January 16, 1984

DMB 016

Dockets Nos. 50-269, 50-270
and 50-287

Mr. H. B. Tucker
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 125 , 125 , and 122 , to Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated February 9, 1983, as supplemented by submittals dated February 28, 1983, and April 28, 1983.

The amendments revise the TSs principally to incorporate changes to the Radiological Effluent Technical Specifications (RETS) in order to bring them into compliance with Appendix I of 10 CFR Part 50 as well as meeting other regulations pertinent to radioactive waste management. As discussed with and agreed to by your staff on January 5, 1984, you are required to submit within 90 days of receipt of this letter a proposed TS addressing explosive gas limitations and monitoring for our review pending completion of the ongoing Duke Waste Gas Study, as discussed in the enclosed Safety Evaluation. We urge and encourage you to exert every effort to complete the remaining evaluations associated with this study and to forward your submittal, as agreed to in the August 18-19, 1982 meeting, to us at the earliest opportunity.

A copy of our Safety Evaluation and also a copy of the Technical Evaluation Report (TER) prepared by our consultant, Franklin Research Center, are enclosed. With regard to the TER conclusions noted on page 23, we advise you that in order to facilitate implementation of the Oconee RETS, your proposed specifications addressing the process control program (PCP) for processing wet radioactive waste are accepted on an interim basis. We will subsequently discuss with you any changes that may be necessary to satisfy 10 CFR Part 61.

In addition, in your submissions dated February 28, 1983, and April 28, 1983, you provided as a reference document an "Offsite Dose Calculation Manual, Duke Power Company: Oconee." We find that the ODCM uses documented and approved methods that are consistent with the methodology and guidelines in

NUREG-0133, and, therefore, is an acceptable reference. You should, however, in the next revision of the ODCM, provide meteorological dispersion data (X/Q and D/Q) for ground level releases from the roof vents.

Notice of Issuance will be included in the Commission's Monthly Notice.

Sincerely,

ORIGINAL SIGNED BY
JOHN F. STOLZ

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 125 to DPR-38
2. Amendment No. 125 to DPR-47
3. Amendment No. 122 to DPR-55
4. Safety Evaluation
5. Technical Evaluation

cc w/enclosures:
See next page

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Duke Power Company

cc w/enclosure(s):

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated February 9, 1983, as supplemented February 28, 1983, and April 28, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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PDR ADOCK 05000269
P PDR

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 16, 1984

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated February 9, 1983, as supplemented February 28, 1983, and April 28, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 16, 1984

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 122
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated February 9, 1983, as supplemented February 28, 1983, and April 28, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 122 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 16, 1984

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO.125 TO DPR-38

AMENDMENT NO.125 TO DPR-47

AMENDMENT NO.122 TO DPR-55

DOCKETS NOS. 50-269, 50-270 AND 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
ii	ii
iii	iii
iv	iv
v	v
-	v-a
vi	vi
1-5	1-5
-	1-6
-	3.5-34
-	3.5-35
-	3.5-36
-	3.5-37
-	3.5-38
-	3.5-39
-	3.5-40
3.9-1	3.9-1
3.9-2	3.9-2
3.9-3	3.9-3
-	3.9-4
-	3.9-5
3.10-1	3.10-1
3.10-2	3.10-2
3.10-3	3.10-3
3.10-4	3.10-4
3.11-1	3.11-1
-	3.11-2
4.1-1	4.1-1
4.1-2	4.1-2
4.1-5	4.1-5
4.1-10	4.1-10
4.1-11	4.1-11
4.1-12	4.1-12
4.1-13	4.1-13
-	4.1-14
-	4.1-15
-	4.1-16
-	4.1-17
-	4.1-18

Remove Pages	Insert Pages
4.11-1	4.11-1
4.11-2	4.11-2
4.11-3	4.11-3
4.11-4	4.11-4
4.11-5	4.11-5
-	4.11-6
-	4.11-7
-	4.11-8
-	4.21-1
-	4.21-2
5.1-1	5.1-1
6.1-1a	6.1-1a
6.1-2	6.1-2
6.1-3	6.1-3
-	6.1-3a
6.1-4	6.1-4
6.1-5	6.1-5
-	6.1-5a
6.4-1	6.4-1
-	6.4-2
6.5-2	6.5-2
6.6-1	6.6-1
6.6-2	6.6-2
6.6-3	6.6-3
6.6-6	6.6-6
-	6.6-6a
6.6-7	6.6-7
6.6-8	Delete
6.6-9	Delete
-	6.8-1

Amendments Nos. 125 , 125 , & 122

LIST OF TABLES

1.8 RADIOLOGICAL EFFLUENT CONTROL

1.8.1 Source Check

A Source Check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.8.2 Offsite Dose Calculation Manual (ODCM)

The Offsite Dose Calculation Manual is a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints and in the conduct of environmental radiological monitoring.

1.8.3 Process Control Program (PCP)

The Process Control Program is a procedure that shall contain the sampling, analysis, and formulation determination by which solidification of radioactive liquid waste is assured.

1.8.4 Solidification

Solidification shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the waste type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the Process Control Program (PCP).

1.8.5 Gaseous Radwaste Treatment System

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

1.8.6 Ventilation Exhaust Treatment System

A Ventilation Exhaust Treatment System is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Engineered Safety Features (ESF) atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment System components.

1.8.7 Purge-Purging

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

1.8.8 Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during Venting. Vent, used in system names, does not imply a venting process.

1.8.9 Member(s) Of The Public

Members(s) Of The Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

1.8.10 Unrestricted Area

An Unrestricted Area shall be any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commerical institutional and/or recreational purposes.

3.5.5 Radioactive Effluent Monitoring Instrumentation

Applicability

Applies to radioactive liquid effluent, gaseous effluent, and gaseous process monitoring instrumentation.

Specifications

3.5.5.1 Liquid Effluents

a. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.5.5-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.1 are not exceeded.

b. If a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. In the event that the number of operable radioactive liquid effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-1, Column A, action shall be as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.2 Gaseous Process and Effluents

a. The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.5.5-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.10.1 are not exceeded.

b. If a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. In the event that the number of radioactive gaseous process or effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-2, Column A, action shall be taken as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.3 Setpoints

The setpoints shall be determined in accordance with the methodology described in the ODCM and shall be recorded. Setpoint correction may be permitted without declaring the channel inoperable.

3.5.5.4 The provisions of Technical Specification 3.0 do not apply.

Bases

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentration of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

Table 3.5.5-1
LIQUID EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS

	INSTRUMENT	A MINIMUM OPERABLE CHANNELS	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1.	Monitors Providing Automatic Termination of Release			
	Liquid Radwaste Effluent Line Monitors			
	1 RIA-33	1	*	(a)
	Turbine Building Sump			
	1 RIA-54 (Units 1 & 2)	1	*	(b)
	3 RIA-54 (Unit 3)	1	*	(b)
2.	Monitors not Providing Automatic Termination of Release			
	Low Pressure Service Water			
	1 RIA-35	1	*	(d)
	2 RIA-35	1	*	(d)
	3 RIA-35	1	*	(d)
3.	Flow Rate Measuring Devices			
	Liquid Radwaste Effluent Line	1	*	(c)
	Keowee Hydroelectric Station Tailrace Discharge **	NA	NA	NA
4.	Continuous Composite Sampler			
	#3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent)	1	*	(d)

*At all times.
**Flow determined from number of hydro units operating; if hydro is not operating, leakage flow, which is measured periodically, is used.

Table 3.5.5-1 NOTES

(a) Effluent releases may continue provided that prior to initiating a release:

1. Two independent samples are analyzed in accordance with Specification 3.9 and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted.

Otherwise, suspend release of radioactive effluents by this pathway.

(b) Effluent releases may continue provided that prior to each discrete release of the sump, grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml.

(c) Effluent releases may continue provided flow rate is estimated at least once per four hours during actual releases.

(d) Effluent releases may continue provided that grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml every 12 hours.

Table 3.5.5-2
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1. Waste Gas Holdup Tanks			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination Of Release (RIA-37, - 38)	1	**	(a)
b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow)	1	**	(b)
2. Unit Vent Monitoring System			
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Containment Purge Release (RIA - 45)	1	*	(a)
b. Iodine Sampler	1	*	(d)
c. Particulate Sampler	1	*	(d)
d. Effluent Flow Rate Monitor (Unit Vent Flow)	1	*	(b)
e. Sampler Flow Rate Monitor	1	*	(e)
f. Effluent Flow Rate Monitor (Containment Purge)	1	**	(b)
3. Interim Radwaste Building Ventilation Monitoring System			
a. Noble Gas Activity Monitor (RIA - 53)	1	*	(c)
b. Iodine Sampler#	1	*	(d)
c. Particulate Sampler#	1	*	(d)

Table 3.5.5-2 (Cont'd)
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust)#	1	*	(b)
e. Sampler Flow Rate Monitor#	1	*	(e)
4. Hot Machine Shop Ventilation Monitoring System			
a. Iodine Sampler#	1	*	(d)
b. Particulate Sampler#	1	*	(d)
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust)#	1	*	(b)
d. Sampler Flow Rate Monitor#	1	*	(e)

* At all times.
** During waste gas holdup tank releases and/or containment purge operation.
\# Effective upon installation of equipment.

Table 3.5.5-2 NOTES

(a) Effluent releases from waste gas tanks or containment purges may continue provided that prior to initiating a release:

1. Two independent samples are analyzed and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted and;

Effluent release from ventilation system or condenser air ejectors may continue provided that grab samples are taken once per 8 hours and these samples are analayzed for gross activity (beta and/or gamma) within 24 hours, or continuously monitor through the unit vent. Otherwise, suspend release of radioactive effluents via this pathway.

(b) Effluent releases may continue provided the flow rate is estimated at least once per 4 hours.

(c) Effluent releases may continue provided grab samples are taken once per 8 hours and these samples are analayzed for gross activity (beta and/or gamma) within 24 hours.

(d) Effluent releases may continue provided samples are continuously collected with auxiliary sampling equipment for periods not to exceed 7 days and analyzed within 48 hours of the end of sample collection.

(e) Alarms indicating low flow may be substituted for flow measuring devices.

3.9 RADIOACTIVE LIQUID EFFLUENTS

Applicability

Applies at all times to the controlled release of all liquid waste discharged from the site which may contain radioactive materials, except as noted. Appendix I dose limits for radioactive liquid effluent releases (T.S. 3.9.2) are applicable only during normal operating conditions which include expected operational occurrences, and are not applicable during unusual operating conditions that result in activation of the Oconee Emergency Plan.

Objective

To establish conditions for the controlled release of radioactive liquid effluents. To implement the requirements of 10 CFR 20, 10 CFR 50.36a, Appendix A to 10 CFR 50, Appendix I to 10 CFR 50, 40 CFR 141 and 40 CFR 190.

Specification

3.9.1 Concentration

a. The concentration of radioactive material released at anytime from the site boundary for liquid effluents to Unrestricted Areas (denoted in Figure 2.1-4(a) of the Oconee Nuclear Station Final Safety Analysis Report) shall be limited to the concentration specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases the concentration shall be limited to 2×10^{-4} μCi/ml total activity.

b. If the concentration of radioactive material released in liquid effluents to Unrestricted Areas exceeds the above Specified limits, without delay restore the concentration to within the above limits.

3.9.2 Dose

a. The dose or dose commitment to a Member Of The Public from radioactive materials in liquid effluents to Unrestricted Areas shall be limited to:

1) during any calendar quarter:

$\leq$ 4.5 mrem to the total body

$\leq$ 15 mrem to any organ and;

2) during any calendar year:

$\leq$ 9 mrem to the total body

$\leq$ 30 mrem to any organ.

b. If the calculated dose from the release of radioactive materials in liquid effluents exceeds any of the above limits, except during unusual operating conditions that result in activation of the Oconee Emergency Plan, and in lieu of any other report required by Section 6.6.2, a report shall be submitted within 30 days from the end of the quarter during which the release occurred, to the regional NRC Office which includes the following:

1. Cause(s) for exceeding the limit(s)

2. A description of the program of corrective action initiated to: reduce the releases of radioactive materials in liquid effluents, and to keep these levels of radioactive materials in liquid effluents in compliance with the above limits, or as low as reasonably achievable.

3. Results of radiological analyses of the drinking water source and the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141.

3.9.3 Liquid Waste Treatment

a. The appropriate subsystems of the liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid waste prior to their discharge, if the projected dose due to liquid effluent releases to unrestricted areas, when averaged over 31 days would exceed 0.18 mrem to the total body or 0.6 mrem to any organ.

b. If radioactive liquid waste is discharged without treatment and in excess of the above limit, a report shall be submitted within 30 days to the regional NRC Office which includes the following:

1. Cause of equipment or subsystem inoperability.

2. Corrective action to restore equipment and prevent recurrence.

3.9.4 Chemical Treatment Ponds (CTP 1 and 2)

a. The quantity of radioactive material in the Chemical Treatment Ponds (CTP) shall be limited so that, for all radionuclides identified, excluding noble gases and tritium, the sum of the ratios of activity (in curies) to the limits in 10 CFR 20, Appendix B, Table II, Column 2 shall not exceed 1.7×10^5.

$$\sum_j \frac{A_j}{C_j} < 1.7 \times 10^5$$

where Aj = pond inventory limit for single radionuclide 'j' (curies)

Cj = 10 CFR 20, Appendix B, Table II, Column 2, concentration for single radionuclide 'j' (curies)

b. After a primary to secondary leak is detected, the initial batch of used Powdex resin shall not be transferred to the CTP. No batch of used powdex resin shall be transferred to the CTP unless the sum of the ratios of the activity of the radionuclides identified in the preceeding batch from any powdex cell in the same unit is less than 0.1% of the limit identified in 3.9.4.a.

$$\sum_j \frac{Qj}{Aj} < 1.0 \times 10^{-3}$$

where Qj = radionuclide activity in the batch

Aj = pond inventory limit for radionuclide 'j'

c. The radionuclide inventory per batch of used powdex resin transferred, averaged over the transfers of the previous 13 weeks, shall not exceed 0.01% of the pond radionuclide inventory limit. If this average exceeds 0.01% of the pond radionuclide inventory limit, then a report will be submitted within 30 days to the Regional NRC Office describing the reason or reasons for exceeding the objective and plans for future operation. Decay of radionuclides may be taken into account in determining inventory levels.

$$\frac{Qj_1 + Qj_2 + \ldots Qj_{(n-1)} + Qjn}{n} \leq .01\% \times Aj$$

where Qj = activity or radionuclide 'j' in the batch

n = number of batches transferred to the chemical treatment ponds during the previous 13-week period.

3.9.5 Liquid Holdup Tanks

a. The quantity of radioactive material contained in each outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

b. The quantity of radioactive material contained in each of the outside temporary tanks shall be determined to be within the above limit by analyzing a representative sample of the tanks contents at least once per 7 days when radioactive materials are being added to the tank.

c. If the quantity of radioactive material in any outside temporary tank exceeds the above limit, suspend all additions to radioactive material to the tank without delay.

3.9.6 The provisions of Technical Specification 3.0 do not apply.

Bases

The concentration specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The dose specification is provided to assure that the release of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I. that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated.

Section IV of Appendix I of 10 CFR 50 states that the licensee is permitted the flexibility of operation during unusual operating conditions, to assure the public is provided with a dependable source of power when compatible with considerations of health and safety of the public. Section I of Appendix I of 10 CFR 50 states that this appendix provides specific numerical guides for design objectives and limiting conditions for operation, to assist holders of licenses for light-water-cooled nuclear power reactors in meeting the requirements to keep releases of radioactive material to unrestricted areas as low as practical, and reasonably achievable, during normal reactor operations, including expected operational occurrences. Using the flexibility granted during unusual operating conditions, and the stated applicability of the design objectives for the Oconee Nuclear Station, Appendix I dose limits for radioactive liquid effluent releases (T.S. 3.9.2), are concluded to be not applicable during unusual operating conditions that result in the activation of the Oconee Emergency Plan.

For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix A to 10 CFR Part 50.

The inventory limits of the chemical treatment ponds are based on limiting the consequences of an uncontrolled release of the pond inventory. The short term rate limit (2 mrem/hr) of 10CFR20.105 is applied to 10CFR20.106 in the following expression:

$$\frac{\dfrac{Aj}{1.3 \times 10^6 \text{ gal}} \times \dfrac{10^6 \ \mu\text{Ci}}{\text{curie}} \times \dfrac{\text{gal}}{3785 \text{ ml}}}{Cj} \leq \frac{2 \text{ mrem/hr}}{500 \text{ mrem/yr}} \times \frac{8760 \text{ hr}}{\text{yr}}$$

$$\frac{Aj}{Cj} \leq 1.7 \times 10^5$$

where Aj = pond inventory limit for radionuclide 'j' (curies)

Cj = 10CFR20 Appendix B, Table II, Column 2 concentration for radionuclide 'j'

1.3×10^6 gal = estimated volume of smaller chemical treatment pond

The batch limits provide assurance that activity input to the CTP will be minimized.

3.10 RADIOACTIVE GASEOUS EFFLUENTS

Applicability

Applies at all times to the controlled release of all gaseous waste discharged from the station which may contain radioactive materials.

Objective

To establish conditions for the controlled release of radioactive gaseous effluents. To implement the requirements of 10CFR20, 10CFR50.36a, Appendix A to 10CFR50, Appendix I to 10CFR50, and 40CFR190.

Specifications

3.10.1 Dose Rate

a. The instantaneous dose rate at the site (exclusion area) boundary for gaseous effluents (Figure 2.1-4(a) of the Oconee Nuclear Station Final Safety Analysis Report) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

1. The dose rate limit for noble gases shall be:

   $\leq$ 500 mrem/yr to the total body

   $\leq$ 3000 mrem/yr to the skin and;

2. The dose rate limit for all radioiodines and for all radioactive materials in particulate form and radionuclides other than noble gases with half-lives greater than 8 days shall be $\leq$ 1500 mrem/yr to any organ.

b. If the dose rate exceeds the above limits, without delay decrease the release rate to within the above limits.

3.10.2 Dose

a. The air dose due to noble gases released in gaseous effluent from the site shall be limited to the following:

1. During any calendar quarter:

   $\leq$ 15 mrad for gamma radiation
   $\leq$ 30 mrad for beta radiation

2. During any calendar year:

   $\leq$ 30 mrad for gamma radiation
   $\leq$ 60 mrad for beta radiation

b. The dose to a Member Of The Public from radioiodines, tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released from the site, shall be limited to the following:

1. During any calendar quarter:

   $\leq$ 22.5 mrem to any organ

2. During any calendar year:

   $\leq$ 45 mrem to any organ.

c. If the calculated dose from these gaseous effluents exceeds any of above limits in lieu of any other report required by Specification 6.6.2, a report shall be submitted within 30 days from the end of the quarter during which the release occured to the regional NRC Office which includes the following:

1. Cause(s) for exceeding the limit(s);

2. A description of the program of corrective action initiated to: reduce the releases of radioactive materials in gaseous effluents, and to keep these levels of radioactive materials in gaseous effluents in compliance with the above limits or as low as reasonably achievable.

3.10.3 Gaseous Radwaste Treatment

a. The Gaseous Radwaste Treatment System shall be used to reduce the noble gases in gaseous wastes prior to their discharge, if the projected gaseous effluent air dose due to gaseous effluent releases from the site, when averaged over 31 days exceeds 0.6 mrad for gamma radiation and 1.2 mrad for beta radiation.

b. The Ventilation Treatment Exhaust System shall be used to reduce radioactive materials other than noble gases in gaseous waste prior to their discharge when the projected doses due to effluent releases to unrestricted areas when averaged over 31 days would exceed 0.9 mrem to any organ. This does not apply to the Auxiliary Building Exhaust System since it is not "treated" prior to release.

c. If radioactive gaseous waste is discharged without treatment for more than 31 days and in excess of the above limits in lieu of any other report required by specification 6.6.2, a report shall be submitted within 30 days to the regional NRC Office which includes the following:

1. Cause of equipment or subsystems inoperability

2. Corrective action to restore equipment and prevent recurrence

3.10.4 Waste Gas Holdup Tanks

a. The quantity of radioactivity contained in each waste gas holdup tank shall be limited to ≤ 3.8E+05 curies noble gases (considered as Xe-133).

b. Daily, when radioactive materials are being added to a waste gas holdup tank, the quantity of radioactive material contained in the tank being filled shall be determined.

c. If the quantity of radioactive material in any waste gas holdup tank exceeds the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours, reduce the tank contents to within the above limit.

3.10.5 Used Oil Incineration

Used oil, contaminated by radioactivity, may be incinerated in the Station auxiliary boiler provided releases do not exceed one-tenth of one percent (0.1%) of the limits in Technical Specification 3.10.2.b.2.

3.10.6 The provisions of Technical Specifications 3.0 do not apply.

Bases

Specification 3.10.1 is provided to assure that the dose rate at anytime at the exclusion area boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactivity material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the exclusion area boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid rate above background to an infant via the milk animal-milk-infant pathway to ≤ 1500 mrem/year for the nearest milk animal to the plant.

Specification 3.10.3 is provided to implement the requirements of Appendix I, 10 CFR Part 50. The specification provides the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." Surveillance requirements are implemented to meet the requirements of Appendix I. Calculational procedures based on models and data show that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated.

The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, "Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

Equations in the ODCM are provided for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the release of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50.

Restricting the quantity of radioactivity contained in each waste gas holdup tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem.

3.11 SOLID RADIOACTIVE WASTE

Applicability

Applies to the processing and packaging of radioactive solid waste prior to shipment from the site.

Specification

3.11.1 Solid Radioactive Waste

a. The Solid Radwaste System shall be used in accordance with a Process Control Program, for the solidification of wet radioactive wastes. Prior to the shipment of containers of radioactive wastes from the site, radioactive wastes shall be processed and packaged to ensure meeting the requirements of 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations governing the disposal of radioactive wastes.

b. If the requirements of 10CFR Part 20 and/or 10CFR Part 71 are not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

3.11.2 Process Control Program

The Process Control Program shall be used to verify the Solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste to be solidified.

1. Solidification

a. If any test specimen fails to verify Solidification, the Solidification of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative Solidification parameters can be determined in accordance with the Process Control Program, and a subsequent test verifies Solidification. Soldification of the batch may then be resumed using the alternative Solidification parameters determined by the Process Control Program.

2. Process Control Program

b. If the initial test specimen from a batch of waste fails to verify Solidification, the Process Control Program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate Solidification. The Process Control Program shall be modified as required to assure Solidification of subsequent batches of waste.

3.11.3 The provisions of Technical Specification 3.0 do not apply.

Bases

The solid radwaste system will be used whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the Process Control Program may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2.

4.1.3 Required sampling should be performed as detailed in Table 4.1-3.

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in Table 4.1-4.

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days.

Bases

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Substantial calibration shifts within a channel (essentially a channel failure) are revealed during routine checking and testing procedures. Thus, the minimum calibration frequencies set forth are considered acceptable.

Periodic use of the Incore Instrumentation System for power mapping is sufficient to assure that axial and radial power peaks and the peak locations are controlled in accordance with the provisions of the Technical Specifications.

<u>REFERENCE</u>

(1) FSAR, Section 7.2.3.4.

TABLE 4.1-1 (Continued)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building Spray System Logic	NA	MO	NA	
21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	RF	
22. Pressurizer Temperature	ES	NA	RF	
23. Control Rod Absolute Position	ES(1)	NA	RF(2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.
24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel.
25. Core Flood Tanks				
a. Pressure	ES	NA	RF	
b. Level	ES	NA	RF	
26. Pressurizer Level	ES	NA	RF	
27. Letdown Storage Tank Level	DA	NA	RF	
28. Delete				
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	RF	

TABLE 4.1-3

Minimum Sampling Frequency And Analysis Program

Item	Check	Frequency
1. Reactor Coolant	a. Gamma Isotopic Analysis	a. 3 times/week*
	b. Radiochemical Analysis for Sr 89, 90	b. Monthly*
	c. Tritium	c. Monthly*
	d. Gross Beta Activity (1)	d. 3 times/week*
	e. Chemistry (Cl, F and O_2)	e. 5 times/week*
	f. Boron Concentration	f. 2 times/week**
	g. Gross Alpha Activity	g. Monthly*
	h. $\bar{E}$ Determination (2)	h. Semi-annually
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly* and after each makeup
3. Core Flooding Tank	Boron Concentration	Monthly* and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly*** and after each makeup
5. OTSG or Final Feedwater	a. Gross Beta Activity	a. Weekly*
	b. Gamma Isotopic Analysis (3)	
6. Concentrated Boric Acid Tank	Boron Concentration	Twice weekly*

*Not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency.
**Applicable only when fuel is in the reactor.
***Applicable only when fuel is in wet storage in the spent fuel pool.

TABLE 4.1-3 Continued

Minimum Sampling Frequency And Analysis Program

Item	Check	Frequency	Lower Limit of Detection[5] of Lab Analysis for Waste
7. Condensate Test Tank, Condensate Monitoring Tank, Laundry-Hot Shower Tank	a. Principal Gamma Emitters[6] including Dissolved Noble Gases	a. Composite Grab Sample prior to release of each batch[11]	a. Ce-144 and Mo-99 $<5\times10^{-6}$ µCi/ml Other Gamma Nuclides $<5\times10^{-7}$ µCi/ml Dissolved Gases $<10^{-5}$ µCi/ml I-131 $<10^{-6}$ µCi/ml
	b. Radiochemical Analysis Sr 89, 90, Fe-55	b. Quarterly from all composited batches[9]	b. $<5\times10^{-8}$ µCi/ml for Sr's $<10^{-6}$ µCi/ml for Fe-55
	c. Tritium	c. Monthly Composite	c. $<10^{-5}$ µCi/ml
	d. Gross Alpha Activity	d. Monthly Composite	d. $<10^{-7}$ µCi/ml
8. Unit Vent Sampling (Includes Waste Gas Decay Tanks, Reactor Building Purges, Auxiliary Building Ventilation, Spent Fuel Pool Ventilation, Air Ejectors)	a. Iodine Spectrum[4]	a. Continuous monitor, weekly sample[8]	a. $<10^{-10}$ µCi/cc (I-133) $<10^{-12}$ µCi/cc (I-131)
	b. Particulates[4]		
	(1) Ce-144 and Mo-99	(1) Weekly Composite[8]	(1) $<5\times10^{-9}$ µCi/cc
	(2) Other Principal Gamma Emitters[7]	(2) Weekly Composite[8]	(2) $<10^{-10}$ µCi/cc
	(3) Gross Alpha Activity	(3) Monthly, using composite samples of one week	(3) $<10^{-11}$ µCi/cc
	(4) Radiochemical Analysis Sr 89, 90	(4) Quarterly Composite	(4) $<10^{-11}$ µCi/cc
	c. Gases by Principal Gamma Emitters[7]	c. Weekly Grab Sample	c. $<10^{-4}$ µCi/cc
	d. Tritium	d. Weekly Grab Sample	d. $<10^{-6}$ µCi/cc

TABLE 4.1-3 Continued

Minimum Sampling Frequency And Analysis Program

	Item	Check	Frequency	Lower Limit of Detection[5] of Lab Analysis for Waste
8a.	Waste Gas Decay Tank	a. Principal Gamma Emitters[7]	a. Grab Sample prior to release of each batch	a. $<10^{-4}$ µCi/cc (gases) $<10^{-10}$ µCi/cc (particulate and iodines)
		b. Tritium	b. Grab sample prior to release of each batch	b. $<10^{-6}$ µCi/cc
8b.	Reactor Building	a. Principal Gamma Emitters[7]	a. Grab Sample each purge	a. $<10^{-4}$ µCi/cc (gases) $<10^{-10}$ µCi/cc (particulates and iodines)
		b. Tritium	b. Grab Sample each purge	b. $<10^{-6}$ µCi/cc
9.	Keowee Hydro Dam Dilution Flow	Measure Leakage Flow Rate	Annually	
10.	Delete			
11.	Backwash Receiving Tanks	Principle Gamma Emitters including dissolved noble gases	Grab Sample prior to release of each batch	
12.	#3 Chemical Treatment Pond Effluent	a. Principal Gamma Emitters[6]	a. Monthly from composite sample[10]	a. Ce-144 and Mo-99 5×10^{-6} µCi/ml Other Gamma Nuclides $<5\times10^{-7}$ µCi/ml Dissolved Gases $<10^{-5}$ µCi/ml I-131 $<10^{-6}$ µCi/ml
		b. Radiochemical Analysis Sr-89, Sr-90, Fe-55	b. Quarterly from composite sample[9]	b. $<5\times10^{-8}$ µCi/ml for Sr's $<10^{-6}$ µCi/ml for Fe-55
		c. Tritium	c. Monthly from composite sample[10]	c. $<10^{-5}$ µCi/ml
		d. Gross Alpha Activity	d. Monthly from composite sample[10]	d. $<10^{-7}$ µCi/ml

TABLE 4.1-3 NOTES

(1) When radioactivity level is greater than 10 percent of the limits of Specification 3.1.4, the sampling frequency shall be increased to a minimum of once each day.

(2) $\bar{E}$ determination will be started when gross gamma activity analysis indicates greater than 10 µCi/ml and will be redetermined for each 10 µCi/ml increase in gross gamma activity analysis thereafter. A radiochemical analysis for this purpose shall consist of a quantitative measurement of 95 percent of the radionuclides in the reactor coolant with half lives greater than 30 minutes. This is expected to consist of gamma isotopic analysis of the primary coolant, including dissolved gaseous activities, radiochemical analysis for Sr-89 and Sr-90, and tritium analysis.

(3) When gross beta activity increases by a factor of two above background, iodine concentrations will be determined by gamma isotopic analysis and performed thereafter when the gross beta activity increases by 10 percent.

(4) Samples shall be changed at least once per 24 hours and analyses shall be completed within 48 hours after changing (on after removal from sampler).

(5) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

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

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

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

E is the counting efficiency (as counts per disintegration),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.

(6) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

(7) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

(8) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specification 3.10.1, 3.10.2.a and 3.10.2.b.

(9) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

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

TABLE 4.1-3 NOTES (Continued)

(11) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, to assure representative sampling.

TABLE 4.1-4

RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL RESPONSE CHECK(4)	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1.	Liquid Radwaste Effluent Line				
	a. Effluent Line Monitor (1 RIA-33)	*	DA	AN	QU(1)
	b. Effluent Flow Rate Monitor	*	NA	AN	NA
	c. Minimum Flow Device	*	NA	AN	NA
2.	Turbine Building Sump				
	a. Sump Monitor (RIA-54)	DA	MO	AN(3)	QU(2)
	b. Minimum Flow Device	*	NA	AN	NA
3.	Low Pressure Service Water				
	a. Effluent Line Monitor (RIA-35)	DA	MO	AN(3)	QU(1)
	b. Minimum Flow Device	*	NA	AN	NA
4.	#3 Chemical Treatment Pond Composite Sampler	DA	NA	AN	NA
5.	Waste Gas Holdup System				
	a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RIA-37, -38)	*	DA	AN(3)	QU(1)
	b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow)	*	NA	AN	NA
6.	Unit Vent Monitoring				
	a. Noble Gas Activity Monitor (RIA-45)	DA	MO	AN(3)	QU(2)
	b. Iodine Sampler	DA	NA	NA	NA
	c. Particulate Sampler	DA	NA	NA	NA
	d. Effluent Flow Rate Monitor (Unit Vent Flow)	DA	NA	AN	NA
	e. Minimum Flow Device	DA	NA	AN	NA

TABLE 4.1-4 (CONTINUED)

RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL RESPONSE CHECK(4)	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
7. Interim Radwaste Building Ventilation Monitoring				
a. Noble Gas Activity Monitor (RIA-53)	DA	MO	AN(3)	QU(2)
b. Iodine Sampler#	DA	NA	NA	NA
c. Particulate Sampler#	DA	NA	NA	NA
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust)#	DA	NA	AN	NA
e. Minimum Flow Device#	DA	NA	AN	NA
8. Hot Machine Shop				
a. Iodine Sampler#	DA	NA	NA	NA
b. Particulate Sampler#	DA	NA	NA	NA
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust)#	DA	NA	AN	NA
d. Minimum Flow Device#	DA	NA	AN	NA

*During each release via this pathway.
#Effective upon installation of equipment.

Frequency Notation

DA -Daily
QU - Quarterly
MO - Monthly
AN - Annually
PR - Completed prior to each release
NA - Not Applicable

TABLE 4.1-4 (Continued)

TABLE NOTATION

(1) The Channel Functional Test shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

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

2. Circuit failure (downscale only).

(2) The Channel Functional Test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists.

1. Instrument indicates measured levels above the alarm setpoint.

2. Circuit failure (downscale only).

(3) The initial Channel Calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. The standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent Channel Calibration sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement).

(4) The Channel Response Check shall consist of verifying indications during periods of release. Channel Response Check shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

4.11 RADIOLOGICAL ENVIRONMENTAL MONITORING

Applicability

Applies to the surveillance of the station environ for radiation and radioactive materials attributable to station operation and effluent releases.

Specification

4.11.1 Radiological Environmental Monitoring Program

a. The radiological environmental monitoring samples shall be collected in accordance with Table 4.11-1 and shall be analyzed pursuant to the requirements of Tables 4.11-1, 4.11-2 and 4.11-3.

b. If the radiological environmental monitoring program is not conducted as required, a description of the reason for not conducting the program as required and plans to prevent a recurrence shall be included in the Annual Radiological Environmental Operating Report. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

c. If samples become permanently unavailable from any of the required sample locations, the locations from which samples were unavailable may then be deleted from the program provided replacement samples were obtained and added to the environmental monitoring program, if available. These new locations will be identified in the next semi-annual report.

4.11.2 Land Use Census

a. A land use census shall be conducted and shall identify the location of the nearest milk animal and the nearest residence in each of the 16 meterological sectors within a distance of five miles. Broad leaf vegetation sampling shall be performed at the site boundary in the direction sector with the highest $\overline{D/Q}$ in lieu of the garden census.

b. If a land use census identifies a location which yields a calculated dose or dose commitment (via the same exposure pathway) greater than a location from which samples are currently being obtained pursuant to Specification 4.11.1, then the new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. These new locations will be identified in the next semi-annual report.

c. The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

4.11.3 Interlaboratory Comparison Program

a. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the NRC.

b. If these analyses are not performed as required, report corrective actions in the Annual Radiological Environmental Operating Report.

c. A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the methodology and parameters in the ODCM shall be included in the Annual Radiological Environmental Operating Report.

4.11.4 The provisions of Technical Specification 3.0 do not apply.

Bases

The 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 resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.11-1 are considered optimum for routine environmental measurements in industrial laboratories. The specified lower limits of detection correspond to less than the 10CFR50, Appendix I, design objective dose-equivalent to 45 mrem/year for atmospheric releases to the most sensitive organ and individual.

The land use census specification is provided to assure 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.

The requirements for participation in an Interlaboratory Comparison Program is provided to assure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

TABLE 4.11-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis***
1. AIRBORNE			
a. Radioiodine and Particulates	5 Locations	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.	Radioiodine canister. Gamma isotopic analysis for I-131 or each sample. Particulate sampler. Gamma isotopic analysis on each sample.
2. DIRECTION RADIATION			
	40 Locations	Continuous integration with collection at least once per 92 days.	Gamma dose on each dosimeter.
3. WATERBORNE			
a. Surface	2 Locations	Composite* sample collected over a period of $\leq$ 31 days.	Gamma isotopic analysis of each composite sample by location. Tritium analyses of composite sample at least once per 92 days.
b. Drinking	3 Locations	Composite* sample collected over a period of $\leq$ 31 days.	Gross beta and gamma isotopic analysis of each composite sample. Tritium analysis of composite sample at least once per 92 days.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.
**Sample locations are identified in the ODCM.
***Frequency of analysis stated only if different from collection frequency.

TABLE 4.11-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis***
c. Sediment from Shoreline	2 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.
4. INGESTION			
a. Milk	3 Locations	At least once per 15 days when animals are on pasture; at least once per 31 days at other times.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	2 Locations	At least once per 184 days. One sample of each of the following species: 1. Bass 2. Catfish	Gamma isotopic analysis on edible portion.
c. Broad-leaf Vegetation	2 Locations	At least once per 31 days.	Gamma isotopic analysis.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.
**Sample locations are identified in the ODCM.
***Frequency of analysis stated only if different from collection frequency.

TABLE 4.11-2

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m³)	Fish (pCi/kg,wet)	Milk (pCi/l)	Broadleaf Vegetation (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4					
^{3}H	2000					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr	30					
^{95}Nb	15					
^{131}I	15[b]	7×10^{-2}		1	60	
$^{134,\ 137}Cs$	15,18	$5,6 \times 10^{-2}$	130,150	15,18	60,80	150,180
^{140}Ba	60			60		
^{140}La	15			15		

TABLE 4.11-2 (Continued)

TABLE NOTATION

a - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample with 95% probability of detection and with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

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

where

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)

E is the counting efficiency (as counts per disintegration)

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

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide.

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances, may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b - LLD for gamma isotopic analysis for I-131 in drinking water samples. Low level I-131 analysis on drinking water will not be routinely performed because the calculated dose from I-131 in drinking water at all locations is less than 1 mrem per year. Low level I-131 analyses will be performed if abnormal releases occur which could reasonably result in $\geq$ 1 pCi/liter of I-131 in drinking water. For low level analyses of I-131 an LLD of 1 pCi/liter will be achieved.

c - Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.11-2, shall be identified and reported.

TABLE 4.11-3

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/Kg,wet)	Milk (pCi/l)	Broad Leaf Vegetation (pCi/Kg,wet)
H-3	2×10^4*				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131	2**	1.0		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2			3×10^2	

*For drinking water samples. This is 40 CFR Part 141 value.
**If low level I-131 analyses are performed.

4.21 DOSE CALCULATIONS

Applicability

Applies to the projected and cummulative dose contributions from all radioactive liquid and gaseous effluents.

Specification

4.21.1 Dose From All Sources

The annual (calendar year) dose or dose commitment to any Member Of The Public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.9.2.a, 3.10.2.a, or 3.10.2.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 4.21.1 have been exceeded. If such is the case in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.3, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a Member Of The Public from uranium fuel cycle sources, (including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe the levels of radiation and concentration of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Technical Specification 3.0 do not apply.

4.21.2 Dose Due to Liquid Effluents

a. Monthly, cummulative dose contributions from liquid effluents shall be determined in accordance with the Offsite Dose Calculation Manual.

4.21.3 Dose Due to Gaseous Effluents

a. Monthly, cummulative dose contributions from gaseous effluents shall be determined in accordance with the Offsite Dose Calculation Manual.

5 DESIGN FEATURES

5.1 SITE

5.1.1 The Oconee Nuclear Station is approximately eight miles northeast of Seneca, South Carolina. Figure 2-3 of the Oconee FSAR shows the plan of the site. The minimum distance from the reactor center line to the boundary of the exclusion area and to the outer boundary of the low population zone as defined in 10 CFR 100.3, shall be one mile and six miles respectively.

5.1.2 For the purposes of satisfying 10 CFR Part 20, the "Restricted Area," for gaseous release purposes only, is the same as the exclusion area as defined above.

REFERENCE

(1) FSAR, Chapter 2

(2) Technical Specification 3.10.

years of the remaining five years of experience may be fulfilled by academic training, or related technical training on a one-for-one time basis. The Operating Engineer shall hold a Senior Reactor Operator license.

6.1.1.5 Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

6.1.1.6 A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.

6.1.1.7 The two functions of the Shift Technical Advisor, namely accident assessment and operating experience assessment, are fulfilled in the following manner:

a. An experienced SRO, who has been instructed in additional academic subjects, will be assigned on-shift to provide the accident assessment capability.

b. The operating experience assessment function will be provided by the Station Safety Review Group.

6.1.2 Technical Review and Control

6.1.2.1 Activities

a. Procedures required by Technical Specification 6.4 and other procedures which affect station nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared by a qualified individual/organization. Each such procedure, or procedure change, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or procedure change, but who may be from the same organization as the individual/group which prepared the procedure, or procedure change. Such procedures and procedure changes may be approved for temporary use by two members of the station staff, at least one of whom holds a Senior Reactor Operator's License on the unit(s) affected. Procedures and procedure changes shall be approved prior to use or within seven days of receiving temporary approval for use by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the Station Manager.

b. Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

c. Proposed modifications to station nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to station nuclear safety-related structures, systems and components shall be approved prior to implementation by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, or the Maintenance Superintendent, as previously designated by the Station Manager.

d. Individuals responsible for reviews performed in accordance with 6.1.2.1.a, 6.1.2.1.b, and 6.1.2.1.c shall be members of the station supervisory staff, previously designated by the Station Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated station review personnel.

e. Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the Station Manager.

f. Incidents reportable pursuant to Technical Specification 6.6.2.1 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and to the Director of the Nuclear Safety Review Board.

g. The Station Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President, Nuclear Production Department.

h. The station security program, and implementing procedures, shall be reviewed at least annually. Changes determined to be necessary as a result of such review shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and to the Director of the Nuclear Safety Review Board.

i. The station emergency plan, and implementing procedures, shall be reviewed at least annually. Changes determined to be necessary as a result of such review shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and the Director of the Nuclear Safety Review Board.

j. The Station Manager shall assure that an independent fire protection and loss prevention inspection and audit shall be performed annually utilizing qualified off-site personnel and that an inspection and audit by a qualified fire consultant shall be performed at intervals no greater than three years.

k. Unplanned onsite releases of radioactive material to the environs shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production Department, or his designee; and to the Director of the Nuclear Safety Review Board.

l. Proposed changes to the Offsite Dose Calculation Manual (ODCM) shall be prepared by a qualified individual/organization. Each proposed change shall be reviewed by an individual/group other than the individual group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the ODCM shall be approved by the Station Manager prior to implementation.

6.1.2.2 Records

Records of the above activities shall be maintained.

6.1.3 Nuclear Safety Review Board

6.1.3.1 Function

The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations
b. Nuclear Engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Administrative control and quality assurance practices

6.1.3.2 Organization

a. The Director, members and alternate members of the NSRB shall be formally appointed by the Vice President, Nuclear Production Department, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in one or more areas given in 6.1.3.1.

b. The NSRB shall be composed of at least five members, including the Director, Members of the NSRB may be from the Nuclear Production Department, from other departments within the Company or from external to the Company. A maximum of one member of the NSRB may be from the Oconee Nuclear Station staff.

c. Consultants may be utilized by the NSRB to provide expert advice to the NSRB, as determined necessary by the Director of the NSRB.

d. Staff assistance may be provided to the NSRB in order to promote the proper, timely and expeditious performance of its functions.

e. The NSRB shall meet at least once per six months. The period between such meetings shall not exceed eight months.

f. A quorum of the NSRB shall consist of the Director, or his designated alternate, and at least two other NSRB members or alternate members. No more than a minority of the quorum shall have line responsibility for operation of Oconee Nuclear Station.

6.1.3.3 Subjects Requiring Review

The following subjects shall be reported to and reviewed by the NSRB:

a. The safety evaluations for (1) changes to procedures, equipment or systems, and (2) tests or experiments completed under the provisions of 10 CFR 50.59(a)(1) to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes in Technical Specifications or the Facility Operating Licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of station equipment that affect nuclear safety.

g. Incidents that are the subject of non-routine reports submitted to the Commission.

h. Quality Assurance Department audits relating to station operations and actions taken in response to these audits.

6.1.3.4 Audits

Audits of station activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of station operation to provisions contained within the Technical Specifications and applicable facility operating license conditions at least once per year.

b. The performance, training and qualifications of the station staff at least once per year.

c. The results of actions taken to correct deficiencies occurring in equipment, structures, systems or methods of operation that affect nuclear safety at least once per six months.

d. The performance of activities required by the quality assurance program to meet the criteria of Appendix B to 10 CFR 50 at least once per two years.

e. The station emergency plan and implementing procedures at least once per 12 months.

f. The station security plan and implementing procedures at least once per 12 months.

g. Any other area of station operation considered appropriate by the NSRB or the Vice President, Nuclear Production Department.

h. The station fire protection program and implementing procedures at least once per 24 months.

i. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

j. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months.

k. The Process Control Program and implementing procedures for solidification of radioactive wastes at least once per 24 months.

l. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 1.21 Revision 1, June 1974 and Regulatory Guide 4.1 Revision 1, April 1975 at least once per 12 months.

6.1.3.5 Responsibilities and Authorities

a. The NSRB shall report to and advise the Vice President, Nuclear Production Department on those areas of responsibility specified in Specifications 6.1.3.3 and 6.1.3.4.

b. Minutes shall be prepared and forwarded to the Vice President, Nuclear Production Department, and to the Executive Vice President, Power Operations, within 14 days following each formal meeting of the NSRB.

c. Records of activities performed in accordance with Specifications 6.1.3.3 and 6.1.3.4 shall be maintained.

d. Audit reports encompassed by Section 6.1.3.4 shall be forwarded to the Vice President, Nuclear Production Department, and to the Executive Vice President, Power Operations and to the management position responsible for the areas audited within 30 days of completion of each audit.

6.4 STATION OPERATING PROCEDURES

Specification

6.4.1

The station shall be operated and maintained in accordance with approved procedures. Written procedures with appropriate check-off lists and instructions shall be provided for the following conditions:

a. Normal startup, operation, and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility.

b. Refueling operations.

c. Actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety and radiation levels, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

d. Emergency procedures involving potential or actual release of radioactivity.

e. Preventive or corrective maintenance which could affect nuclear safety or radiation exposure to personnel.

f. Station survey following an earthquake.

g. Personnel radiation protection procedures.

h. Operation of radioactive waste management systems.

i. Control of pH in recirculated coolant after loss-of-coolant accident. Procedure shall state that pH will be measured and the addition of appropriate caustic to coolant will commence within 30 minutes after switchover to recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours.

j. Nuclear safety-related periodic test procedures.

k. Long-term emergency core cooling systems. Procedures shall include provision for remote or local operation of system components necessary to establish high and low pressure injection within 15 minutes after a line break.

l. Fire Protection Program implementation.

m. Offsite Dose Calculation Manual implementation.

n. Process Control Program implementation.

6.4.2

A respiratory protective program approved by the Commission shall be in force.

h. By-product material inventory records.

i. Minutes of Nuclear Safety Review Board Meetings.

j. Training records.

k. Test results, in units of microcuries, for leak tests performed pursuant to Specification 4.16.

l. Radioactive liquid effluent, gaseous effluent, and gaseous process monitoring instrumentation alarm/trip setpoints.

6.6 STATION REPORTING REQUIRMENTS

6.6.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator Region II unless otherwise noted.

6.6.1.1 Startup Report

A summary report of unit startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the facility license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly alered the nuclear, thermal, or hydraulic performance of the unit. Startup reports shall be submitted (1) within 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) nine months following initial criticality, whichever occurs first. If a startup report does not cover all three events, i.e., initial criticality, completion of the startup test program and resumption or commencement of commercial power operation supplementary reports shall be submitted at least every three months until all three events are completed.

6.6.1.2 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

6.6.1.3 Personnel Exposure and Monitoring Report

Prior to March 1 of each year, a tabulation shall be submitted to the NRC of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total body dose received from external sources shall be assigned to specific major work functions.

6.6.1.4 Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operating of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the station.

The Radioactive Effluent Release Reports shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter.

The Radioactive Effluent Release Reports shall include an assessment of the radiation doses from radioactive effluents to individuals due to their activities inside the unrestricted area boundary during the report period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports.

The Radioactive Effluent Release Reports shall include the following information for all unplanned releases to unrestricted areas of radioactive materials in gaseous and liquid effluents:

a. A description of the event and equipment involved.

b. Cause(s) for the unplanned release.

c. Actions taken to prevent recurrence.

d. Consequences of the unplanned release.

The Radioactive Effluent Release Reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the station during each calendar quarter. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The annual average meteorological conditions shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual.

The Radioactive Effluent Release Reports shall include the following information for each type of solid waste shipped offsite during the report period:

a. container volume,

b. total curie quantity (determined by measurement or estimate),

c. principal radionuclides (determined by measurement or estimate),

d. type of waste, (e.g., spent resin, compacted dry waste evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. solidification agent (e.g., cement, or other approved agents (media)).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to Unrestricted Areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 4.11.2.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed Member Of The Public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Methods for calculating the dose contribution from liquid and gaseous effluents are given in the ODCM.

6.6.1.5 Radiological Environmental Monitoring

Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses required by Specification 4.11. If harmful effects are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The Annual Radiological Environment Operating Report shall include a summary of the results obtained as part of the required Interlaboratory Comparison Program and in accordance with the ODCM. Alternatively, participants in the EPA cross-check program shall provide the EPA program code designation for the unit.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the radiological environmental samples required by Specification 4.11 taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as practical in a supplementary report.

The initial report shall also include the following: a summary description of the radiological environmental monitoring program including sampling methods for each sample type, size and physical characteristics of each sample type, sample preparation methods, analytical methods, and measuring equipment used; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the result of land use censuses required by Specification 4.11. Subsequent reports shall describe all substantial changes in these aspects.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

b. Thirty-Day Written Reports

The types of events listed below shall be the subject of written reports to the Regional Administrator, Region II, within 30 days of discovery of the event. (Copy to the Director, Office of Management Information and Program Control.)

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or shutdown required by a limiting condition for operation.

(3) Observed inadequacies in the implementation of administrative or procedural controls during operation of a unit which could cause reduction of degree of redundancy provided in the Reactor Protective System or Engineered Safety Feature Systems.

(4) Occurrence of radioactive material contained in liquid or gaseous holdup tanks in excess of that permitted by the limiting condition for operation established in the technical specifications.

(5) An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved.

2. Cause(s) for the unplanned release.

3. Action taken to prevent recurrence.

4. Consequences of the unplanned release.

(6) Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 4.11-3 when averaged over any calendar quarter sampling period. When more than one of the radionuclides in Table 4.11-3 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 4.11-3 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year objectives of Specifications 3.9 and 3.10. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

6.6.2.2 Environmental Monitoring

a. If individual milk samples show I-131 concentrations of 10 picocuries per liter or greater, a plan shall be submitted within one week advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.

b. If milk samples collected over a calendar quarter show average concentrations of 4.8 picocuries per liter or greater, a plan shall be submitted within 30 days advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.

6.6.3 Special Reports

Special reports shall be submitted to the Regional Administrator Region II, 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 specifications:

a. Single Loop Restrictions, Specification 3.1.8

b. Auxiliary Electrical Systems, Specification 3.7

c. Radioactive Liquid Effluents,
   Dose, Specification 3.9.2
   Liquid Waste Treatment, Specification 3.9.3
   Chemical Treatment Ponds, Specification 3.9.4

d. Radioactive Gaseous Effluents,
   Dose, Specification 3.10.2
   Gaseous Radwaste Treatment, Specification 3.10.3

e. Fire Protection and Detection Systems, Specification 3.17

f. Reactor Coolant System Surveillance,
   Inservice Inspection, Specification 4.2.1
   Reactor Vessel Specimen, Specification 4.2.4

g. Reactor Building Surveillance,
   Containment Leakage Tests, Specification 4.4.1

h. Structural Integrity Surveillance,
   Tendon Surveillance, Specification 4.4.2.2

i. Radiological Environmental Monitoring
   Program, Specification 4.11.1
   Land Use Census, Specification 4.11.2

j. Dose Calculations (40 CFR 190), Specification 4.21

6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.8.1

The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications.

The ODCM shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

6.8.2 Any changes to the ODCM shall be made by the following method:

1. Shall be submitted to the Commission by inclusion in the semi-annual Effluent Release Report for the period in which the change(s) was made and shall contain:

    a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

    b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

    c. documentation of the fact that the change has been reviewed in accordance with Technical Specification 6.1.2.1.(1) and found acceptable by the Station Manager.

2. Shall become effective upon review and acceptance by the Station Manager after confirmation of receipt unless otherwise acted upon by the Commission through written notification to the licensee.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 122 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DOCKETS NOS. 50-269, 50-270 AND 50-287

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, the Duke Power Company has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. The Duke Power Company filed this information with the Commission by letter dated February 9, 1983, which requested changes to the Technical Specifications appended to Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station, Units Nos. 1, 2, and 3. The proposed technical specifications update those portions of the technical specifications addressing radioactive waste management and make them consistent with the current, NRC staff positions as expressed in NUREG-0472. These revised technical specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR Parts 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64; and with 10 CFR Part 50, Appendix B.

2.0 BACKGROUND AND DISCUSSION

2.1 Regulations

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

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

10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases 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 NRC staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of certain key values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent technical specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment sytems.

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

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

(1) Instrumentation

(2) Radioactive effluents

(3) Radiological environmental monitoring

(4) Design features

(5) Administrative controls.

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

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

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

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

3.0 EVALUATION

The enclosed report (TER-C5506-103/104/105) was prepared for us by Franklin Research Center (FRC) as part of our technical assistance contract program. Their report provides their technical evaluation of the compliance of the Licensee's submittal with NRC provided criteria. The NRC staff has reviewed this TER and agrees with the evaluation.

In addition, as a result of the August 18-19, 1982 meeting with Duke Power Company, FRC, and the NRC staff, the issue of explosive gas limitations and monitoring was deferred and will be handled as a separate issue following completion of the ongoing Duke and NRC studies of system requirements. Pending completion of these studies the licensee is required, and has committed via discussions on January 5, 1984, to submit a proposed technical specification for NRC review addressing explosive gas limitations and monitoring within 90 days of receipt of the RETS package addressed herein. To date the engineering study portion of Duke's Waste Gas Study has been completed and several options regarding proposed modifications are being evaluated. The Oconee waste gas system presently operates with a hydrogen gas concentration at or below 2% and a presently existing program of monitoring and administrative

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.

controls imposes an allowable upper limit of 3% for hydrogen. If the 3% limit is exceeded (determined by analysis), the licensee activates a nitrogen overblanket feature of the waste gas system to prevent the hydrogen concentration from reaching the 4% regulatory limit. Notwithstanding the above controls, if a postulated explosion of the waste gas system were to occur the total contribution from the system to offsite dose concentrations is predicted to be 0.1 mrem, which is well within Appendix I criteria. Pending receipt and evaluation of the proposed TS, the staff finds this acceptable.

3.1 SAFETY CONCLUSIONS

The proposed radiological effluent technical specifications for Oconee Nuclear Station Units 1, 2, and 3 have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0472 (the Oconee Nuclear Station is comprised of three pressurized water reactors) and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

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

4.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendments would not authorize a significant change in the types, or a significant increase in the amounts, of effluents or in the authorized power level, and that the amendments will not result in any significant environmental impact. Having made these determinations, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Part 51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 16, 1984

The following NRC staff personnel have contributed to this Safety Evaluation: W. Meinke, F. Congel, C. Willis and J. Suermann.

TECHNICAL EVALUATION REPORT

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION IMPLEMENTATION (A-2)

DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNITS 1, 2, AND 3

NRC DOCKET NO. 50-269, 50-270, 50-287

NRC TAC NO. 8119, 8120, 8121

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 4

FRC TASKS 103, 104 105

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May 24, 1983

Prepared by:	Reviewed by:	Approved by:
C. Fernandez	S. Pandey	T P Cardegno
Principal Author		Department Director
Date: May 24, 1983	Date: 5/24/83	Date: 5-24-83

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FOREWORD

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

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of Oconee Nuclear Station Units 1, 2, and 3 with regard to Radiological Effluent Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM).

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

1.2 GENERIC BACKGROUND

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

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

These other issues are specifically stipulated by the following regulations:

- o 10CFR20 [4], "Standards for Protection Against Radiation," Paragraphs 20.105(c), 20.106(g), and 20.405(c) require that nuclear power plants and other licensees comply with 40CFR190 [5], "Environmental Radiation Protection Standards for Nuclear Power Operations," and submit reports to the NRC when the 40CFR190 limits have been or may be exceeded.
- o 10CFR50, Appendix A [6], "General Design Criteria for Nuclear Power Plants," contains Criterion 60 - Control of releases of radioactive materials to the environment; Criterion 63 - Monitoring fuel and waste storage; and Criterion 64 - Monitoring radioactivity releases.
- o 10CFR50, Appendix B [7], establishes the quality assurance required for nuclear power plants.

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

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

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

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

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

1.3 PLANT-SPECIFIC BACKGROUND

In conformance with the 1975 directive [3], Duke Power Company, the Licensee for Oconee Nuclear Station Units 1, 2, and 3, submitted information for 10CFR50, Appendix I Evaluation, dated June 4, 1976 [13].

The RETS and ODCM were addressed in the next submittal by the Licensee, dated March 29, 1979 [14]. The submittal was a response to the November 15-16, 1978 NRC request and followed the format of NUREG-0472 for PWRs. On June 7, 1982, Franklin Research Center (FRC), selected as an independent reviewer, initiated a review and evaluation of the RETS and ODCM submittals. These submittals were compared to the model RETS [1] and to the general provisions for the ODCM [15] which were given to each operating reactor (OR) as guidelines for preparing the RETS and the ODCM. The Licensee's RETS and ODCM submittals were assessed for compliance with the requirements of 10CFR50, Appendix I, and the "General Design Criteria," 10CFR50, Appendix A.

Copies of the draft review reports dated July 30, 1982 [16, 17] were delivered to the NRC and to the Licensee prior to a site visit to the Oconee Nuclear Station in Oconee County, SC. The purpose of the site visit was to resolve questions raised in the draft review reports.

The site visit was conducted on August 18-19, 1982. Discussions were held with Duke Power and Oconee Station personnel to review the RETS and ODCM reports. Agreement was reached on most items dicussed at the meetings, at which time the Licensee made a commitment to resubmit drafts of the RETS and ODCM by November 15, 1982. A trip report was prepared and delivered to the NRC on September 20, 1982 [18]. The report included the resolutions reached, as well as "open items" to be resolved by the NRC with the Licensee.

On February 16, 1983, revised draft copies [19] of the Licensee's RETS were received by the FRC review team and the final review was initiated. Under a cover letter dated February 9, 1983, Duke Power company delivered their final proposed RETS [20] to the NRC. Copies of this submittal were delivered to FRC on February 25, 1983. The proposed RETS was reviewed and evaluated based on the draft model RETS, NUREG-0472, Revision 3 [12], and comments on the proposed RETS were supplied to the NRC on March 16, 1983 [21]. On May 17, 1983, copies [22] of the Licensee's final ODCM Appendix A submittal [23] were received by the FRC RETS review team for evaluation. Appendix A of the ODCM submittal contains Oconee site-specific information and is supplemented by the generic ODCM [24], which applies to all Duke nuclear power plants. The generic ODCM, which has been approved by the NRC staff as part of the McGuire Nuclear Station submittal, and the site-specific Appendix A were included in the ODCM evaluation. The proposed ODCM submittal was evaluated according to the existing guidelines specified by NUREG-0133 [9]. A process control program has not been submitted with the RETS and ODCM submittals.

Details of the RETS review are documented in the comparison copy [25], which contains resolutions on open items received from the NRC [26].

2. REVIEW CRITERIA

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

NUREG-0472, RETS for PWRs

NUREG-0473, RETS for BWRs

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

Twelve essential criteria are given for the RETS and ODCM:

1. All significant releases of radioactivity shall be controlled and monitored.

2. Offsite concentrations of radioactivity shall not exceed the 10CFR20, Appendix B, Table II limits.

3. Offsite radiation doses shall be ALARA.

4. Equipment shall be maintained and used to keep offsite doses ALARA.

5. Radwaste tank inventories shall be limited so that failures will not cause offsite doses exceeding 10CFR20 limits.

6. Hydrogen and/or oxygen concentrations in the waste gas system shall be controlled to prevent explosive mixtures.

7. Wastes shall be processed to shipping and burial ground criteria under a documented program, subject to quality assurance verification.

8. An environmental monitoring program, including a land-use census, shall be implemented.

9. The radwaste management program shall be subject to regular audits and reviews.

10. Procedures for control of liquid and gaseous effluents shall be maintained and followed.

11. Periodic and special reports on environmental monitoring and on releases shall be submitted.

12. Offsite dose calculations shall be performed using documented and approved methods consistent with NRC methodology.

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

Review of the ODCM was based on the following NRC guidelines: Branch Technical Position, "General Content of the Offsite Dose Calculation Manual" [15]; NUREG-0133 [9]; and Regulatory Guide 1.109 [34]. The ODCM format is left to the Licensee and may be simplified by tables and grid printouts.

3. TECHNICAL EVALUATION

3.1 GENERAL DESCRIPTION OF RADIOLOGICAL EFFLUENT SYSTEMS

This section briefly describes the liquid and gaseous radwaste effluent systems, release paths, and control systems installed at Oconee Nuclear Station Units 1, 2, and 3; all three are PWRs.

3.1.1 Radioactive Liquid Effluent

The liquid radwaste treatment system, which is common to all three units, has the capability to collect, treat, store, and dispose of most radioactive liquid wastes. The wastes are collected in sumps and drain tanks in the various buildings and are then transferred to the appropriate tanks in the radwaste building for further treatment, temporary storage, and disposal. The processed liquid wastes are either returned to the chemical and volume control system or released to the environment through the Hartwell Discharge Canal. Slightly radioactive spent powdered resin backflush water from the demineralizer system for the control of secondary water purity is placed in the chemical treatment ponds prior to release. Batches of radioactive liquid waste are discharged to the environment if the concentration of radioactive materials is within the allowable limits.

A diagram of the liquid effluent release paths indicating the location of the liquid effluent monitors is shown in Figure 1. The radioactive liquid wastes originating from the primary drains, high level process drains, and contaminated drains are processed through evaporators and demineralizers prior to release, and the laundry drains are processed through a filter prior to discharge. These wastes are monitored and controlled by liquid effluent radiation monitors (1RIA-33 and 1RIA-34). The radioactive liquid wastes originating from chemical wastes and turbine building floor drains are placed in the No. 3 chemical treatment pond prior to release. The turbine building floor drains are monitored by 1RIA-54 and 3RIA-54 prior to being discharged to the chemical treatment pond and the service water system effluents are monitored by RAI-35. A continuous composite sampler is provided for

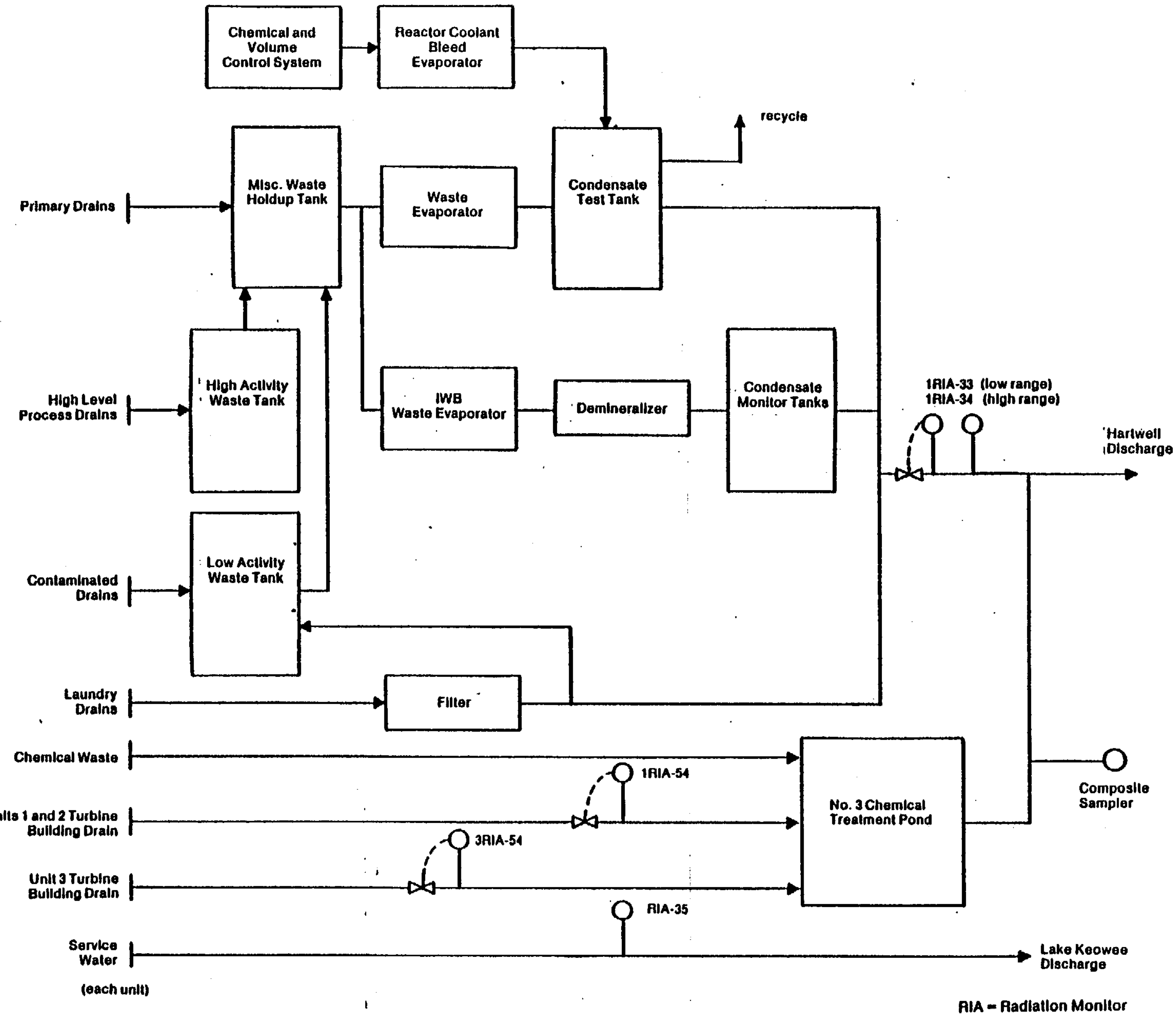

Figure 1. Liquid Radwaste Treatment Systems, Effluent Paths, and Controls for Oconee Nuclear Station Units 1, 2, and 3

discharges from the No. 3 chemical treatment pond. As a safety measure, the liquid radwaste effluent radiation monitor and turbine buidling floor drain monitors are provided with automatic termination of release upon a high concentration alarm signal.

3.1.2 Radioactive Gaseous Effluent

Airborne particulates and gases vented from process equipment and building ventilation exhaust air are the normal sources of radioactive gaseous effluents from the Oconee site. The major source from each unit is the process gas system which contains decay tanks, prefilters, HEPA filters, and charcoal adsorbers to ensure that effluent releases are ALARA.

A diagram of the radioactive gaseous effluents showing the location of effluent radiation monitors and process treatment equipment is shown in Figure 2. Each of the three units has a plant vent (mixed mode model used for dispersion) which is a combined release point for the major sources of gaseous effluents for that unit. Other combined gaseous effluent releases (ground level model used for dispersion) from the site are rooftop releases from the turbine building, interim radwaste builidng, and the hot machine shop. Releases from the interim radwaste building are monitored by 3RIA-53 and releases from the hot machine shop are sampled; releases from the turbine building are unmonitored.

The Unit 1 plant vent is comprised of the following effluent substreams, each of which is equipped with a process radiation monitor as indicated: Unit 1 containment purge (1RIA-49), Unit 1 condenser air ejector (1RIA-40), Units 1 and 2 process gas (1RIA-37), and auxiliary building (1RIA-32). The effluent radiation monitor for the Unit 1 plant vent is 1RIA-45.

The Unit 2 plant vent is comprised of the following effluent substreams, each of which is equipped with a process radiation monitor as indicated: Unit 2 containment purge (2RIA-49), Unit 2 condenser air ejector (2RIA-40), Units 1 and 2 spent fuel pool area (2RIA-41), and auxiliary building (2RIA-32). The effluent radiation monitor for the Unit 2 plant vent is 2RIA-45.

The Unit 3 plant vent is comprised of the following effluent substreams, each of which is equipped with a process radiation monitor as indicated:

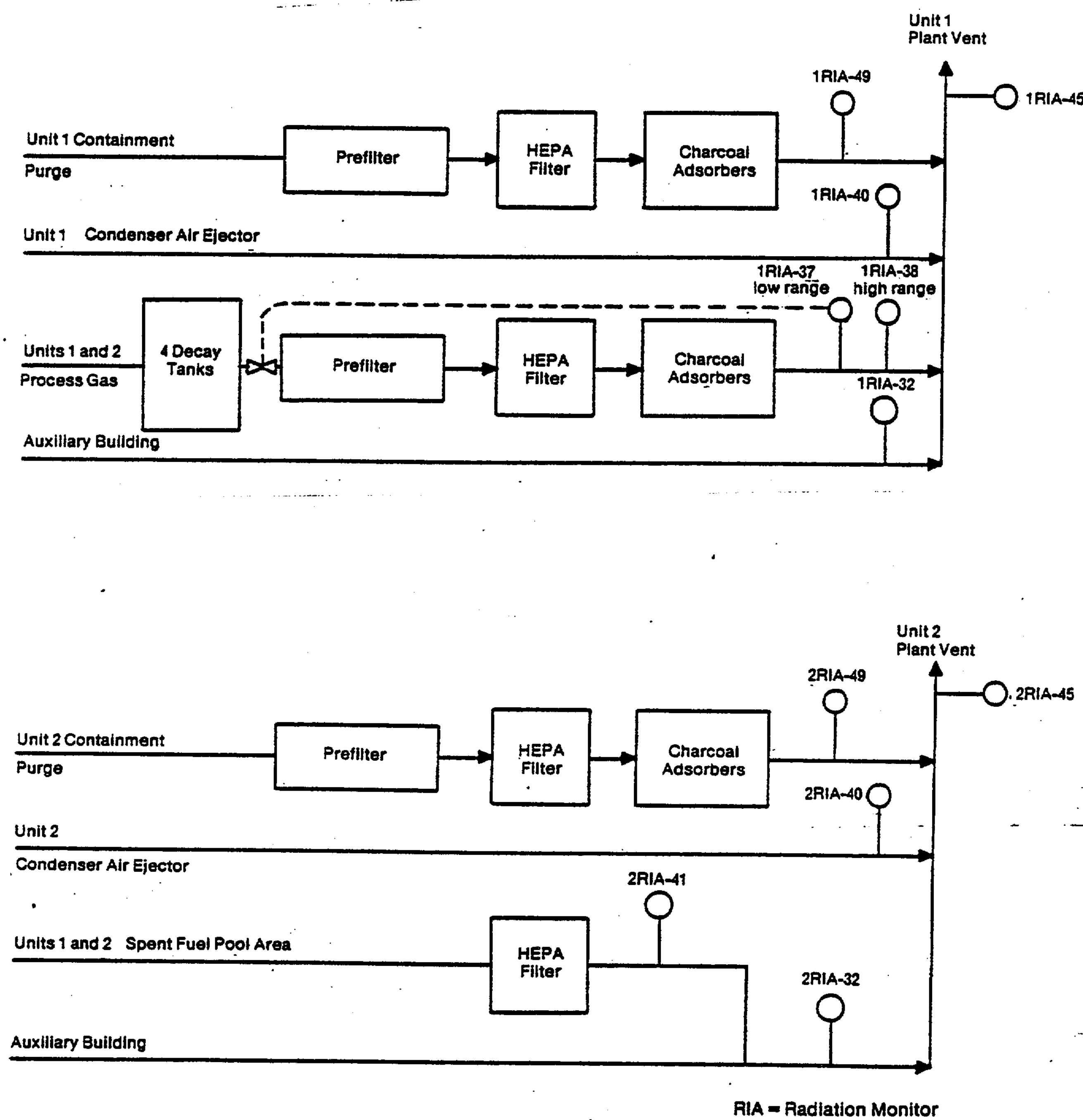

Figure 2. Gaseous Radwaste Treatment Systems, Effluents Paths, and Controls for Oconee Nuclear Station Units 1, 2, and 3

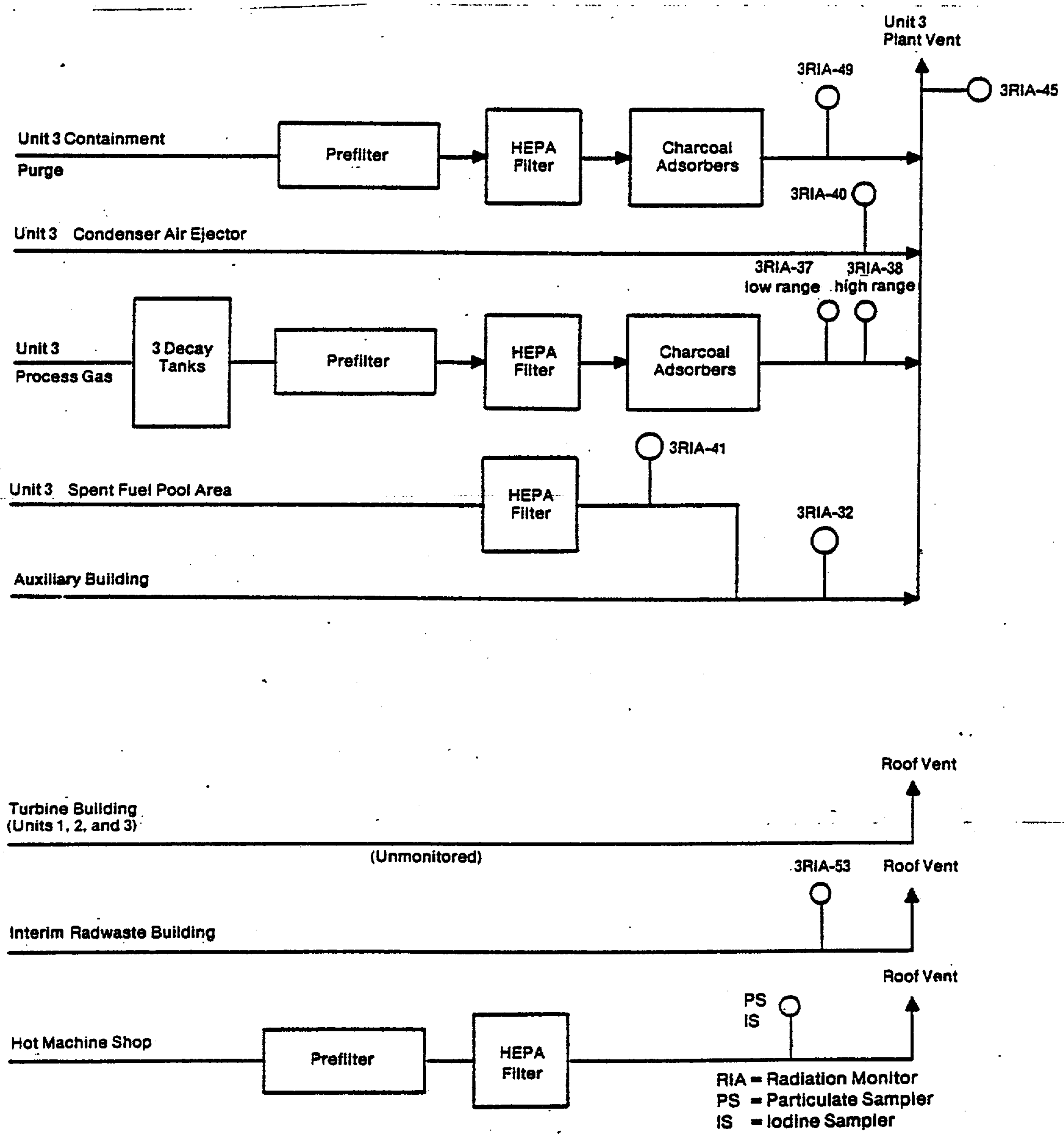

Figure 2 (Cont.)

Unit 3 containment purge (3RIA-49), Unit 3 condenser air ejector (3RIA-40), Unit 3 process gas (3RIA-37), Unit 3 spent fuel pool area (3RIA-41), and the auxiliary building (3RIA-32). The effluent radiation monitor for the Unit 3 plant vent is 3RIA-45.

3.2 RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

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

- o a review of information provided by the Licensee in the 1979 proposed RETS submittal [14]
- o resolution of problem areas in that submittal by means of a site visit [18]
- o review of the Licensee's February 9, 1983 final RETS submittal [20].

3.2.1 Effluent Instrumentation

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

The Licensee has provided radiation monitors for potential liquid or gaseous effluent lines. In addition, automatic isolation is provided for major effluent lines such as the liquid radwaste effluent, the turbine building sump effluent, and the gaseous waste decay tank effluent.

The Licensee has provided gaseous process monitors for all of the major gaseous substreams of the plant vent effluent release points. Effluent radiation monitors have also been provided for releases from the interim radwaste building and particulate and iodine samplers for releases from the hot machine shop. The Licensee has established an ongoing sampling and analysis program for the batch release tanks as well as for continuous releases, as described in the Licensee's final submittal.

Since there are no steam generator blowdown effluent releases directly to the atmosphere, the alternative provisions discussed in NUREG-0133 for the steam generator blowdown vent are not applicable. The Licensee has also established a sampling and analysis program for effluents released from the waste gas storage tank, unit vent sampling, and the reactor building.

The Licensee's proposed RETS submittal on liquid and gaseous effluent monitoring instrumentation has satisfied the provisions set forth in the model RETS and thus meets the intent of NUREG-0472.

3.2.2 Concentration and Dose Rates of Effluents

3.2.2.1 Liquid Effluent Concentration

In Section 3.9.1 of the Licensee's submittal, a commitment is made to maintain the concentration of radioactive liquid effluents released from the site to the unrestricted areas to within 10CFR20 limits, and if the concentration of liquid effluents to the unrestricted area exceeds these limits, it will be restored without delay to a value equal to or less than the MPC values specified in 10CFR20. Both batch and continuous releases are sampled and analyzed periodically in accordance with a sampling and analysis program (Table 4.1-3 of the Licensee's submittal), which meets the intent of NUREG-0472. Technical specifications are given to limit the radioactive inventory of the chemical treatment ponds so as to control the concentration of radioactive liquid effluent releases. These technical specifications are consistent with and meet the intent of NUREG-0472.

3.2.2.2 Gaseous Effluent Dose Rate

In Section 3.10.1 of the Licensee's submittal, a commitment is made to maintain the offsite gaseous dose rate from the site to areas at and beyond the site boundary to within 10CFR20 limits, and if the concentration of gaseous effluents exceeds these limits or the equivalent dose values, it will be restored without delay to a value equal to or less than these limits.

The radioactive gaseous waste sampling and analysis program (Table 4.1-3 of the Licensee's submittal) provides adequate sampling and analysis of the vent discharges, including the substreams, and therefore meets the intent of NUREG-0472.

3.2.3 Offsite Doses from Effluents

The objective of the RETS with regard to offsite doses from effluents is to ensure that offsite doses are kept ALARA, are in compliance with the dose specifications of NUREG-0472, and are in accordance with 10CFR50, Appendix I, and 40CFR190. The Licensee has made a commitment to (1) meet the quarterly and yearly dose limitations for liquid effluents, per Section 3.11.1.2 [12]; (2) restrict the air doses for beta and gamma radiation in unrestricted areas as specified in 10CFR50, Appendix I, Section II.B; and (3) maintain the dose level to the maximally exposed member of the public from releases of radioiodines, tritium, and particulates with half-lives greater than 8 days within the design objectives of 10CFR50, Appendix I, Section II.C. The dose commitment limits proposed by the Licensee for the common Oconee technical specifications are three times the design objectives contained in Appendix I for one unit. The Licensee has stated that there exists no positive means to separate the releases on a per unit basis because of the shared treatment equipment and release points; therefore, the dose commitment limits are given on a per site basis. The Licensee has made a commitment to limit the annual dose to the maximally exposed member of the public due to releases of radioactivity and radiation from uranium fuel cycle sources to within the requirements of 40CFR190. These offsite dose specifications satisfy the intent of NUREG-0472.

3.2.4 Effluent Treatment

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

3.2.5 Tank Inventory Limits

The objective of the RETS with regard to tank inventory limits is to ensure that the rupture of a radwaste tank would not cause offsite doses greater than the limits set in 10CFR20 for non-occupational exposure. The Licensee has put ~~a curie limit of 10 curies~~ on all outside liquid tanks listed in the specifications and has made a commitment to perform surveillance according to the provisions of NUREG-0472. This limit excludes tritium and dissolved or entrained noble gases. For gas storage tanks, a curie limit of 380,000 curies has been set for noble gases which are considered to be represented by xenon-133. The Licensee's commitment to comply with tank inventory limits has satisfied the intent of NUREG-0472.

3.2.6 Explosive Gas Mixtures

The objective of the RETS with regard to explosive gas mixtures is to prevent hydrogen explosions in the waste gas systems. The Licensee has stated that "The resolution of the technical specification regarding explosive gas mixtures in the waste gas system will be delayed until the various modifications proposed as part of the Waste Gas Study can be evaluated and any necessary modifications implemented." In the interim, until modifications are completed, an adequate sampling program should be provided to monitor concentration of explosive gas mixtures in the waste gas system. The omission of technical specifications on explosive gas mixtures does not meet the intent of NUREG-0472.

3.2.7 Solid Radwaste System

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

3.2.8 Radiological Environmental Monitoring Program

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

The Licensee has followed NUREG-0472 guidelines, including the Branch Technical Position dated November 1979 [32], and has provided an adequate number of sample locations for pathways identified.

The 40 thermoluminescent dosimeter (TLD) monitoring stations proposed by the Licensee satisfy the specification of NUREG-0472. The Licensee's method of analysis and maintenance of the monitoring program satisfies the requirements of Appendix I, 10CFR50. The Licensee has also made a commitment to describe the specific sample locations in the ODCM. This meets the intent of NUREG-0472.

The commitments to a yearly land-use census within NUREG-0472 specifications and to an ongoing interlaboratory comparison program equivalent to the model RETS guidelines on environmental monitoring meet the intent of NUREG-0472.

3.2.9 Audits and Reviews

The objective of the RETS with regard to audits and reviews is to ensure that audits and reviews of the radwaste and environmental monitoring programs are properly conducted. The Licensee's administrative structure designates the station safety review group (SSRG) and the nuclear safety review board (NSRB) as the two groups responsible for reviews and audits, respectively. Their responsibilities also include the ODCM, PCP, and QA program. The two committees encompass the total responsibility for reviews and audits as specified in NUREG-0472.

3.2.10 Procedures and Records

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

3.2.11 Reports

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

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

1. Annual radiological environmental operating report. In Section 6.6.1.5 of the Licensee's submittal, a commitment is made to provide an annual radiological environmental operating report that includes summaries, interpretations, and statistical evaluation of the results of the environmental surveillance program. The report also includes the results of land use censuses, and participation in an interlaboratory comparison program specified by Specification 3.12.3 of NUREG-0472.

2. Semiannual radioactive and solid waste release reports. In Section 6.6.1.4 of the Licensee's submittal, a commitment is made to provide semiannual radioactive effluent and solid waste release reports which include a summary of radioactive liquid and gaseous effluents and solid waste released, an assessment of offsite doses, and a list of unplanned releases. Listing of new location for dose calculations identified by the land use census as well as any changes to ODCM are also included in the report.

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

   o exceeding radioactive liquid effluents limits according to:

     Dose, Specification 3.9.2
     Liquid Waste Treatment, Specification 3.9.3
     Chemical Treatment Ponds, Specification 3.9.4

   o exceeding radioactive gaseous effluents limits according to:

     Dose, Specification 3.10.2
     Gaseous Radwaste Treatment, Specification 3.10.3

   o exceeding radiological environmental monitoring limits according to:

     Program, Specification 4.11.1
     Land Use Census, Specification 4.11.2

   o exceeding dose calculation (40CFR190) limits according to Specification 4.21

4. Thirty-Day Written Reports. The License has made a commitment to file a 30-day written report for:

   o an unplanned offsite release of (1) more than 1 curie of radioactive material in liquid effluents, (2) more than 150 curies of noble gas in gaseous effluents, or (3) more than 0.05 curies of radioiodine in gaseous effluents

- o measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values.

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

3.2.12 Implementation of Major Programs

One objective of the administrative controls is to ensure that implementation of major programs such as the PCP, ODCM, and major changes to the radioactive waste treatment system follow appropriate administrative procedures. The Licensee has made a commitment to review, report, and implement major programs such as the ODCM but has not included the PCP. The PCP has been treated as a procedure and not as a major program. This meets the intent of NUREG-0472 on an interim basis [26].

3.3 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

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

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

3.3.1 Evaluation

The Licensee has followed the methodology of NUREG-0133 [9] to determine the alarm and trip setpoints for the liquid and gaseous effluent monitors. To ensure that the MPC, as specified in 10CFR20, will not be exceeded even in the case of simultaneous discharge, the Licensee will administratively control the number of releases occurring at one time and/or apportion the release rate among the units.

The Licensee has demonstrated the method of calculating the radioactive liquid concentration after releasing liquid effluents into the Hartwell discharge canal. The method provides added assurance of compliance with 10CFR20 for liquid releases.

Methods are also included for showing that dose rates at or beyond the site boundary due to noble gases, radioiodines, tritium, and radionuclides in particulate form with half-lives greater than 8 days are in compliance with 10CFR20. In this calculation, the Licensee has considered effluent releases from the various release points; releases from the unit vents are treated as mixed level, and releases from the rooftop vents are treated as ground level. The Licensee has used the highest annual average values of relative concentration (X/Q) and relative deposition (D/Q) to determine the controlling locations. The Licensee, however, has not provided data for (X/Q) and (D/Q) for ground level releases from the roof vents. The Licensee intends to use the maximally exposed individual and the critical organ as the reference receptor. For noble gases, the Licensee has considered the total body dose and the skin dose resulting from gamma and beta radiation, respectively. For radioiodines, tritium, and particulates, the Licensee has considered the inhalation pathway for estimating the doses. The Licensee has demonstrated that the described methods and relevant parameters have followed the conservative approaches provided by NUREG-0133 and Regulatory Guide 1.109.

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

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

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

For radioiodines, tritium, and particulates with half-lives greater than 8 days, the Licensee has provided a method to demonstrate that cumulative doses calculated from the release meet both quarterly and annual design objectives. The Licensee has demonstrated a method of calculating the dose using maximum annual average (X/Q) values for the inhalation pathway and has included (D/Q) values for ingestion pathways. This approach is consistent with the methodology of NUREG-0133.

To comply with the total dose limits specified by 40CRF190, the Licensee has also included the dose contribution from the direct radiation. However, the Licensee concluded that since direct radiation doses are normally less than 0.01 mrem/yr (a negligible amount), direct radiation doses are not calculated routinely.

Using a simplified methodology for gaseous and liquid dose calculations, the Licensee has demonstrated a procedure to project the monthly dose and to ensure that the design objectives for the liquid radwaste system and the gaseous radwaste system are not exceeded. This simplified method considers the critical populations, critical pathways, and critical radionuclides determined for the Duke nuclear stations. The method is consistent with NUREG-0133.

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

The Licensee has provided a description of sampling locations in the ODCM. This description is consistent with the sampling locations specified in the Licensee's RETS Table 4.11-1 on environmental monitoring.

In summary, the Licensee's ODCM submittal addresses the provisions of NUREG-0472 and uses approved methods that are consistent with the methodology and guidance in NUREG-0133; therefore, the ODCM submittal satisfies the intent of these guidelines, except that the Licensee has not provided a set of meteorological data for ground level releases from the rooftop vents.

4. CONCLUSIONS

Table 1 summarizes the results of the final review and evaluation of the submittal from Oconee Nuclear Station Units 1, 2, and 3. The Licensee has made one radiological effluent technical specifications (RETS) submittal for Units 1, 2, and 3 [20], and a thorough review reveals that the RETS are equivalent for all three units. The offsite dose calculation manual (ODCM) was submitted under separate covers for Units 1, 2, and 3 [23, 24].

The following conclusions have been reached:

1. The Licensee's proposed RETS submitted February 9, 1983 meets the intent of the NRC staff's "Standard Radiological Effluent Technical Specifications," NUREG-0472, for Oconee Nuclear Station Units 1, 2, and 3 with the following exceptions:

   a. The submittal of the technical specifications for explosive gas mixture monitoring has been deferred to a later date. In the interim, an adequate sampling program should be provided to monitor concentrations of explosive gas mixtures in the waste gas system.

   b. The implementation of major programs such as the process control program has not been addressed in a manner consistent with NUREG-0472. The process control program is defined as a procedure instead of being defined as a major program.

2. The Licensee's ODCM Appendix A, submitted April 28, 1983 and the generic ODCM, submitted February 28, 1983, use documented and approved methods that are consistent with the criteria of NUREG-0133 and are applicable to Oconee Nuclear Station Units 1, 2, and 3 with the following exception:

   a. The Licensee has not provided meteorological dispersion data (X/Q and D/Q) for ground level releases from the roof vents.

Table 1. Evaluation of Proposed Radiological Effluent Technical Specifications (RETS), Oconee Nuclear Station Units 1, 2, and 3

	Technical Specifications: NRC Staff Std. RETS NUREG-0472 (Section)*	Technical Specifications: Licensee Proposal (Section)	Replaces or Updates Existing Tech. Specs. (Section)	Evaluation
Effluent Instrumentation	3/4.3.3.3.10 3/4.3.3.3.11	3.5.5/4.1	3.5/4.1, 4.15 Appendix A	Meets the intent of NRC criteria
Radioactive Effluents	3/4.11.1.1 3/4.11.2.1	3.9.1/4.1 3.10.1/ 4.1, 4.21	3.9, 3.10 Appendix A	Meets the intent of NRC criteria
Offsite Doses	3/4.11.1.2, 3/4.11.2.2, 3/4.11.2.3, 3/4.11.4	3.9.2/4.21 3.10.2/4.21 3.10.2/4.21 4.21.1	To be added to Appendix A	Meets the intent of NRC criteria
Effluent Treatment	3/4.11.1.3 3/4.11.2.4	3.9.3/4.21 3.10.3/4.21	3.9, 3.10 Appendix A	Meets the intent of NRC criteria
Tank Inventory Limits	3/4.11.1.4 3/4.11.2.6	3.9.5 3.10.4	To be added to Appendix A	Meets the intent of NRC criteria
Explosive Gas Mixtures	3/4.11.2.5B	None	None	Does not meet the intent of NRC criteria
Solid Radioactive Waste	3/4.11.3	3.11	To be added to Appendix A	Meets the intent of NRC criteria
Environmental Monitoring	3/4.12.1	4.11	4.11	Meets the intent of NRC criteria
Audits and Reviews	6.5.1, 6.5.2	6.1.2, 6.1.3	6.1.2, 6.1.3	Meets the intent of NRC criteria
Procedures and Records	6.8, 6.10	6.4, 6.5	6.4, 6.5	Meets the intent of NRC criteria
Reports	6.6	6.6	6.6	Meets the intent of NRC criteria
Implementation of Major Programs	6.13, 6.14, 6.15	6.8	To be added to Appendix A	Meets the intent of NRC criteria in the interim

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

5. REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

13. "Evaluation of Compliance with 10CFR50 Appendix I"
Oconee Nuclear Station Units 1, 2, and 3
June 4, 1976

14. W. O. Parker (Duke)
Letter to H. Denton (USNRC) with RETS and ODCM attached
March 29, 1979

15. General Contents of the Offsite Dose Calculation Manual," Rev. 1
NRC, 1979

16. "Radiological Effluent Technical Specification Implementation, Comparison of Plant and Model RETS"
Franklin Research Center, Draft dated July 30, 1982

17. "Radiological Effluent Technical Specification Implementation, Technical Review of Plant Offsite Dose Calculation Manual"
Franklin Research Center, Draft dated July 30, 1982

18. Trip Report to Oconee Nuclear Power Station
Trip Date: August 18-19, 1982
C. Fernandez/S. Pandey (FRC) to F. Congel/C. Willis/W. Meinke (NRC)
September 20, 1982

19. Draft Final Oconee RETS Submittal
Tranmitted to S. Pandey (FRC) from W. Meinke (NRC)
February 16, 1983

20. H. B. Tucker (Duke)
Letter of Transmittal to NRC
Subject: Proposed Radiological Effluent Technical Specifications for the Oconee Nuclear Station
February 9, 1983

21. Informal Technical Communication
from C. Fernandez/S. Pandey (FRC) to W. Meinke (NRC)
"RETS Review Questions"
March 16, 1983

22. Draft Final Oconee ODCM Appendix A Submittal
Transmitted to S. Pandey (FRC) from W. Meinke (NRC)
May 17, 1983

23. H. B. Tucker (Duke)
Letter of Transmittal to NRC
Subject: Proposed Oconee ODCM Appendix A Submittal
April 28, 1983

24. H. B. Tucker (Duke)
Letter of Transmittal to NRC
Subject: Revised Offsite Dose Calculation Manual
February 28, 1983

25. "Comparison of Specification NUREG-0472, Radiological Effluent Technical Specifications for PWRs, vs. Licensee Final Submittal of Radiological Effluent Technical Specifications, dated February 9, 1983, for Oconee Nuclear Power Station"
Franklin Research Center
May 2, 1983

26. W. Meinke (NRC)
Memo to S. Pandey (FRC)
"NRC Resolutions on Final RETS Submittal"
April 26, 1983

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

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

29. C. Willis and F. Congel (NRC)
"Status of NRC Radiological Effluent Technical Specification Activities"
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November 4, 1981

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

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

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

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