3)	1.0 E-06
C. Condenser Air Ejector Releases	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 Emitters (5) 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 twelve months.

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

g. The Safeguards Contingency Plan and implementing procedures at least once every twelve 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 least 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 5.5.2.7e, f, g, h, and i 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.

5.0 ADMINISTRATIVE CONTROLS

5.5.3 Fire Protection Inspection

a. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site 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 3 years.

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.

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

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

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).

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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)

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.

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 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.

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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-40

FORT CALHOUN STATION

OMAHA PUBLIC POWER DISTRICT

Docket No. 50-285

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, the Omaha Public Power District (OPPD) has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. OPPD filed this information with the Commission by letter dated October 18, 1984, which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-40 for Fort Calhoun Station. The proposed technical specifications update those portions of the technical specifications addressing radioactive waste management and make them consistent with the current staff positions as expressed in NUREG-0472. These revised technical specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR Parts 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63 and 64; and with 10 CFR Part 50, Appendix B.

2.0 BACKGROUND AND DISCUSSION

2.1 Regulations

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.36a, "Technical specifications on effluents from nuclear power reactors," provides that each license authorizing operation of a nuclear power reactor will include technical specifications that (1) require compliance with applicable provisions of Part 20.106, "Radioactivity in effluents to unrestricted areas;" (2) require that operating procedures developed for the control of effluents be established and followed; (3) require that equipment installed in the radioactive waste system be maintained and used; and (4) require the periodic submission of reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents, any quantities of radioactive materials released that are significantly above design objectives, and such other information as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

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10 CFR Part 20, "Standards for Protection Against Radiation," paragraphs 20.105 (c), 20.106(g), and 20.405(c), require that nuclear power plant and other licensees comply with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded. 10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases of radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage; and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

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

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

2.2 Standard Radiological Effluent Technical Specifications

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

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

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

(1) Instrumentation
(2) Radioactive effluents
(3) Radiological environmental monitoring
(4) Design features
(5) Administrative controls.

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

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

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

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

3.0 EVALUATION

The attached report (EGG-PBS-6765) was prepared for us by EG&G Idaho, Inc. as part of our technical assistance contract program. Their report provides their technical evaluation of the compliance of the licensee's submittal with NRC-provided criteria. The staff has reviewed this TER and agrees with the evaluation.

3.1 Summary

The proposed changes to the radiological effluent technical specifications for Fort Calhoun Station have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0472 (Fort Calhoun Station is a pressurized water reactor) and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications. The proposed changes will not remove or relax any existing requirement related to the probability or consequences of accidents previously considered and do not involve a significant hazards consideration.

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

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 GENERAL CONCLUSION

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

Attachment:
INEL Technical Evaluation Report

Date: April 3, 1985

Principal Contributor:
C. Miller, DSI

Table 1. Relation Between Provisions of the Regulations and the Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors and Boiling Water Reactors

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

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

*Note: Needed to fully implement other specifications.

EGG-PBS-6765

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) IMPLEMENTATION - FORT CALHOUN STATION

Thomas E. Young
William Serrano
Stephen W. Duce
John W. Mandler
Ferrol B. Simpson

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EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

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