

PROPOSED CHANGE

RADIOLOGICAL EFFLUENT AND ENVIRONMENTAL
TECHNICAL SPECIFICATIONS

APPENDIX A

TO

PROVISIONAL OPERATING LICENSE DPR-16

OYSTER CREEK NUCLEAR GENERATING STATION
GPU NUCLEAR CORPORATION

DOCKET NO. 50-219

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1.18 PROTECTION INSTRUMENTATION LOGIC DEFINITIONSA. INSTRUMENT CHANNEL

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

B. TRIP SYSTEM

A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, a containment spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).

1.19 INSTRUMENTATION OF SURVEILLANCE DEFINITIONSA. CHANNEL CHECK

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

B. CHANNEL TEST

Injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action.

C. CHANNEL CALIBRATION

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip.

D. Source Check

A SOURCE CHECK is the qualitative assessment of channel response when the channel sensor is exposed to a source of radioactivity.

1.20 FDSAR

Oyster Creek Unit No. 1 Facility Description and Safety Analysis Report as amended by revised pages and figure changes contained in Amendments 14, 31, and 45.

1.26 FRACTION OF LIMITING POWER DENSITY (FLPD)

The fraction of limiting power density is the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

1.27 MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The maximum fraction of limiting power density is the highest value existing in the core of the fraction of limiting power density (FLPD).

1.28 FRACTION OF RATED POWER (FRP)

The fraction of rated power is the ratio of core thermal power to rated thermal power.

1.29 TOP OF ACTIVE FUEL (TAF) - 353.3 inches above vessel zero.1.30 PROCESS CONTROL PLAN

The PROCESS CONTROL PLAN shall generally describe the essential operational controls and surveillance checks for processing wet radioactive waste in order to provide reasonable assurance of compliance with class B or C stability requirements of 10 CFR Part 61.56 (b) before disposal.

1.31 AUGMENTED OFFGAS SYSTEM (AOG)

The AUGMENTED OFFGAS SYSTEM is a system designed and installed to holdup and/or process radioactive gases from the main condenser offgas system for the purpose of reducing the radioactive material content of the gases before release to the environs.

1.32 MEMBER OF THE PUBLIC

A MEMBER OF THE PUBLIC is a person who is not occupationally associated with GPU Nuclear and who does not normally frequent the Oyster Creek Nuclear Generating Station site. The category does not include contractors, contractor employees, vendors, or persons who enter the site to make deliveries, to service equipment, work on the site, or for other purposes associated with plant functions.

1.33 OFFSITE DOSE CALCULATION MANUAL

An OFFSITE DOSE CALCULATION MANUAL (ODCM) states the methodology and parameters to be used in the calculation of radiation doses offsite due to radioactive gaseous and liquid effluents and in the calculation of radioactive gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints.

1.34 PURGE

PURGE or PURGING is the controlled process of discharging air or gas from a confinement and replacing it with air or gas.

1.35 EXCLUSION AREA

EXCLUSION AREA is defined in 10 CFR Part 100.3(2). As used in these technical specifications, the Exclusion Area boundary is the perimeter line around the OCNGS beyond which the land is neither owned, leased, nor otherwise subject to control by GPU (ref. ODCM Figure 1-1). The area outside the Exclusion Area is termed OFFSITE.

1.36 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable				Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels per Operable Trip Systems	Action Required*
		Shutdown	Refuel	Startup	Run			
2. Low-Low-Low Reactor Water Level	>4'8" above top of active fuel	X (v)	X (v)	X (v)	X	2	2	See note h
3. AC Voltage	NA			X (v)	X	2	2	Prevent auto depressurization on loss of AC power. See note i
H. Isolation Condenser Isolation								
1. High Flow Steam Line	<20 psig ΔP	X (s)	X (s)	X	X	2	2	Isolate Affected isolation condenser comply with Spec. 3.8 See Note dd
2. High Flow Con- densate Line	<27" H ₂ O	X (s)	X (s)	X	X	2	2	
I. Offgas System Isolation								
1. High Radiation In Offgas Line (e)	<2.1/E Ci/sec	X (s)	X (s)	X	X	1(ee)	2(ee)	See Note ff
J. Reactor Building Isolation and Standby Gas Treatment System Initiation								
1. High Radiation Reactor Bldg. Operation Floor	<100 Mr/Hr	X (w)	X (w)		X	1	1	Isolate Reactor Bldg. and initiate Standby Gas Treatment System Manual Surveillance for not more than 24 hours (total for all Instruments under J) in any 30 day period.
2. Reactor Bldg. Ventilation Exhaust	≤ 17 Mr/Hr	X (w)	X (w)	X	X	1	1	
3. High Drywell Pressure	≤ 2 psig	X (u)	X (u)	X	X	1(k)	2(k)	
4. Low Low Reactor Water Level	> 7'2" above top of active fuel	X	X	X	X	1	2	

TABLE 3.1.1 (Cont'd)

- * Action required when minimum conditions for operation are not satisfied. Also permissible to trip inoperable trip system. When necessary to conduct tests and calibrations, one channel may be made inoperable for up to one hour per month without tripping its trip system.
- ** See Specification 2.3 for Limiting Safety System Settings.

NOTES:

- a. Permissible to bypass, with control rod block, for reactor protection system reset in refuel mode.
- b. Permissible to bypass below 800 psia in refuel and startup modes.
- c. One (1) APRM in each operable trip system may be bypassed or inoperable provided the requirements of Specification 3.1.C and 3.10.C are satisfied. Two APRM's in the same quadrant shall not be concurrently bypassed except as noted below or permitted by note.

Any one APRM may be removed from service for up to one hour for test or calibration without inserting trips in its trip system only if the remaining operable APRM's meet the requirements of Specification 3.1.B.1 and no control rods are moved outward during the calibration or test. During this short period, the requirements of Specifications 3.1.B.2, 3.1.C and 3.10.C need not be met.
- d. The (IRM) shall be inserted and operable until the APRM's are operable and reading at least 2/150 full scale.
- e. Offgas system isolation trip set at $\leq 2.1/\bar{E}$ Ci/sec where \bar{E} = average gamma energy from noble gas in offgas (Mev). Air ejector isolation valve closure time delay shall not exceed 15 minutes.
- f. Unless SRM chambers are fully inserted.
- g. Not applicable when IRM on lowest range.
- h. One instrument channel in each trip system may be inoperable provided the circuit which it operates in the trip system is placed in a simulated tripped condition. If repairs cannot be completed within 72 hours the reactor shall be placed in the cold shutdown condition. If more than one instrument channel in any trip system becomes inoperable the reactor shall be placed in the cold shutdown condition. Relief valve controllers shall not be bypassed for more than 3 hours (total time for all controllers) in any 30-day period and only one relief valve controller may be bypassed at a time.

- i. The interlock is not required during the start-up test program and demonstration of plant electrical output but shall be provided following these actions.
- j. Not required below 40 of turbine rated steam flow.
- k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (see Specification 4.5), provided that the plant is in the cold shutdown condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.
- l. Bypassed in IRM Ranges 8, 9, and 10.
- m. There is one time delay relay associated with each of two pumps.
- n. One time delay relay per pump must be operable.
- o. There are two time delay relays associated with each of two pumps. One timer per pump is for sequence starting (SK1A, SK2A) and one timer per pump is for tripping the pump circuit breaker (SK7A, SK8A).
- p. Two time delay relays per pump must be operable.
- q. Manual initiation of affected component can be accomplished after the automatic load sequencing is completed.
- r. Time delay starts after closing of containment spray pump circuit breaker.
- s. These functions not required to be operable with the reactor temperature less than 212°F and the vessel head removed or vented.
- t. These functions may be inoperable or bypassed when corresponding portions in the same core spray system logic train are inoperable per Specification 3.4.A.
- u. These functions not required to be operable when primary containment integrity is not required to be maintained.
- v. These functions not required to be operable when the ADS is not required to be operable.
- w. These functions must be operable only when irradiated fuel is in the fuel pool or reactor vessel and secondary containment integrity is required per Specification 3.5.B.

- y. The number of operable channels may be reduced to 2 per Specification 3.9-E and F.
- z. The bypass function to permit scram reset in the shutdown or refuel mode with control rod block must be operable in this mode.
- aa. Pump circuit breakers will be tripped in 10 seconds \pm 15% during a LOCA by relays SK7A and SK8A.
- bb. Pump circuit breakers will trip instantaneously during a LOCA.
- cc. Only applicable during startup mode while operating in IRM range 10.
- dd. If an Isolation condenser inlet (steam side) isolation valve becomes or is made Inoperable in the open position during the run mode comply with Specification 3.8.E. If an AC motor-operated outlet (condensate return) isolation valve becomes or is made inoperable in the open position during the run mode comply with Specification 3.8.F.
- ee. Instrument shall be operable during main condenser air ejector operation except that a channel may be taken out-of-service for the purpose of a check, calibration, test, or maintenance without declaring it to be inoperable.
- ff. With no channel OPERABLE, main condenser offgas may be released to the environment for as long as 72 hours provided the stack radioactive noble gas monitor is OPERABLE. Otherwise, be in at least SHUTDOWN CONDITION within 24 hours.

3.6 Radioactive Effluents

Applicability: Applies to the radioactive effluents of the facility.

Objective: To assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that the radioactive concentrations of any material released is kept as low as is reasonably achievable and, in any event, within the limits of 10 CFR Part 20.106 and 40 CFR Part 190.10(a).

Specification

3.6.A. Reactor Coolant Radioactivity

The specific activity of the radioiodine in the reactor coolant shall be limited to less than or equal to 0.2 $\mu\text{Ci/gram}$ DOSE EQUIVALENT (D.E.) I-131.

Limiting Condition for Operation

a. Coolant Chemistry

1. Whenever an isotopic analysis shows reactor coolant activity exceeds 0.2 $\mu\text{Ci/gram}$ DOSE EQUIVALENT I-131, additional analyses shall be done at least 6 times within 48 hours.
2. If the reactor coolant activity is greater than 0.2 $\mu\text{Ci/gram}$ D.E. I-131 but equal to or less than 4 $\mu\text{Ci/gram}$ D.E. I-131 for more than 48 hours during one continuous time interval, an orderly shutdown shall be immediately initiated.
3. If an initial sample of the reactor coolant activity is greater than 4 $\mu\text{Ci/gram}$ D.E. I-131, a second sample shall be taken and analyzed within 8 hours. If the second sample indicates that the reactor coolant activity is greater than 4 $\mu\text{Ci/gram}$ D.E. I-131, an orderly shutdown shall be immediately initiated. If the second sample indicates that the reactor coolant activity is less than or equal to 4 $\mu\text{Ci/gram}$ D.E. I-131, the statement 3.6.A.a.2 shall apply.

3.6.B Liquid Radwaste Treatment

Applicability: To liquid radwaste batches before discharge as aqueous effluent.

1. Any untreated batch of liquid radwaste shall be treated (in appropriate liquid radwaste treatment equipment) before discharge as aqueous effluent when the radioactivity concentration, exclusive of tritium and dissolved noble gases, in the batch exceeds 0.001 $\mu\text{Ci/ml}$.
2. When radioactive liquid waste is discharged without treatment and in excess of the above limit, in lieu of any other report, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.3 a Special Report that includes the following information:
 - a. Identification of any inoperable equipment or subsystems, and the reason for the inoperability.
 - b. Action(s) taken to restore the inoperable equipment to OPERABLE status, and a
 - c. Summary description of action(s) taken to prevent a recurrence.
3. Specifications 3.0.A and 3.0.B do not apply.

3.6.C

Radioactive Liquid Storage

Applicability: Applies at all times to specified outdoor tanks used to store radioactive liquids.

1. The quantity of radioactive material, excluding tritium, noble gases, and radionuclides having half-lives shorter than three days, contained in any of the following outdoor tanks shall not exceed 10.0 curies:
 - a. Waste Surge Tank, HP-T-3.
 - b. Condensate Storage Tank
2. In the event the quantity of radioactive material in any of the tanks named exceeds 10.0 curies, begin treatment as soon as reasonably achievable, continue it until the total quantity of radioactive material in the tank is 10 curies or less, and describe the reason for exceeding the limit in the next Semiannual Radioactive Effluent Release Report.
3. Specifications 3.0.A and 3.0.B do not apply.

3.6.D

Condenser Offgas Treatment

Applicability: Whenever the main condenser air ejector system is in operation except during startup or shutdown with reactor power less than 40 percent of rated. In addition, the Augmented Offgas System need not be in operation during end of cycle coast-down periods when the system can no longer function due to low offgas flow.

1. Every reasonable effort shall be made to maintain and operate charcoal absorbers in the Augmented Offgas System to treat radioactive gas from the main condenser air ejector.
2. If gaseous effluent is released without treatment for more than 30 consecutive days and either Specification 3.6.L or 3.6.M is exceeded, in lieu of any other report, submit a Special Report pursuant to Specification 6.9.3 to the NRC within 30 days from the end of the quarter during which the release occurred which includes the following information:
 - a. Identification of the inoperable equipment or subsystem and the reason for inoperability; and
 - b. Action(s) taken to restore the inoperable equipment to OPERABLE status and to prevent a recurrence.

3.6.E

Main Condenser Offgas Radioactivity

1. The gross radioactivity in noble gases discharged from the main condenser air ejector shall not exceed $0.21/\bar{E}$ Ci/sec after the holdup line where \bar{E} is the average gamma energy (Mev per atomic transformation).
2. In the event Specification 3.6.E.1 is exceeded, reduce the discharge rate below the limit within 72 hours or be in at least SHUTDOWN CONDITION within the following 12 hours.

3.6.F

Condenser Offgas Hydrogen Concentration

1. The concentration of hydrogen in the Augmented Offgas System (AOG) downstream of the recombiner during AOG operation shall not exceed 4 percent by volume.
2. In the event the hydrogen concentration downstream of a recombiner exceeds 4 percent by volume, the concentration shall be reduced to less than 4 percent within 48 hours.
3. In the event the hydrogen concentration is not reduced to <4 percent within 48 hours, be in at least SHUTDOWN CONDITION or within the limit within the following 24 hours.

3.6.G

Not used.

3.6.H

Not used.

3.6.I

Radioactivity Concentration in Liquid Effluent

1. The concentration of radioactive material, other than noble gases, in liquid effluent in the discharge canal at the Route 9 bridge (see ODCM Figure 1-1) shall not exceed the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2.
2. The concentration of noble gases dissolved or entrained in liquid effluent in the discharge canal at the Route 9 bridge shall not exceed 2×10^{-4} microcuries/ milliliter.
3. In the event the concentration of radioactive material in liquid effluent released into the Offsite area beyond the Route 9 bridge exceeds either the concentration limit in 3.6.I.1 or 3.6.I.2, reduce the release rate without delay to bring the concentration below the limit.
4. The provisions of Specification 6.9.2 are not applicable.

3.6.J

Limit on Dose Due to Liquid Effluent

1. The dose to a MEMBER OF THE PUBLIC due to radioactive material in liquid effluents beyond the outside of the EXCLUSION AREA shall not exceed:

1.5 mrem to the total body during any calendar quarter,

5 mrem to any body organ during any calendar quarter,
3 mrem to the total body during any calendar year, or
10 mrem to any body organ during any calendar year,

2. When the calculated dose from the release of radioactive materials in liquid effluents exceeds any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days from the end of the quarter during which the release occurred, pursuant to Specification 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken and/or will be taken.
3. The provisions of Specifications 3.0.A and 3.0.B are not applicable.

3.6.K

Dose Rate Due to Gaseous Effluent

1. The dose equivalent rate outside of the EXCLUSION AREA (see ODCM Figure 1-1) due to radioactive noble gas in gaseous effluent shall not exceed 500 mrem/year to the total body or 3000 mrem/year to the skin.
2. The dose equivalent rate outside of the EXCLUSION AREA due to H-3, I-131, I-133, and to radioactive material in particulate having half-lives of 8 days or more in gaseous effluents shall not exceed 1500 mrem/year to any body organ when the dose rate due to H-3, Sr-89, Sr-90, and alpha-emitting radionuclides is averaged over no more than 3 months and the dose rate due to other radionuclides is averaged over no more than 31 days.
3. In the event the dose equivalent rate exceeds any of the limits in 3.6.K.1 or 3.6.K.2, decrease the release rate without delay to comply with the limit. If the gaseous effluent release rate cannot be reduced to meet the limits, the reactor shall be in at least SHUTDOWN CONDITION within 48 hours unless corrective actions have been completed and the release rate restored to below the limit.
4. The provisions of Specification 6.9.2 do not apply.

3.6.L

Air Dose Due to Noble Gas in Gaseous Effluent

1. The air dose outside of the EXCLUSION AREA (see ODCM Figure 1-1) due to noble gas released in gaseous effluent shall not exceed:
5 mrad/calendar quarter due to gamma radiation,
10 mrad/calendar quarter due to beta radiation,
10 mrad/calendar year due to gamma radiation, or
20 mrad/calendar year due to beta radiation.
2. If the calculated air dose due to noble gas released in gaseous effluent exceeds any limit in Specification 3.6.L.1, prepare and submit a Special Report to the Commission which identifies the cause(s) for exceeding the limit and describes the corrective action taken. The Special Report shall be pursuant to Specification 6.9.3, shall be in lieu of any other report, and shall be submitted to the Commission within 30 days from the end of the quarter during which the release occurred.
3. The provisions of Specifications 3.0.A and 3.0.B do not apply.

3.6.M

Dose Due to Radioiodine and Particulates in Gaseous Effluent

1. The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, and from radionuclides in particulate form having half-lives of 8 days or more in gaseous effluents, outside of the EXCLUSION AREA shall not exceed 7.5 mrem to any body organ per calendar quarter or 15 mrem to any body organ per calendar year.
2. When the calculated dose from I-131, I-133, and from radionuclides in particulate form having half-lives of 8 days or more in gaseous effluent exceeds any limit in Specification 3.6.M.1, prepare and submit a Special Report to the Commission which identifies the cause(s) for exceeding the limit and describes the corrective action taken. The Special Report shall be pursuant to Specification 6.9.3, shall be in lieu of any other report, and shall be submitted to the Commission within 30 days from the end of the quarter during which the release occurred.
3. The provisions of Specifications 3.0.A and 3.0.B do not apply.

3.6.N

Annual Total Dose Due to Radioactive Effluents

1. The annual dose to a MEMBER OF THE PUBLIC due to radiation and radioactive material in effluents from the OCNGS outside of the EXCLUSION AREA shall not exceed 75 mrem to his thyroid or 25 mrem to his total body or to any other organ.
2. In the event the calculated dose due to radioactive material released in liquid or gaseous effluent exceeds twice the limits of Specification 3.6.J.1, 3.6.L.1, or 3.6.M.1, perform an assessment of compliance with Specification 3.6.N.1 in accordance with methodology in the ODCM.
3. In the event an assessment shows Specification 3.6.N.1 to have been exceeded, prepare and submit a Special Report to the Commission within 30 days, pursuant to Specification 6.9.3 and in lieu of any other report. The report shall include information specified in 10 CFR 20.405(c). If the condition causing the limit(s) to be exceeded has not been corrected, the Special Report may also state a request for a variance in accordance with the provisions of 40 CFR Part 190. In that event, the request is timely and a variance is granted until NRC action on the request is complete.
4. The provisions of Specification 3.0.A and 3.0.B do not apply.

Basis: 3.6.A

During the integrated assessment of the SEP Topic XV-16 (Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment), the NRC staff concluded based on their analysis that reactor coolant activity limits at Oyster Creek should be maintained within the limits imposed on new operating reactors, that is, within the limits of the Standard Technical Specifications (STS) for General Electric Boiling Water Reactors (NUREG-0123). This is necessary to limit plant operation with potentially significant amounts of failed fuel so that the radiological consequences of events that do not damage fuel but do involve a release of reactor coolant to the environment will be low. Adoption of the STS reactor coolant activity limit ($0.2\mu\text{Ci}/\text{gram D.E. I-131}$) would not result in calculated thyroid dose within the limit (30 Rem) specified in current licensing criteria (Standard Review Plan 15.6.2) for the exclusion area. However, the NRC staff concluded that an adoption of the BWR STS (NUREG-0123) limit of $0.2\mu\text{Ci}/\text{gram D.E. I-131}$ for reactor coolant activity is sufficient to ensure that the radiological consequences to the environment from a failure of small lines are acceptably low (NUREG-0822). The calculated dose is based on a double-ended break of a one-inch instrument line, upstream of the outboard isolation valve. Altogether, there are 59 such lines at Oyster Creek which extend from the reactor vessel through the primary containment to instruments and gauges in the reactor building. Because of a lack of inboard isolation valves, the discharge from the break would continue until the reactor vessel is depressurized and action can be taken to plug the leak. The model also assumes that 38% of the discharge flashes to steam and is released to the environment without credit for Standby Gas Treatment System filtration or plateout in the reactor building (SRP Section 6.2.3 guidance in Branch Technical Position 6-3).

In addition, an iodine spike occurs as a result of the reactor shutdown or depressurization of the primary system. The spike is modelled by increasing the equilibrium iodine release rate from the fuel by a factor of 500. These assumptions are in accordance with the Standard Review Plan.

Basis: 3.6.B

This specification implements the requirements of 10 CFR Part 50.36a related to operation of radioactive waste treatment equipment to keep radioactive material in effluents to unrestricted areas as low as reasonably achievable. Radioactive liquid wastes

generated at the OCNCS are controlled on a batch basis with each batch processed by a method appropriate for the quality and concentration of material present. Below 0.001 $\mu\text{Ci/ml}$, it is not cost-beneficial to treat a batch of aqueous waste for the purpose of reducing potential radiation exposure offsite. Hence specification 3.6.B implements 10 CFR Part 50 Appendix I provisions for cost-beneficial treatment of radioactive liquid waste before release in effluent. Each batch of radioactive liquid waste is sampled and analyzed for radioactivity before release to the discharge canal so that an appropriate discharge rate can be determined, accounting for dilution by condenser cooling water and/or canal flow.

Basis: 3.6.C

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2 in the canal at the Route 9 bridge.

Retaining radioactive liquids on-site in order to permit systematic and appropriate processing is consistent with maintaining radioactive discharges to the environment as low as practicable. Limiting the contents of each outside tank to 10 curies or less assures that even if the contents of a tank were released onto the ground and drained into the discharge canal, the potential dose to a member of the public is estimated to be less than 1 percent of the 500 mrem/year limit to the total body of a member of the public and only 1 percent of the corresponding 1500 mrem/year standard for a single organ.

In the highly unlikely event that every outside tank named in Specification 3.6.C were to contain 10 curies and the contents of all were to spill into the discharge canal, the potential dose to a member of the public is estimated to be only about 2 percent of the 500 mrem/year limit to the total body and about 6 percent of the corresponding 1500 mrem/year standard.

Basis: 3.6.D

The operability of the AUGMENTED OFFGAS SYSTEM (AOG) charcoal absorber ensures that they will be available for use whenever main condenser offgases require treatment prior to release to the environment and implements 10 CFR Part 50 Appendix A Criterion 60.

The appropriate portions of this system provide reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". A Special Report is required in the event the Augmented Offgas System charcoal absorber is not operated and a concentration or dose exceeds a relevant limit offsite.

Basis: 3.6.E Some radioactive material is released from the plant under controlled conditions as part of the normal operation of the facility. Other radioactive material not normally intended for release could be inadvertently released in the event of an accident. Therefore, limits in 10 CFR Part 20 apply to releases during normal operation and limits in 10 CFR Part 100 apply to accidental releases.

Radioactive gases from the reactor pass through the steam lines to the turbine and then to the main condenser where they are extracted by the air ejector, passed through holdup piping and released via the plant stack preferably after treatment in the Augmented Offgas System. Radioactive materials release limits for the plant stack have been calculated using meteorological data from a 400 ft. tower at the plant site. The analysis of these on-site meteorological data shows that a release of radioactive gases after 30 minutes holdup in the offgas system of 0.3 Ci/sec., would not result in a whole body radiation dose exceeding the 10 CFR 20 value of 0.5 rem per year.

The Holland plume rise model with no correction factor was used in the calculation of the effect of momentum and buoyancy of a continuously emitted plume.

Independent dose calculations for several locations offsite were made by the AEC staff from onsite meteorological data developed by the licensee and diffusion assumptions appropriate to the site. The procedure followed is described in Section 7-5.2.5 of "Meteorology and Atomic Energy - 1968", equation 7.63 being used. The results of these calculations were equivalent to those generated by the licensee provided the average gamma energy per disintegration for the assumed noble gas mixture with a 30 minute holdup is 0.7 MeV per disintegration. Based on these calculations, a maximum release rate limit of gross activity, except for iodines and particulates with half lives longer than eight days, in the amount of $0.21/\bar{E}$ curies per second will not result in offsite annual doses in excess of the limits specified in 10 CFR Part 20. The \bar{E} determination need consider only the average gamma energy per disintegration since the controlling whole body is due to the cloud passage over the receptor and not cloud submersion, in which the beta dose could be additive.

The above discussion does not take into consideration the reduction in release rate afforded by operation of the Augmented Offgas System.

- Basis: 3.6.F The purpose of Specification 3.6.F is to require that the concentration of potentially explosive gas mixtures in the Augmented Offgas System be maintained below the flammability limit of hydrogen in air, although the AOG is designed to withstand a hydrogen explosion. Specification 3.6.F applies to the hydrogen concentration downstream of a recombiner during AOG operation. The AOG has redundant recombiners so that the recombiner in use can be isolated and purged with air in the event hydrogen in it exceeds the specified limit.
- Basis: 3.6.I The purpose of Specification 3.6.I is to require that concentrations of radioactive material in aqueous effluents to OFFSITE areas comply with 10 CFR Part 20.106. The concentration limit for dissolved or entrained noble gas is based on assumed exposure by immersion in water containing Xe-135 (assumed to be the critical radioactive noble gas). The concentration limit of noble gases is applied independently of the limit for other radionuclides because the exposure pathway is separate.
- Basis: 3.6.J The purpose of Specification 3.6.J is to require compliance with 10 CFR Part 50 Appendix I, Section IV.A to assure that radioactive material in liquid effluent is kept as low as is reasonably achievable and to permit operating flexibility under unusual operating conditions.
- Basis: 3.6.K The purpose of Specification 3.6.K is to require that concentrations of radioactive material in airborne effluents to OFFSITE areas comply with 10 CFR Part 20.106. The occupancy of a Member of the Public who may from time to time be within the EXCLUSION AREA is taken to be sufficiently low to compensate for any increase in atmospheric concentration within the area, thereby causing the exposure of those Members of the Public to be less than the equivalent annual limit on radiation exposure to a Member of the Public incurred Offsite.
- Basis: 3.6.L
3.6.M The purpose of Specifications 3.6.L and 3.6.M is to require compliance with 10 CFR Part 50 Appendix I, Section IV.A and to provide operating flexibility under unusual operating conditions as permitted in 10 CFR Part 50.36a. Assessment of compliance is implemented by calculational methods specified in the ODCM provided by the Surveillance Requirements. The ODCM methodology provides for assessing compliance with dose limits at or beyond the Site Boundary based on either historical average atmospheric conditions or conditions averaged over the period of interest.

The occupancy of a Member of the Public who may from time to time be within the EXCLUSION AREA is taken to be sufficiently low to compensate for any increase in atmospheric concentration within the area, thereby causing the exposure of those Members of the Public to be less than the equivalent radiation exposure to a Member of the Public incurred Offsite.

Basis: 3.6.N Annual Total Dose

Specifications 3.6.N and 4.6.N implement the provisions of 40 CFR Part 190.10a as incorporated into 10 CFR Part 20.405(c). It is unlikely that the dose to any Member of the Public will exceed the limits of 40 CFR Part 190.102 as long as the exposure remains within the limits of specifications 3.6.J, 3.6.L, and 3.6.M. Only exposure to radioactive effluent and direct gamma radiation from the OCNGS is considered in assessing compliance because the dose to a Member of the Public from fuel cycle sources other than the OCNGS is negligible since there is no other fuel cycle facility within ten miles.

3.14 Solid Radioactive Waste

Applicability: Processing wet radioactive waste destined for disposal by burial in land as class B or C waste.

Objective: To provide reasonable assurance that the applicable waste satisfies stability requirements for classes B and C wastes stated in 10 CFR Part 61.56(b) before disposal.

Specification: 3.14.A Wet radioactive waste destined for disposal by land burial as class B or C waste shall be processed and/or contained in accordance with a Process Control Plan to meet appropriate waste stability characteristics required by 10 CFR Part 61.56(b) before being shipped to a disposal facility.

3.14.B The provisions of 3.0.A, 3.0.B and 6.9.2 do not apply.

Basis: 3.14 10 CFR Part 61.55 defines classes B and C radioactive wastes which, among others, must meet certain requirements on waste form to insure stability after disposal. 10 CFR 61.56 states the requirements which apply to characteristics of radioactive waste being disposed of by land burial. Specification 3.14 and 4.14 apply essential operational controls and surveillance checks to processing Class B or C wet radioactive waste and/or placing it into a high integrity container in order to provide reasonable assurance that it satisfies the class B or C stability requirement 10 CFR Part 61.56(b) before disposal.

A contractor who has an NRC approved Process Control Plan may perform the wet radioactive waste processing function.

3.15 Radioactive Effluent Monitoring Instrumentation

Applicability: Applies to instrumentation whose function is to monitor aqueous and airborne radioactive effluents from the Station.

Objective: To assure that instrumentation to monitor radioactive effluents is OPERABLE when effluent is discharged or that means of measuring effluent is provided.

Specification

3.15.A Liquid Effluent Instrumentation

1. The radioactive liquid effluent monitoring channels listed in Table 3.15.1 shall be OPERABLE with their alarm/trip setpoints set to initiate alarm/trip in the event the limit of Specification 3.6.1.1 is exceeded.
2. The alarm or trip setpoint of these channels shall be determined and set in accordance with the method described in the OCDM.
3. When a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required by the above specification, 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, or provide for manual initiation of the alarm/trip function(s).
4. When less than the minimum number of radioactive liquid effluent monitoring instrumentation channels are OPERABLE, take the ACTION shown in Table 3.15.1. Make every reasonable effort to restore the instrument 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.

The Provisions of Specifications 3.0.A, 3.0.B, and 6.9.2 are not applicable.

3.15.B

Gaseous Effluent Instrumentation

1. Each radioactive effluent noble gas monitoring channel listed in Table 3.15.2 shall be OPERABLE with its alarm setpoint set to cause automatic alarm in the event a limit of Specification 3.6.K.1 is exceeded.
2. The alarm or trip setpoint of these channels shall be determined and set in accordance with the method described in the ODCM.
3. When a radioactive effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required by Specification 3.15.B.1 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.
4. When less than the minimum number of radioactive gaseous monitoring instrumentation channels are OPERABLE, take the ACTION shown in Table 3.15.2. Make every reasonable effort to restore the instrument 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.
5. The Provisions of Specifications 3.0.A, 3.0.B, and 6.9.2 are not applicable.

TABLE 3.15.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum ^a Channels Operable	Applicability	Action
1. GROSS RADIOACTIVITY MONITORS			
a. Liquid Radwaste Effluent Line	1	b	110
b. Reactor Building Service Water System Effluent Line	1	b	112
c. Turbine Building Sump No.1-5 (to be installed by 6/85)	1	b	114
2. FLOW MEASUREMENT DEVICES			
a. Liquid Radwaste Effluent Line	1	b	113

Table 3.15.1 Notations

- a. Instrument channels shall be OPERABLE and in service as indicated except that a channel may be taken out-of-service for the purpose of a check, calibration, test, or maintenance without declaring the channel to be inoperable.
- b. During releases via this pathway.

ACTION 110 With no channel OPERABLE, effluent may be released provided that:

1. At least two independent samples are taken, one prior to discharge and one near the completion of discharge. These will be analyzed per Specification 4.6.I.1, and
2. Before initiating a release Qualified personnel must determine the acceptable release rate and proper discharge valving and other qualified personnel independently verify that the release rate and discharge valving are acceptable.

Otherwise, suspend release of radioactive effluent via this pathway.

ACTION 112 With no channel OPERABLE, effluent releases via this pathway may continue provided that, at least once per 24 hours during the release, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-6} microcuries/ml.

ACTION 113 With no channel OPERABLE effluent releases via the affected pathway may continue provided the flow is estimated with the pump curve or change in tank level, at least once per batch during a release.

ACTION 114 With no channel operable effluent may be released provided that before initiating a release:

1. A sample is taken and analyzed in accordance with Specification 4.6.I.1.
2. Qualified personnel determine and independently verify the acceptable release rate.

TABLE 3.15.2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum ^a Channels Operable	Essential Function	Applicability	Action
1. Main Condenser Offgas Treatment System Recombiner Effluent Hydrogen Monitor	2 ^d	Monitor hydrogen concentration	c	125
2. Stack Monitoring System				
a. Radioactive Noble Gas Monitor	1	Monitor activity concentration, alarm	b,e	124
b. Iodine Sampler	1	Collect sample	b,e	127
c. Particulate Sampler	1	Collect sample	b,e	127
d. Effluent Flow Measuring Device	1	Measure air flow	b	122
e. Sampler Flow Measuring Device	1	Measure air flow	b	128
3. Turbine Building Ventilation Monitoring System				
a. Radioactive Noble Gas Monitor	1	Monitor activity concentration	b	123
b. Iodine Sampler	1	Collect sample	b	127
c. Particulate Sampler	1	Collect sample	b	127
d. Effluent Flow Measuring Device	1	Measure air flow	b	122
e. Sampler Flow Measuring Device	1	Measure air flow	b	128
4. Offgas Building Exhaust Ventilation Monitoring System				
a. Radioactive Noble Gas Monitor	1	Monitor activity concentration	b	123
b. Iodine Sampler	1	Collect sample	b	127
c. Particulate Sampler	1	Collect sample	b	127
d. Sampler Flow Measuring Device	1	Measure air flow	b	128

TABLE 3.15.2 NOTATIONS

- a. Channels shall be OPERABLE and in service as indicated except that a channel may be taken out of service for the purpose of a check, calibration, test, maintenance or sample media change without declaring the channel to be inoperable.
- b. During releases via this pathway.
- c. During Augmented Offgas Treatment System Operation.
- d. One hydrogen and one temperature sensor.
- e. Monitor/sampler or an alternate shall be OPERABLE to monitor/sample Stack effluent whenever the drywell is being purged.

ACTION 121 With no channel OPERABLE, main condenser offgas may be released to the environment for up to 72 hours provided:
a. the offgas treatment system is not bypassed, and
b. the main stack radioactive noble gas monitor is OPERABLE.
Otherwise, be in the SHUTDOWN CONDITION within 24 hours.

ACTION 122 With no channel OPERABLE, effluent releases via this pathway may continue provided the flow rate is estimated whenever the exhaust fan combination in this system is changed.

ACTION 123 With no channel OPERABLE, effluent releases via this pathway may continue provided a grab sample is taken at least once per 48 hours and is analyzed for gross radioactivity within 24 hours thereafter or provided an alternate monitoring system with local display is utilized.

ACTION 124 With no channel OPERABLE, effluent releases via this pathway may continue provided a grab sample is taken at least once per 8 hours and analyzed for gross radioactivity within 24 hours or provided an alternate monitoring system with local display is utilized. Drywell purge is permitted only when the radioactive noble gas monitor is operating.

ACTION 125 With one channel OPERABLE, operation of the main condenser offgas treatment system may continue provided a recombiner temperature sensing instrument is operable. When only one of the types of instruments, i.e., hydrogen monitor or temperature monitor, is operable, the offgas treatment system may be operated provided a gas sample is collected at least once per day and is analyzed for hydrogen within four hours. In the event neither a hydrogen monitor nor a recombiner temperature sensing instrument is operable when required, the Offgas Treatment System may be operated provided a gas sample is collected at least once per 8 hours and analyzed within the following 4 hours.

ACTION 127 With no channel OPERABLE, effluent releases via this pathway may continue provided the required sampling is initiated with auxiliary sampling equipment as soon as reasonable after discovery of inoperable primary sampler(s).

ACTION 128 With no channel OPERABLE, effluent releases via the sampled pathway may continue provided the sampler air flow is estimated and recorded at least once per day.

Basis: 3.15.A

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 of liquid effluents. The use of this instrumentation is consistent with the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50. Radioactivity monitors on the liquid radwaste effluent line and in the Turbine Building Sump No. 1-5 initiate a trip to stop the effluent discharge pump when the trip setpoint is exceeded. The reactor service water system discharge line radioactivity monitor initiates an alarm in the reactor control room when the alarm setpoint is exceeded.

The alarm/trip setpoint for each of these instruments is calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.106.

Basis: 3.15.B

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during releases of gaseous effluents. The alarm/trip setpoint for each of the noble gas monitors is calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.106. This instrumentation also includes provisions for monitoring hydrogen below the explosive level in the offgas system downstream from the recombiner. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50. The offgas hydrogen monitor and the radioactive gas monitors for the condenser air ejector offgas, the stack effluent, and the offgas building exhaust ventilation have alarms which report in the reactor control room. The offgas hydrogen monitor initiates a bypass of the Augmented Offgas System in the event the setpoint is exceeded.

The Stack and the Turbine Building Exhaust Ventilation effluent air are monitored by a radioactive gaseous effluent monitoring system (RAGEMS). It can sample, and analyze effluent for radioactive particulates, iodine, and noble gases. It can measure the concentration of radioactive noble gases and can identify the principal noble gas radionuclides. In the event the capability to identify noble gas radionuclides on-line becomes inoperable, a grab sample of the effluent air will be taken at least once per month and analyzed for the principal noble gas radionuclides (Reference Table 4.6.2).

The gross gamma activity concentration of noble gas in Stack effluent is displayed in the reactor control room. That channel also causes an alarm in the reactor control room in the event a high activity concentration setpoint is exceeded. Low flow of sampled Stack effluent through the RAGEMS would also cause an alarm in the reactor control room.

Data collected by the RAGEMS are displayed separately via a control computer. In the event the control computer and or control room display fail to function or are voluntarily taken out of service, sampling particulates and iodine can be continued; the noble gas activity concentration, the sample flow, and the sampled stream flow (stack and turbine building vent) can be observed at a display located near the monitoring instrument (in which case the affected channel continues to serve its essential function and remains OPERABLE. If the noble gas activity concentration display and the associated alarm become inoperable in either the reactor control room or in the locale of the monitor, then OCNGS will perform the appropriate action according to Table 3.15.2 or will provide an auxiliary monitoring system. This permits OCNGS to retain the GE instrumentation as an alternate noble gas monitor or to substitute another noble gas monitor, particulate sampler, and iodine sampler in the event the RAGEMS and/or the NaI noble gas monitor which displays in the reactor control room becomes inoperable. The alternate sampling and/or monitoring system would be subject to the requirements stated in Specifications 3.15.B and 4.15.B.

Purging the drywell to purify its atmosphere may discharge most of the air and gases in a brief time. Hence, the drywell is purged only when the radioactive noble gas monitor in the stack monitoring system is operating in order to ensure measurement of radioactive gases discharged.

Frequently, the drywell is vented to control its pressure. But since the release rate is comparatively small, the effluent is monitored as usual and the extra requirement in Table 3.15.2 Action 124 that is applied during purging is not imposed during drywell venting.

TABLE 4.1.1 (CONTINUED)

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

<u>Instrument Channel</u>	<u>Check</u>	<u>Calibrate</u>	<u>Test</u>	<u>Remarks (Applies to Test and Calibration)</u>
14. High Radiation in Reactor Building				
Operating Floor	1/s	1/3 mo	1/wk	Using gamma source for calibration
Ventilation Exhaust	1/s	1/3 mo	1/wk	Using gamma source for calibration
15. High Radiation on Air Ejector Off-Gas	1/s 1/mo	1/3 mo Each refueling outage	1/wk Each refueling outage	Using built-in calibration equipment Channel check Source check Calibration according to established station calibration procedures. Note a
16. IRM Level	N/A	Each Startup	N/A	
IRM Scram	*	*	*	Using built-in calibration equipment
17. IRM Blocks	N/A	Prior to Startup and Shutdown	Prior to Startup and Shutdown	Upscale and downscale
18. Condenser Low Vacuum	N/A	Each Refueling outage	Each Refueling outage	

* Calibrate prior to startup and normal shutdown and thereafter check 1/s and test 1/wk until no longer required.

Legend N/A = Not Applicable; 1/s = Once per shift; 1/d = Once per day; 1/3d = Once per 3 days; 1/wk = Once per week; 1/3 mo = Once every 3 months; 1/mo = once per month

TABLE 4.1.1 NOTATIONS

The following notes are only for Item 15 of Table 4.1.1:

A channel may be taken out of service for the purpose of a check, calibration, test or maintenance without declaring the channel to be inoperable.

a. 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) Instrument indicates a downscale failure.
- 3) Instrument controls not set in operate mode.
- 4) Instrument electrical power loss.

TABLE 4.1.2

MINIMUM TEST FREQUENCIES FOR TRIP SYSTEMS

<u>Trip System</u>	<u>Minimum Test Frequency</u>
1) <u>Dual Channel (Scram)</u>	Same as for respective instrumentation in Table 4.1.1
2) <u>Rod Block</u>	Same as for respective instrumentation in Table 4.1.1
3) <u>Containment Spray</u> , each trip system, one at a time	1/3 mo and each refueling outage
4) <u>Automatic Depressurization</u> , each trip system, one at a time	Each refueling outage
5) <u>MSIV Closure</u> , each closure logic circuit independently (1 valve at a time)	Each refueling outage
6) <u>Core Spray</u> , each trip system, one at a time	1/3 mo and each refueling outage
7) <u>Primary Containment Isolation</u> , each closure circuit independently (1 valve at a time)	Each refueling outage
8) <u>Refueling Interlocks</u>	Prior to each refueling operation
9) <u>Isolation Condenser Actuation and Isolation</u> , each trip circuit independently (1 valve at a time)	Each refueling outage
10) <u>Reactor Building Isolation and SGTS Initiation</u>	Same as for respective instrumentation in Table 4.1.1
11) <u>Condenser Vacuum Pump Isolation</u>	Prior to each startup
12) <u>Air Ejector Offgas Line Isolation</u>	Each refueling outage

4.6 RADIOACTIVE EFFLUENT

Applicability: Applies to monitoring of gaseous and liquid radioactive effluents of the Station during release of effluents via the monitored pathway(s). Each Surveillance Requirement applies whenever the corresponding Specification is applicable unless otherwise stated in an individual Surveillance Requirement. Surveillance Requirements do not have to be performed on inoperable equipment.

Objective To measure radioactive effluents adequately to verify that radioactive effluents are as low as is reasonably achievable and within the limit of 10 CFR Part 20.106.

Specification: 4.6.A Reactor Coolant

Applicability: During reactor Power Operation.

1. Reactor coolant shall be sampled and analyzed at least once every 72 hours and analyzed for DOSE EQUIVALENT I-131.

4.6.B (See 4.6.I)

4.6.C Radioactive Liquid Storage

1. Liquids contained in the following tanks shall be sampled and analyzed for radioactivity at least once per 7 days when radioactive liquid is being added to the tank:

- a. Waste Surge Tank, HP-T-3;
- b. Condensate Storage Tank.

4.6.D Main Condenser Offgas Treatment

1. Operation of the Offgas System charcoal absorbers shall be verified by verifying the AOG System bypass valve (V-7-31) alignment or alignment indication closed at least once every 12 hours whenever the main condenser air ejector is operating.

4.6.E Main Condenser Offgas Radioactivity

1. The gross radioactivity in fission gases discharged from the main condenser air ejector shall be measured by sampling and analyzing the gases
 - a. at least once per month, and
 - b. When the reactor is operating at more than 40 percent of rated power, within 4 hours after an increase in the fission gas release via the air ejector of more than 50 percent, as indicated by the Condenser Air Ejector Offgas Radioactivity Monitor after factoring out increase(s) due to change(s) in thermal power level.

4.6.F Condenser Offgas Hydrogen Concentration

The concentration of hydrogen in offgases downstream of the recombiner in the Offgas System shall be monitored with hydrogen monitoring instrumentation as described in Table 3.15.2.

4.6.G Not Used

4.6.H Not Used

4.6.I Radioactivity Concentration in Liquid Effluent

1. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program in Table 4.6-1.

Alternately, pre-release analysis of batch(es) of radioactive liquid waste may be by gross beta or gamma counting provided a maximum concentration limit of 1×10^{-7} $\mu\text{Ci/ml}$ in the discharge canal at the Route 9 bridge is applied.

2. The alarm or trip setpoint of each radioactivity monitoring channel in Table 3.1.5.1 shall be determined on the basis of sampling and analyses results obtained according to Table 4.6.1 and setpoint method in the ODCM and set to alarm or trip before exceeding the limits of Specification 3.6.1.

4.6.J Dose Due to Liquid Effluent

An assessment shall be performed in accordance with the ODCM at least once a month to determine compliance with Specification 3.6.J.

4.6.K Dose Rate Due to Gaseous Effluent

Radioactive noble gaseous effluent shall be monitored in accordance with Specification 3.15.B. Radioactive noble gas monitors named in Table 3.15.2 shall be set to cause automatic alarm when the monitor setpoint, determined as specified in the ODCM, is exceeded.

4.6.L (not used)

4.6.M Dose Due to Radioiodine and Particulates in Gaseous Effluent

An assessment shall be performed in accordance with the ODCM at least once every month to verify that the cumulative dose from I-131, I-133, and radionuclides in particulate form with half-lives of 8 days or more released in gaseous effluent does not exceed any limit in Specification 3.6.M.1.

4.6.N Annual Total Dose Due to Radioactive Effluents

The cumulative dose to a Member of the Public offsite contributed by liquid and gaseous effluents shall be evaluated in accordance with the methodology and parameters in the ODCM at least once per year.

TABLE 4.6.1
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection ^a (LLD) ($\mu\text{Ci/ml}$)
A. Batch Waste Release Tanks	P	PC	Principal Gamma Emitters	1×10^{-6}
	Each Batch ^b	Each Batch	I-131	1×10^{-6}
	P	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	One Batch/M ^b			
	P	M	H-3	1×10^{-5}
	Each Batch ^b	Composited ^d	Gross Alpha	1×10^{-7}
	P	Q	Sr-89, Sr-90	5×10^{-8}
	Each Batch ^b	Composited ^d	Fe-55	1×10^{-6}
B. Reactor Building Service Water Effluent and Turbine Bldg. Sump No. 1-5	W	W	Principal Gamma Emitters	1×10^{-6}
	Grab Sample ^e		I-131	1×10^{-6}
	(note f)	M	H-3	1×10^{-5}
		Composite ^g	Gross Alpha	1×10^{-7}
	(note f)	Q	Sr-89, Sr-90	5×10^{-8}
		Composite ^g	Fe-55	1×10^{-6}

Legend:

S = once per 12 hours, D = once per 24 hours, W = once per 7 days,
M = once per 31 days, Q = once per 92 days, SA = once per 184 days,
R = once per 18 months, S/U = before each reactor startup,
P = completed before each release, N.A. = Not Applicable

TABLE 4.6.1 NOTATIONS

- a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

The LLD is applicable to the capability of a measurement system under typical conditions and not as a limit for the measurement of a particular sample in the radioactive liquid waste sampling and analysis program.

For a particular measurement system, which may include radiochemical separation:

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

where

LLD is the lower limit of detection as defined above (microcuries 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 (counts per minute)

E is the counting efficiency (counts per disintegration)

V is the sample size (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 the end of sample collection and the time of counting.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions with typical values of E, V, Y, and λt for the radionuclides Mn-54, Fe-59, Co-58, Co-60, Zn-65, Ce-141, Cs-134, Cs-137; an LLD of 1×10^{-5} $\mu\text{Ci/ml}$ should typically be achieved for Mo-99 and Ce-144.

Occasionally background fluctuations, interfering radionuclides, or other uncontrollable circumstances may render these LLDs unachievable.

TABLE 4.6.1 NOTATIONS

When calculating the LLD for a radionuclide determined by gamma ray spectrometry, the background may include the typical contributions of other radionuclides normally present in the samples. The background count rate of a Ge(Li) detector is determined from background counts that are determined to be within the full width of the specific energy band used for the quantitative analysis for that radionuclide.

The principal gamma emitters for which the LLD specification will apply are exclusively 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. The LLD for Mo-99 and Ce-144 is 1×10^{-5} $\mu\text{Ci/ml}$ whereas the LLD for other principal gamma emitters is 1×10^{-6} $\mu\text{Ci/ml}$. Nuclides which are below the LLD for the analysis should not be reported.

- b. A batch release is the discharge of liquid wastes of a discrete volume. Before sampling for analysis, each batch should be thoroughly mixed.
- c. In the event a gross radioactivity analysis is performed in lieu of an isotopic analysis before a batch is discharged, a sample shall be analyzed for principal gamma emitters afterward.
- d. 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.
- e. Analysis may be performed after release.
- f. In the event a grab sample contains more than 1×10^{-6} $\mu\text{Ci/ml}$ of I-131 and principal gamma emitters or in the event the effluent radioactivity monitor indicates more than 1×10^{-6} $\mu\text{Ci/ml}$ radioactivity in effluent, as applicable, sample Reactor Building Service Water Effluent daily or sample Turbine Building Sump No. 1-5 each discharge until analysis confirms the activity concentration in the effluent does not exceed 1×10^{-6} $\mu\text{Ci/ml}$.
- g. A composite sample is produced by combining grab samples, each having a defined volume, collected routinely from the sump or stream being sampled.

TABLE 4.6.2

4.6-7

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection ^a (LLD) (Ci/ml)
Stack	Q Grab Sample ^f	Q	H-3	1×10^{-6}
Stack; Turbine Building Exhaust Vent; Offgas Building Vent	M Grab Sample ^{c,d,f}	M	Principal Gamma Emitters ^b (noble gases)	1×10^{-4}
	Continuous ^f	W Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous ^f	W Particulate Sample	Principal Gamma Emitters ^b (particulates)	1×10^{-11}
	Continuous ^f	M ^e Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous	Q ^e Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Gamma Radioactivity	1×10^{-6}

Legend:

S = once per 12 hours, D = once per 24 hours, W = once per 7 days,
M = once per 31 days, Q = once per 92 days, SA = once per 184 days,
R = once per 18 months, S/U = before each reactor startup,
P = completed before each release, N.A. = Not Applicable

TABLE 4.6.2 NOTATIONS

- a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

The LLD is applicable to the capability of a measurement system under typical conditions and not as a limit for the measurement of a particular sample in the radioactive gaseous waste sampling and analysis program.

For a particular measurement, which may include radiochemical separation:

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

Where

LLD is the lower limit of detection as defined above (microcuries 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 (counts per minute)

E is the counting efficiency (counts per disintegration)

V is the sample size (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

t is the elapsed time between the end of sample collection and the time of counting.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions with typical values of E , V , Y , and λt for the radionuclides Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, Ce-141; an LLD of 1×10^{-10} $\mu\text{Ci/ml}$ should typically be achieved for Mo-99 and Ce-144. Occasionally background fluctuations, interfering radionuclides, or other uncontrollable circumstances may render these LLD's unachievable.

When calculating the LLD for a radionuclide determined by gamma ray spectrometry, the background may include the typical contributions of other radionuclides normally present in the samples. The background count rate of a Ge(Li) detector is determined from background counts that are determined to be within the full width of the specific energy band used for the quantitative analysis for that radionuclide.

- b. 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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Material Release Report. The LLD for Mo-99 and Ce-144 is 1×10^{-10} $\mu\text{Ci/ml}$ whereas the LLD for other principal gamma emitting particulates is 1×10^{-11} $\mu\text{Ci/ml}$. Radionuclides which are below the LLD for the analysis should not be reported.
- c. The noble gas radionuclides in gaseous effluent may be identified by either on-line (gamma spectrum) analysis of a flowing sample of effluent or by taking a grab sample of effluent and analyzing it.
- d. In the event the reactor power level increases more than 15 percent in one hour and the Stack noble gas radioactivity monitor shows an activity increase of more than a factor of three after factoring out the effect due to the change in reactor power, an on-line analysis for noble gas radionuclides in Stack effluent shall be performed or a grab sample of Stack effluent shall be collected and analyzed.
- e. A composite particulate sample shall include an equal fraction of at least one particulate sample collected during each week of the compositing period.
- f. In the event a sample is collected for 24 hours or less, the LLD may be increased by a factor of 10.

Basis: 4.6.A

The purpose of sampling and analyzing reactor coolant is to get an indication of fuel integrity and to determine whether Specification 3.6.A is exceeded. The radioiodine concentration in reactor coolant is expected to change only gradually over several days. The trend of the main condenser offgas radioactivity is also an indicator of the trend of radioiodine in the reactor coolant.

Basis: 4.6.I

The alarm setpoint of the monitor of a continuous, aqueous radioactive release is derived from historical, or post-release, analyses. The trip setpoint of the liquid radwaste effluent monitor is determined on the basis of pre-release sampling and analysis for the batch releases.

4.14 Solid Radioactive Waste

Applicability: During processing wet radioactive wastes destined for disposal by land burial as class B or C waste.

Objective: To verify that class B or C wet radioactive solid waste satisfies stability requirements before disposal.

Specification: Assessment to verify that class B or C wet radioactive waste satisfies stability requirements in 10 CFR Part 61.56(b) before delivery to a carrier for transport to a licensed disposal facility shall be performed according to a Process Control Plan.

4.15 Radioactive Effluent Monitoring Instrumentation Applicability

States surveillance requirements for OPERABILITY of radioactive effluent monitoring instrumentation.

Objective: To demonstrate the OPERABILITY of radioactive effluent monitoring instrumentation.

Specification: A. Liquid Effluent Instrumentation

Each radioactive liquid effluent monitoring instrument channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.15.1.

B. Gaseous Effluent Instrumentation

Each radioactive gaseous effluent monitoring instrument channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.15.2.

TABLE 4.15.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test	Surveillance Required ^a
1. Gross Radioactivity Monitors					
a. Liquid Radwaste Effluent Line	D	D ^g	R ^f	Q ^d	b
b. Reactor Building Service Water System Effluent Line	D	M	R ^f	Q ^e	b
c. Turbine Building Sump No. 1-5	D	M	R ^f	Q ^e	b
2. Flow Rate Measurement Devices					
a. Liquid Radwaste Effluent Line	D ^h	N.A.	R	Q	b

Legend:

S = once per 12 hours, D = once per 24 hours, W = once per 7 days,
 M = once per 31 days, Q = once per 92 days, SA = once per 184 days,
 R = once per 18 months, S/U = before each reactor startup,
 P = completed before each release, N.A. = Not Applicable

TABLE 4.15.1 NOTATIONS

- a. Instrumentation shall be OPERABLE and in service except that a channel may be taken out of service for the purpose of a check, calibration, test or maintenance without declaring it to be inoperable.
- b. During releases via this pathway.
- c. This notation not used.
- d. The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and alarm annunciation in the Radwaste Control Room occur if the instrument indicates measured levels above the alarm setpoint.
- e. 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. Instrument indicates a downscale failure.
 - 3. Instrument controls not set in operate mode.
 - 4. Instrument electrical power loss.
- f. The CHANNEL CALIBRATION shall be performed according to established station calibration procedures.
- g. On any day during which a release is made, a SOURCE CHECK shall be made at least once, before the first release.
- h. A CHANNEL CHECK shall consist of verifying indication of flow during effluent release. A CHANNEL CHECK shall be made at least once during any day on which a release is made.
- i. The CHANNEL FUNCTIONAL TEST shall also demonstrate that Control Room alarm annunciator occurs if any of the following conditions exist:
 - 1. Instrument indicates measured levels above the alarm setpoint.
 - 2. Instrument indicates a downscale failure.

TABLE 4.15.2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test	Surveillance ^a Required
1. Main Condenser Offgas Treatment System Hydrogen Monitor	D	N.A.	Q ^g	M	c
2. Main Stack Monitoring System					
a. Radioactive Noble Gas Monitor	D	M	R ^f	Q ^e	D
b. Iodine Sampler	W	N.A.	N.A.	N.A.	b
c. Particulate Sampler	W	N.A.	N.A.	N.A.	b
d. Effluent Flow Measuring Device	D	N.A.	R	Q	b
e. Sampler Flow Measuring Device	D	N.A.	R	Q	b
3. Turbine Building Ventilation Monitoring System					
a. Radioactive Noble Gas Monitor	D	M	R ^f	Q ^e	b
b. Iodine Sampler	W	N.A.	N.A.	N.A.	b
c. Particulate Sampler	W	N.A.	N.A.	N.A.	b
d. Effluent Flow Measuring Device	D	N.A.	R	Q	b
e. Sampler Flow Measuring Device	D	N.A.	R	Q	b
4. Offgas Building Exhaust Ventilation Monitoring System					
a. Radioactive Noble Gas Monitor	D	M	R ^f	Q ^e	b
b. Iodine Sampler Cartridge	W	N.A.	N.A.	N.A.	b
c. Particulate Sampler	W	N.A.	N.A.	N.A.	b
d. Sampler Flow Measuring Device	D	N.A.	R	N.A.	b

Legend:

S = once per 12 hours, D = once per 24 hours, W = once per 7 days,
M = once per 31 days, Q = once per 92 days, SA = once per 184 days,
R = once per 18 months, S/U = before each reactor startup,
P = completed before each release, N.A. = Not Applicable

- a. Instrumentation shall be OPERABLE and in service except that a channel may be taken out of service for the purpose of a check, calibration, test, or maintenance, or sample media change without declaring it to be inoperable.
- b. During releases via this pathway.
- c. During main condenser offgas treatment system operation.
- d. During operation of the condenser air ejector.
- e. 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. Instrument indicates a downscale failure.
 - 3. Instrument controls not set in operate mode.
 - 4. Instrument electrical power loss.
- f. The CHANNEL CALIBRATION shall be performed according to established station calibration procedures.
- g. The CHANNEL CALIBRATION shall include the use of at least two standard gas samples, each containing a known volume percent hydrogen in the range of the instrument, balance nitrogen.

Applicability: Environmental surveillance of radiation and radioactive effluent from the OCNGS.

Objective: Measurement and assessment of radiation and radioactive material in the environment which was discharged from the OCNGS.

Specifications: 4.16.A Radiological Environmental Monitoring

1. Environmental samples shall be collected and analyzed according to Table 4.16.1. Analytical techniques shall be used such that the detection capabilities indicated in Table 4.16.2 are achieved. Locations from which radiological environmental samples are intended to be collected shall be identified in the Offsite Dose Calculation Manual (ODCM).
2. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, faunal population fluctuation, malfunction of the automatic sampling equipment and other legitimate reasons beyond the control of GPUN.
3. In the event an environmental sample required by Table 4.16.1 is not collected and analyzed in accordance with the provisions of the Table, the deviation shall be documented in the Annual Radiological Environmental Report.
4. If a required specimen is unobtainable due to sampling equipment malfunction, every reasonable effort shall be made to complete corrective action prior to the end of the next sampling period.
5. Any location from which environmental samples or dosimetry can no longer be obtained may be dropped from the surveillance program upon notifying the NRC in writing, in lieu of any other report, that they are no longer obtainable at that location. GPU shall establish a replacement sampling or dosimetry location and shall revise the ODCM in accordance with Specification 6.18.

6. If a confirmed³ measured radionuclide concentration in an environmental sampling medium averaged over any quarter sampling period exceeds the reporting level given in Table 4.16.3, a written report shall be submitted to the NRC within sixty days of the end of the quarter during which the licensee received confirmation that a radiological limit was exceeded. If it can be demonstrated that the level is not a result of plant effluents (i.e., by comparisons with control station, natural radioactivity, or pre-operational data) a report need not be submitted. When more than one of the radionuclides in Table 4.16.3 are detected in the medium, the reporting level shall have been exceeded if:

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

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- ³ A confirmatory re-analysis of the original, a duplicate, or a new sample may be desirable as appropriate. The results of the confirmatory analyses should be completed at the earliest time consistent with the analysis, but in any case within sixty days.

If radionuclides other than those in Table 4.16.3 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objectives of 10 CFR 50, Appendix I. This report may include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

TABLE 4.16.1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Medium Sampled	Minimum Number of Sampling Locations	Sampling and Collection Frequency	Analysis Type
Airborne 1) Particulate	4 Indicator/1 Background	Biweekly	Gross Beta
2) Iodine	4 Indicator/1 Background	Quarterly	Gamma Isotopic
		Weekly	I-131
Gamma radiation	30 Indicator/2 Background	Quarterly	Gamma Dose (TLD)
Groundwater	2 Indicator wells 1 Background well	Semi-Annually Semi-Annually	Isotopic Gamma H-3
Surface Water	1 Indicator/1 Background	Monthly	Gamma Isotopic
Sediment	1 Indicator/1 Background	Semi-Annually	Gamma Isotopic
Fish	1 Indicator/1 Background	Semi-Annually (when available)	Gamma Isotopic
Shellfish (Clams)	1 Indicator/1 Background	Semi-Annually (when available)	Gamma Isotopic
Food Products/Ingestion	1 Indicator (where available) 1 Background (where available)	Monthly (when available)	Gamma Isotopic

TABLE 4.16.2
DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

Lower Limit of Detection ^a					
Isotope	Water (pCi/liter)	Air (pCi/m ³)	Food Products (pCi/kg wet)	Sediments/Soils (pCi/kg dry)	Aquatic Biota (pCi/kg wet)
H-3	2000 ^b 3000 ^c				
Mn-54	15				130
Fe-59	30				260
Co-60	15				130
Co-58	15				130
Zn-65	30				260
Zr-95	30				
Nb-95	15				
Cs-134	15	5x10 ⁻²	60	150	130
Cs-137	18	6x10 ⁻²	80	180	150
La-140	15				
Ba-140	60				
I-131	1000 ^c	7x10 ⁻²	60		
Gross Beta	4	1x10 ⁻²			

TABLE 4.16.2 NOTATIONS

- a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

The LLD is applicable to the capability of a measurement system under typical conditions and not as a limit for the measurement of a particular environmental sample.

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 (picocuries 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 (counts per minute),

E is the counting efficiency (counts per disintegration),

2.2 is the number of disintegrations per minute per curie,

γ is the fractional radiochemical yield, when applicable,

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

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Report pursuant to Specification 6.9.1.e.

- b. For a sample of drinking water. See Basis note 4.16E.
- c. For a sample of water not used as a source of drinking water. See Basis note 4.16E.

TABLE 4.16.3
REPORTING LEVELS (RL) FOR NONROUTINE OPERATING REPORTS

Analysis	Water (pCi/liter)	Airborne Particulate or Gases (pCi/m ³)	Fish pCi/Kg, wet)	Milk (pCi/l)	Broad Leaf Vegetation (pCi/Kg, wet)
H-3	$2 \times 10^4(a)$				
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	0.9		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	

a. Well Water Only - See Basis note 4.16E.

1. The laboratories of the licensee and licensee's contractors which analyze radiological environmental samples shall participate in an NRC-approved environmental radioactivity intercomparison program, if available.
2. In the event comparison samples are not analyzed, the reason shall be reported in the Annual Radiological Environmental Report in lieu of any other report.
3. The provisions of 3.0.A, 3.0.B and 6.9.2 are not applicable.

4.16.C Land Use Survey

A land use survey shall be conducted annually during the growing season to determine the location of the nearest milk animal and nearest garden greater than 50 square meters (500 square feet) producing broadleaf vegetation in each of the sixteen meteorological sectors within a distance of 8 kilometers (5 miles),¹ and the locations of all milk animals and gardens greater than 50 square meters producing broadleaf vegetation out to a distance of 5 kilometers (3 miles) for each radial sector. Methods shall be used that are appropriate for the residential, non-agricultural and highly transient population and associated land uses that exist around the OCMGS. If it is learned from this survey that the milk animals or gardens are present at a location which yields a calculated thyroid dose at least 20 percent greater² than those previously sampled, or if the survey results in changes in the location used in the radioactive effluent technical specifications for dose calculations, the new location (distance and direction) shall be identified in the Annual Radiological Environmental report. Milk animal or garden locations resulting in at least 20 percent higher calculated doses shall be added to the surveillance program and a station exhibiting lower calculated doses may then be dropped from the surveillance program. If the survey reveals that milk animals are not present or are unavailable for sampling, then broadleaf vegetation shall be sampled.

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- (1) Broadleaf vegetation sampling may be performed near the site boundary in the 2 sectors with the highest D/Q in lieu of the garden census.
- (2) As calculated according to the ODCM.

- Basis: 4.16.A It should be noted that in addition to the sampling and analysis required by the proposed technical specifications, GPU Nuclear may choose, to conduct additional sampling and analysis as deemed advisable to assure adequate protection of the health and safety of the public and monitoring of the environment. The "Pathway to Man" concept is emphasized throughout, and the resultant program is directed toward evaluating those media, locations, isotopes, etc. that affect the radiological impact on man.
- The detection capability stated in Table 4.16.2 and the reporting level stated in Table 4.16.3 for each radionuclide in environmental samples are derived from the USNRC Branch Technical Position on Radiological Environmental Monitoring, Rev. 1, Tables 2 and 4, November, 1979.
- Basis: 4.16.C GPUN may propose any of the following methods to accomplish the land use survey. These methods are generally listed in order of overall preference - considering quality of data, cost, and speed of accomplishment on an annual basis. Interpretation of aerial photographs may be the most desirable method for accomplishing an annual land use survey within the vicinity of the Oyster Creek Nuclear Generating Station. In addition to this, information from local and state government agencies will be utilized. Door to door census in the vicinity of Oyster Creek Nuclear Generating Station are not usually a desirable way to produce land use information because of the high number of seasonal/rented residencies in this area. In addition, the high number of dwellings would require an inordinate manpower effort to accomplish a complete census on an annual basis. GPUN may elect, however, to conduct field checks of selected areas that are not fully understood after the interpretation of aerial photographs and the use of state and local government data.
- Basis: 4.16.D Recent on site research conducted by GPUN (final report, March, 1984) has demonstrated that the ground water pathway is not a potential pathway to man from the OCNGS. This recent site research also installed new sampling wells on site.
- Basis: 4.16.E It is to be noted that the surface water that the OCNGS discharges into is a marine estuary containing brackish to salt water that is not used as drinking water or irrigation water by man.
- Basis: 4.16.F The area within five miles of the OCNGS is not well farmed but primarily residential in nature. In addition, the use of vacant land for suburban home tracts is on the increase. At the time of this submittal, limited quantities of locally grown vegetables were available for sampling.
- Basis: 4.16.G The source for Tables 4.16.2 and 4.16.3 is the USNRC Branch Technical Position on Radiological Environmental Monitoring, Revision 1, dated November 1979.

QUALIFICATIONS

- 6.5.2.6 The independent reviewer(s) shall either have a Bachelor's Degree in Engineering or the Physical Sciences and five (5) years of professional level experience in the area being reviewed or have 9 years of appropriate experience in the field of his speciality. An individual performing reviews may possess competence in more than one speciality area. Credit toward experience will be given for advanced degrees on a one-for-one basis up to a maximum of two years.

RECORDS

- 6.5.2.7 Reports of reviews encompassed in Section 6.5.2.5 shall be prepared, maintained and transmitted to the cognizant division Vice President.

6.5.3 AUDITS

- 6.5.3.1 Audits of facility activities shall be performed under the cognizance of the Vice President Nuclear Assurance. These audits shall encompass:
- a. The conformance of facility operations to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.
 - b. The performance, training and qualifications of the facility staff at least once per 12 months.
 - c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.
 - d. The Facility Emergency Plan and implementing procedures at least once per 12 months.
 - e. The Facility Security Plan and implementing procedures at least once per 12 months.
 - f. The radiological environmental monitoring program and the results thereof at least once per 12 months.
 - g. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.
 - h. The PROCESS CONTROL PROGRAM and implementing procedures for radioactive wastes at least once per 24 months.
 - i. Any other area of facility operation considered appropriate by the IOSRG or the Office of the President-GPUNC.
- 6.5.3.2 Audits of the following shall be performed under the cognizance of the Vice President - Technical Functions:
- a. An independent fire protection and loss prevention program

requirements related to safety.

5) Any other matter involving safe operation of the nuclear power plant that the Manager - Nuclear Safety deems appropriate for consideration.

AUTHORITY

6.5.4.4

The IOSRG shall have access to the facility and facility records as necessary to perform its evaluations and assessments. Based on its reviews, the IOSRG shall provide recommendations to the management positions responsible for the areas reviewed.

QUALIFICATIONS

6.5.4.5

IOSRG engineers shall have either (1) a Bachelor's Degree in Engineering or appropriate Physical Science and three years of professional level experience in the nuclear power field which may include technical supporting functions or (2) eight years of appropriate experience in nuclear power plant operations and/or technology. Credit toward experience will be given for advance degrees on a one-to-one basis up to a maximum of two years.

RECORDS

6.5.4.6

Reports of evaluations and assessments encompassed in Section 6.5.4.3 shall be prepared, approved, and transmitted to the Nuclear Safety Assessment Director, Oyster Creek and Nuclear Assurance division Vice Presidents, and the management positions responsible for the areas reviewed.

6.6 REPORTABLE EVENT ACTION

6.6.1

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50; and

b. Each REPORTABLE EVENT shall be reported to the cognizant manager and the cognizant division Vice President and the Vice President a Director Oyster Creek. The functionally cognizant division staff shall prepare a Licensee Event Report (LER) in accordance with the guidance outlined in 10 CFR 50.73(b). Copies of all such reports shall be submitted to the functionally cognizant division Vice President and the Vice President a Director Oyster Creek.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. If any Safety Limit is exceeded, the reactor shall be shut down immediately until the Commission authorizes the resumption of operation.
- b. The Safety Limit violation shall be reported to the Commission and the Vice President and Director, Oyster Creek.
- c. A Safety Limit Violation Report shall be prepared. The report shall be submitted to the Vice President and Director Oyster Creek. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components systems or structures, and (3) corrective action taken to prevent recurrence.
- d. The Safety Limit Violation Report shall be submitted to the Commission with 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained that meet or exceed the requirements of Section 5.2 and 5.3 of American National Standard N18.7-1976 and Appendix "A" of the Nuclear Regulatory Commission's Regulatory Guide 1.33-1972 except as provided in 6.8.2 and 6.8.3 below.

Written procedures shall be adopted and maintained to implement the:

Process Control Plan

Offsite Dose Calculation Manual

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed as described in 6.5.1.1 and approved as described in 6.5.1 prior to implementation and periodically as specified in the Administrative Procedures.

6.8.3 Temporary changes to procedures 6.3.1 above may be made provided:

- a. The intent of the original procedure is not altered.
- b. The change is approved by two members of GPUNC Management Staff authorized under Section 6.5.1.12 and knowledgeable in the area affected by the procedure. For changes which may affect the operational status of facility systems or equipment, at least one of these individuals shall be a member of facility management or supervision holding a Senior Reactor Operator's License on the facility.
- c. The change is documented, subsequently reviewed and approved as described in 6.8.2 within 14 days of implementation.

basis which will include a narrative of operating experience, to the Director, Office of Management and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of I and E, no later than the 15th of each month following the calendar month covered by the report.

- d. Semiannual Radioactive Effluent Release Report: A report of radioactive materials released from the Station during the preceeding six months shall be submitted to the NRC within 60 days after January 1 and July 1 of each year. Each report shall include the following information:
1. a summary by calendar quarter and by radionuclide of the quantities of radioactive liquid and gaseous effluents from the Station,
 2. a summary of radioactive solid waste shipped from the Station including:
 - a. physical description of the waste
 - b. classification of the waste, per 10 CFR Part 61
 - c. solidification agent (if solidified)
 - d. total volume shipped
 - e. total quantity of radioactive material shipped (curies)
 - f. best knowledge of identity of principal radionuclides
 3. a description of any changes to the PCP or ODCM,
 4. a summary of meteorological data collected during the year shall be included in the report submitted within 60 days after January 1 of each year. Alternatively, summary meteorological data may be retained by GPU Nuclear and made available to the NRC upon request.
- e. Annual Radiological Environmental Report: A report of radiological environmental surveillance activities during each year shall be submitted before May 1 of the following year. Each report shall include the following information required in Specification 4.16 for radiological environmental surveillance:
1. a summary description of the radiological environmental monitoring program,
 2. a map and a table of distances and directions (compass azimuth) of locations of sampling stations from the reactor,
 3. results of analyses of samples and of radiation measurements, (In the event some results are not available, the reasons shall be explained in the report. In the event the missing results are obtained, they shall be reported to the NRC as soon as is reasonable.)

4. deviation(s) from the environmental sampling schedule in Table 4.16.1.
5. identification of environmental samples analyzed when instrumentation was not capable of meeting detection capability in Table 4.16.2.
6. a summary of the results of the land use survey,
7. a summary of the results of licensee participation in an NRC approved inter-laboratory crosscheck program for environmental samples.
8. results of dose evaluations to demonstrate compliance with 40 CFR Part 190.10a.

Basis: 6.9.1.e An annual report of radiological environmental surveillance activities includes factual data summarizing results of activities required by the surveillance program. In order to aid interpretation of the data, GPUN may choose to submit analysis of trends and comparative non regional radiological environmental data. In addition, the licensee may choose to discuss previous radiological environmental data as well as the observed radiological environmental impacts of station operation (if any) on the environment.

6.9.2 REPORTABLE EVENTS

The submittal of Licensee Event Reports shall be accomplished in accordance with the requirements set forth in 10 CFR 50.73.

6.9.3 UNIQUE REPORTING REQUIREMENTS

Special reports shall be submitted to the Director of Regulatory Operations, Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

- a. Materials Radiation Surveillance Specimen Reports (4.3A)
- b. Integrated Primary Containment Leakage Tests (4.5)
- c. Results of required leak tests performed on sealed sources if the tests reveal the presence of 0.005 microcuries or more of removeable contamination.
- d. Inoperable Fire Protection Equipment (3.12)
- e. Core Spray Sparger Inservice Inspection (Table 4.3.1-9)

Prior to startup of each cycle, a special report presenting the results of the inservice inspection of the Core Spray Spargers during each refueling outage shall be submitted to the Commission for review.
- f. Liquid radwaste batch discharge exceeding Specification 3.6.B.1.
- g. Main condenser offgas discharge without treatment per Specification 3.6.D.1.

- h. Dose due to radioactive liquid effluent exceeding Specification 3.6.J.1.
- i. Air dose due to radioactive noble gas in gaseous effluent exceeding Specification 3.6.L.1.
- j. Air dose due to radioiodine and particulates exceeding Specification 3.6.M.1.
- k. Annual total dose due to radioactive effluents exceeding Specification 3.6.N.1.
- l. Records of results of analyses required by the Radiological Environmental Monitoring Program.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for a least five years:

- a. Records and logs of facility operation covering time interval at each power level.
- b. Records and logs of principle maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
- c. Reportable occurrence reports.
- d. Records of surveillance activities, inspections and calibrations required by these technical specifications.
- e. Records of reactor tests and experiments.
- f. Records of changes made to operating procedures.
- g. Records of radioactive shipments.
- h. Records of sealed source leak tests and results.
- i. Records of annual physical inventory of all source material of record.

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

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

- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
- c. Records of facility radiation and contamination surveys.
- d. Records of radiation exposure for all individuals entering radiation control areas.
- e. Records of gaseous and liquid radioactive material released to the environs.
- f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.
- g. Records of training and qualification for current members of the plant staff.
- h. Records of inservice inspections performed pursuant to these technical specifications.
- i. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
- j. Records of reviews by the Independent Onsite Safety Review Group.
- k. Records for Environmental Qualification which are covered under the provisions of paragraph 6.14.
- l. Records of results of analyses required by the Radiological Environmental Monitoring Program.

6.10.3 Quality Assurance Records shall be retained as specified by the Quality Assurance Plan.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 DELETED

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).

NOTE: Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are following plant radiation protection procedures for entry into high radiation areas.

An individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

- a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.
- c. A health physics qualified individual (i.e. qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive exposure control over the activities within the area and who will perform periodic radiation surveillance at the frequency in the RWP. The surveillance frequency will be established by the Radiological Controls Manager.

6.13.2 Specification 6.13.1 shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of operations and/or radiation protection supervision on duty.

6.14 ENVIRONMENTAL QUALIFICATION

- A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1 E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-16 dated October 24, 1980.
- B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.15 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and
2. System leak test requirements, to the extent permitted by system design and radiological conditions, for each system at a frequency not to exceed refueling cycle intervals. The systems subject to this testing are (1) Core Spray, (2) Containment Spray, (3) Reactor Water Cleanup, (4) Isolation condenser and (5) Shutdown Cooling.

6.16 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas* under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

* Areas requiring personnel access for establishing hot shutdown condition.

6.17 PROCESS CONTROL PLAN

1. GPU may change the Process Control Plan provided each change is submitted to the Commission by inclusion in the Semiannual Radioactive Material Release Report for the period in which the change is made effective and contains:
 - a. Sufficiently detailed information to support the rationale for the change,
 - b. a determination that the product waste form will conform to the requirements of 10 CFR Part 61.56.
2. Change(s) shall become effective after review and approval in accordance with Section 6.8.2.

6.18 OFFSITE DOSE CALCULATION MANUAL

1. GPU may make changes to the Offsite Dose Calculation Manual (ODCM) provided each change is submitted to the Commission in the Semiannual Radioactive Material Release Report for the period in which the change is made effective. The submittal shall contain:
 - a. sufficiently detailed information to support the rationale for the change,
 - b. a determination that the change will not substantially reduce the ability of dose calculations or setpoint determinations to assess compliance with Specifications, and

2. Change(s) shall become effective after review and approval in accordance with Section 6.8.2.

6.19 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

1. Each modification to a radioactive waste treatment system which does not involve an unreviewed safety question:
 - a. Shall be performed in accordance with the provisions of 10 CFR Part 50.59, except
 - b. The description of the modification and a written safety evaluation which includes the bases for the change shall be submitted as part of the annual FSAR update, and
 - c. Shall become effective upon review and approval by the Vice President and Director.

Basis: 6.1.9 The radioactive waste treatment systems are those systems described in the Facility Safety Analysis Report (FSAR) and amendments thereto that are used to maintain control over radioactive materials in liquid and gaseous effluents and in solid waste packages for shipment offsite to a radioactive waste disposal facility and that are required to meet the conditions in Specifications 3.6.B, 3.6.D, and 3.1.4. The NRC is notified of major changes to these radioactive waste treatment systems under the provisions of 10 CFR Part 50.59 and Part 50.71, the FSAR update.

ATTACHMENT I

RESPONSES TO NRC COMMENTS RELATED TO GPU NUCLEAR
PROPOSED RADIOACTIVE EFFLUENT AND ENVIRONMENTAL TECHNICAL SPECIFICATIONS
FOR THE OYSTER CREEK NUCLEAR GENERATING STATION

1.0 UNRESTRICTED AREA

ITEM: Consider defining "unrestricted area" in specifications section 1.

RESPONSE: For the purpose of radioactive effluent technical specifications implementing 10 CFR Part 20.106 and Part 50 Appendix I, the term, EXCLUSION AREA, as defined in 10 CFR Part 100.3(a) seems well suited for the intended provisions of the RETS in that it accommodates public activity within the area under appropriate public health and safety limitation. As applied, activities by Members of the Public unrelated to nuclear station operation in identified public areas within the Exclusion Area would be subject to limits in the pertinent technical specifications. Presently, GPU expects that low occupancy time by any Member of the Public within the Exclusion Area, for example, at the visitor's center, will preclude him from being the most exposed Member of the Public.

2.0 SITE BOUNDARY MAP

ITEM: Consider including a map of the Site Boundary in specifications section 5.

RESPONSE: GPU proposes to include a map describing the Exclusion Area boundary in the Offsite Dose Calculation Manual (ODCM) where it is more readily associated with offsite dose assessment. 10 CFR Part 50.36 (c)(4), the determinant of specification section 5 seems to be intended to accommodate facility design features rather than the site.

3.0 SOLIDIFICATION

ITEM: If the term, "solidification", is used in the specifications, then it should be defined in specifications section 1.

RESPONSE: The term, "solidification", is not used in the specifications in a way that an auditor is likely to need an inspectable definition.

4.0 VENTILATION EXHAUST TREATMENT SYSTEM

ITEM: The proposed RETS do not include a definition or specification for "ventilation exhaust treatment system."

RESPONSE: The Standby Gas Treatment System is considered to be an engineered safety feature. According to the NRC model definition 1.19, it would not be a ventilation exhaust treatment system. Otherwise, the only ventilation exhaust air treatment defined according to NRC model definition 1.19 and determined to be cost-beneficial is HEPA filtration of radwaste building air that is exhausted. The filter is a passive device whose maintenance is adequately assured by operating procedures. It does not warrant the extent of controls proposed in NRC model specification 3.11.2.5 and associated model definition 1.19.

5.0 LIQUID RADWASTE TREATMENT

ITEM: Decide whether proposed specification 3.6.B should be based upon a dose limit or a concentration limit. Explain the choice.

RESPONSE: Specification 3.6.B implements 10 CFR 50.36a related to operation of radioactive waste treatment equipment to keep radioactive material in effluents to unrestricted areas as low as reasonably achievable. It particularly implements the 10 CFR Part 50 Appendix I section II.D provision for cost-beneficial treatment of radioactive liquid waste before release in effluent. Separately, specification 3.6.I implements the 10 CFR Part 20.106 concentration limit, and specification 3.6.J implements the 10 CFR Part 50 Appendix I section IV.A dose limit for radioactive material in effluent. It is therefore rational and appropriate to formulate proposed specification 3.6.B on the basis of an ALARA concentration derived from cost-benefit considerations.

The Oyster Creek Nuclear Generating Station has two main systems to treat aqueous radwaste. The High Purity Waste System includes filtration and ion exchange. The Chemical and Floor Drain Waste System includes filtration, evaporation, and condensate ion exchange.

The radioactivity concentration, 0.001 uCi/ml, below which it is not cost beneficial to treat aqueous radwaste and thus is ALARA, is demonstrated below. It is derived on the assumption that aqueous radwaste may optionally be treated by either available system.

Aqueous radwaste discharge data during a year of OCNCS operation, 1981, is used as a basis of the analysis. During 1981 about 8.9 million gallons of aqueous radwaste were processed, and about 1/4 of that was discharged to the canal after treatment. The water discharged contained 0.248 curie of radioactive material and 26.7 curies of tritium.¹

¹GPU Oyster Creek Nuclear Generating Station, Effluent and Waste Disposal Semianual Reports, 1981.

The resulting population dose commitment calculated by GPU for these aqueous discharges, less tritium, is

thyroid	1.5 man rem
total body	<u>2.9</u> man rem
total	4.4 man rem

The thyroid and total body dose equivalents are added because 10 CFR Part 50, Appendix I, section I.D prescribes that man-thyroid-rem and total body man rem each be valued at \$1000 for the purpose of cost-benefit analysis. Accordingly, the computed population total body plus thyroid dose is about 17.7 man rem per curie of radioactive material other than tritium discharged in radioactive waste water.

If the volume of aqueous radwaste generated during 1981 were all processed through the High Purity Waste System, the total cost, estimated according to Regulatory Guide 1.110, would be \$617,000. Likewise, if the same volume of radwaste, alternatively were processed through the Chemical and Floor Drain Waste System, the total cost is estimated to be \$678,000.

The unit volume total treatment cost is therefore about

$$\begin{array}{l} \text{cost to treat in} \\ \text{High Purity Waste System} \end{array} = \frac{\$617,000}{8.9E6 \text{ gal}} = \$0.069/\text{gal}$$

$$\begin{array}{l} \text{cost to treat in} \\ \text{Chemical and Floor Drain} \\ \text{Waste System} \end{array} = \frac{\$678,000}{8.9E6 \text{ gal}} = \$0.076/\text{gal}$$

Assuming the cost-benefit balance occurs at \$1000 expenditure per man rem reduction, and assuming treatment removes all radioactivity other than tritium from the liquid, then, except for tritium

- (1) the activity concentration below which treatment in the High Purity Waste System is not cost beneficial is

$$C = \frac{\$617,000}{8.9E6 \text{ gal} \times 3785 \text{ ml/gal}} \times \frac{1 \text{ Ci}}{17.7 \text{ man rem}} \times \frac{10 \text{ uCi}}{\text{Ci}} \times \frac{1 \text{ man rem}}{\$1000}$$

$$C = 0.0010 \text{ uCi/ml}$$

- (2) the activity concentration below which treatment in the Chemical and Floor Drain Waste System is not cost beneficial is

$$C = \frac{\$678,000}{8.9E6 \text{ gal} \times 3785 \text{ ml/gal}} \times \frac{1 \text{ Ci}}{17.7 \text{ man rem}} \times \frac{10 \text{ uCi}}{\text{Ci}} \times \frac{1 \text{ man rem}}{\$1000}$$

$$C = 0.0011 \text{ uCi/ml}$$

Therefore, treatment of aqueous radwaste containing less than 0.001 uCi/ml is not justified in either of the liquid radwaste systems.

6.0 LIQUID RADWASTE TREATMENT SURVEILLANCE

ITEM: There appears to be no surveillance condition in Specification 3.6.B.

RESPONSE: The essence of proposed specification 3.6.B is an activity concentration limit applicable to radwaste water to be released to offsite environs. Proposed specification 4.6.I.1 provides for sampling and analysis of liquid radwaste batch tanks for radioactivity concentration before release. Appropriate surveillance is thereby provided in the proposed RETS.

7.0 RADIOACTIVE LIQUID STORAGE IN OUTDOOR TANKS

ITEM: Explain the rationale for omitting from proposed specification 3.6.C any outdoor tank that is used to store radioactive liquid.

RESPONSE: Four tanks other than those named in the proposed specification receive low radioactivity concentration water but are not included for the following reasons. The tanks, HPT-2A AND HPT-2B, both high purity waste sample tanks, and WC-T-3A AND WC-T-3B, both chemical waste distillate sample tanks, receive radwaste water after radwaste treatment. Each tankfull is sampled and analyzed for radioactivity in accordance with proposed specification 4.6.I.1 before controlled release to the discharge canal. In the ordinary course of events, a history of the expected low activity concentration in these tanks will be accumulated. During a year of station operation, 1981, only 26.7 curies of tritium and 0.248 curie of other radionuclides were discharged via the four tanks. It would be unlikely that any of the tanks contain as much as 10 curies at one time, the limit proposed in the specification.

These four tanks are within a curbed area that can contain about 200 gallons of liquid. In the event of a leak into the curbed area, valving in the drainage line can route the liquid via a floor drain sump to the radwaste system collection tanks.

8.0 CONDENSER OFFGAS TREATMENT

ITEM: Explain why a special report should not be submitted if gases from the main condenser air ejector are released without treatment in the Augmented Offgas System for more than 15 days.

RESPONSE: Specification 3.6.D implements 10 CFR Part 50.36a and Appendix I section IV relative to maintenance and operation of the AOG system to treat offgas to keep radioactive material in effluents ALARA. This implies a reasonable effort to maintain and operate the AOG charcoal absorbers, which specification 3.6.D.1 addresses. Even with reasonable maintenance effort, the failure rate and time to repair certain components in the AOG system, such as blowers, experience at the OCNGS indicates that more than 15 days of AOG system outage is some-

times necessary.

Insofar as reporting such an outage is concerned, 10 CFR Part 50.36a(a)(2) and Part 50 Appendix I section IV.A seem to indicate that the Commission intends to require reporting in the event the amount of radioactive material actually released is significantly above design dose objectives and not for arbitrarily more restrictive conditions. Nevertheless, GPU Nuclear has proposed earlier reporting in specification 3.6.D.2 than it interprets that the Commission intends.

9.0 DOSE DUE TO RADIONUCLIDES IN AIRBORNE EFFLUENT

ITEM: Explain the rationale for omitting tritium from the source of airborne effluent that is the basis for assessing compliance with specification 3.6.M.

RESPONSE: Specification 3.6.M implements 10 CFR Part 50 Appendix I section IV.A. Dose limits appropriate for section IV.A are derived as specified in section IV.A. from section II, the guidance on design objectives. The pertinent paragraph, C, in section II does not include tritium. Nor is it supported by the Opinion of the Commission in the rulemaking.² The reason is that it is clearly impractical and not reasonably achievable to remove tritium from the effluent. As a result it was not considered a contributor in establishing the design objective, nor should its contribution to the dose be charged against the dose limit in the specification that implements the Appendix I limit on doses.

10.0 REPORTING OFFSITE DOSE EXCEEDING APPENDIX I DOSE LIMIT

ITEM: Explain proposed reporting of offsite dose in excess of an Appendix I dose limit within 30 days from the end of the quarter in which the causative release occurred. This appears in proposed specifications 3.6.J.2, 3.6.L.2, and 3.6.M.2.

RESPONSE: The NRC Staff's guidance with respect to reporting when the radiation exposure in an unrestricted area due to radioactive material released in effluent from a light-water-cooled nuclear power reactor exceeds the Commission's intended action level for the ALARA criterion is stated in 10 CFR Part 50 Appendix I section IV.A.3. OCNCS proposed specifications 3.6.J.2, 3.6.L.2, and 3.6.M.2 are consistent with the Commission's guidance in Appendix I section IV.A.3.

²Opinion of the Commission in the Matter of the Rulemaking on Numerical Guides and Limiting Conditions for Operation to Meet the Criterion "As Low as Practicable" . . . , Docket No. RM-50-2, April 30, 1975.

11.0 GASEOUS EFFLUENT MONITORING INSTRUMENTATION

ITEM: In proposed Table 3.15.2, the offgas building exhaust ventilation monitoring system does not include an effluent air flow measuring device.

RESPONSE: Since the system is not equipped with such a device, fan curves are used to estimate exhaust air flow.

ITEM: In the event an exhaust air flow measuring device that measures air flow in the stack or turbine building exhaust ventilation is inoperable, decide whether the preferable alternative (Table 3.15.2 Action 122) is to estimate the air flow within a stated frequency or whenever the exhaust fan combination is changed.

RESPONSE: The most pronounced change in discharge air flow would be expected when the exhaust fan combination is changed. To account for discharge air flow in the absence of flow measurement, it would seem rational to estimate the air flow after a fan combination changes in order to associate the estimate with the factor likely to be affecting it. The alternative of estimating the air flow on an arbitrary frequency would not seem to collect any additional data that would improve the estimation of radioactive material discharged in the air.

12.0 AIR EJECTOR OFFGAS RADIATION MONITOR

ITEM: Confirm the action initiated when the main condenser air ejector offgas radiation monitor ("pretreat" monitor) trips due to high radiation.

RESPONSE: The monitor trip in response to high radiation causes the augmented offgas system bypass valve (V-7-31), and the offgas drain valve (V-7-29) to close. Note, however, that during normal reactor operation, the air ejector and AOG system would be in service and the AOG bypass valve would be closed already.

13.0 SUMP NUMBER 1-5 RADIATION MONITOR

ITEM: Confirm that the Turbine Building Sump Number 1-5 radiation monitor is planned for future service.

RESPONSE: The current surveillance practice for water accumulated in the sump is to sample and analyze it for radioactivity before pumping it on a batch basis to the discharge canal.

A level controller has been installed which will empty the sump automatically. The level controller has not yet been put in service. The radiation monitor mentioned in proposed Table 3.15.1 will have its detector immersed in water in the sump. In the event the activity concentration in the water exceeds the monitor trip setpoint, it will cause the sump evacuation (discharge) pump to turn off. The radiation monitor which will be installed and level controller are expected to be ready for service by June 1985.

14.0 EFFLUENT RADIATION MONITOR ALARM/TRIP FUNCTIONS

ITEM: Consider a recommendation to state in the technical specifications which effluent radiation monitors cause an alarm and which cause a trip.

RESPONSE: A description of factual information of this kind seems to be appropriate for inclusion in the bases of the technical specifications. Accordingly, a statement of which effluent monitors cause an alarm and which cause a trip when its high radiation setpoint is exceeded is included in Bases 3.15.A and 3.15.B.

15.0 ALTERNATE GASEOUS EFFLUENT MONITOR

ITEM: Consider putting action statement permitting an auxiliary monitoring system in the technical specifications rather than in the bases.

RESPONSE: The bases of specification 3.15.B describe the radioactive gaseous effluent monitoring system (RAGEMS) to be implemented in the stack and the turbine building exhaust air streams. In the event the RAGEMS radioactive noble gas monitor is inoperable, CPU proposes either to grab sample the effluent and analyze it for radioactive gas or to provide an alternate monitoring system. This action is stated in proposed Table 3.15.2 Actions 123 and 124. The purpose of the discussion in Basis 3.15.B is to explain the intent and to identify the existing GE instrumentation to be used as the auxiliary monitor.

16.0 GASEOUS EFFLUENT SURVEILLANCE

ITEM: Consider additional action statements in specification 4.6.K.

RESPONSE: Specifications 3.15.B and 4.6.K, together, provide for surveillance of radioactive noble gases and include essential actions.

Surveillance for 3.6.K is accomplished by specifications 3.15.B, 4.16.K, 4.16.L, and 4.16.M in combination. When each radioactive noble gas effluent monitor setpoint is at its limit, the dose rate is determined to be within the dose rate limit as long as an alarm does not occur. If an alarm occurs when the monitor setpoint is below its limit, compliance can be assessed by comparing the monitor record with the setpoint (limit) calculated according to the method in the ODCM. If an alarm occurs and the monitored release exceeds the setpoint (limit), then compliance can be evaluated by calculating the dose rate by a method in the ODCM.

17.0 RADIOACTIVE EFFLUENT MONITOR SURVEILLANCE

ITEM: Discuss how a circuit failure in radioactive effluent monitoring instrumentation is indicated.

RESPONSE: It is doubtful that instrumentation is available that will indicate every kind of circuit failure. At the OCNGS most of the instruments of interest cause annunciation in the control room if electrical power to the circuit is lost. Included are the radioactivity monitor on the liquid radwaste effluent line, the reactor building service water effluent line, the Stack noble gas monitor, the turbine building noble gas monitor and the offgas building noble gas monitor. Tables 4.15.1 and 4.15.2 have accompanying notes concerning channel functional tests which have been revised to include demonstration of annunciation if the instrument loses electrical power.

18.0 RADIOACTIVE EFFLUENT MONITOR CALIBRATION

ITEM: Consider requiring that radioactive effluent monitoring instrument calibration be related to the initial calibration and use standards traceable to NBS standards.

RESPONSE: The channel calibration requirement in question appears in proposed Tables 4.15.1 and 4.15.2, note f. The corresponding NRC model Tables 4.3.7.11-1 and 4.3.7.12-1 note 3 states, "Operating plants may substitute previously established calibration procedures for this requirement." GPU has elected this option as its means of controlling channel calibration.

19.0 MAJOR CHANGES TO RADWASTE TREATMENT SYSTEMS

ITEM: Consider providing more detail in proposed specification 6.19 describing major changes to radioactive waste treatment systems.

RESPONSE: 10 CFR Parts 50.59 and 50.71 regulate licensee initiated changes to a facility with an operating license, prescribe FSAR updating, and schedule submittal of the information to the NRC. Major changes to the radioactive waste treatment systems are subject to these regulations. GPU does not believe that radioactive waste treatment systems merit unduly more detail in a submittal than is required for other important systems. It seems questionable to single out radwaste systems for such a specification as the NRC model specification at all. Nevertheless, a technical specification 6.19 is proposed that GPU believes is in concert with 10 CFR Parts 50.59 and 50.71 and that provides for submittal to the NRC of a degree of detail describing major changes to radwaste treatment systems equivalent to what is provided for other important systems. A Basis 6.19 has been added to clarify the intent of proposed specification 6.19.

20.0 QUALITY ASSURANCE

ITEM: Consider proposing specifications requiring quality control procedures and audits thereof applicable to radioactive effluent and environmental monitoring.

RESPONSE: Existing specification 6.8.1, which provides for written procedures, applies to radioactive effluent and environmental monitoring. The OCNCS Quality Assurance Auditing group, implementing the existing Corporate Quality Assurance Plan, audits radioactive effluent and environmental monitoring programs. Commercial laboratory quality control of environmental sample analyses is audited by the OCNCS Environmental Controls Department.

Since these functions are already being performed under broader plans or specifications approved by the NRC, additional specifications are not needed.

21.0 RADIOLOGICAL ENVIRONMENTAL MONITORING

ITEM: Consider describing generic locations of radiological environmental monitoring station locations in Table 4.16.1.

RESPONSE: GPU has a long-standing environmental monitoring program around the OCNGS with established sampling locations. Many of these locations do not seem to lend themselves to a generic description which can be written into the technical specifications. To write their location into the specification would stifle the ability to make warranted adjustments to sampling locations in the future. Rather, current station locations corresponding to proposed Table 4.16.1 are described in the ODCM as suggested by the NRC model specification. Any change in a station location described in the ODCM will be reviewed by GPU Nuclear and reported to the NRC in accordance with the provisions of proposed Specification 6.18 governing changes to the ODCM.