Oyster Creek Nuclear Generating Station (OCNGS)
Operating License No. DPR-16
Docket No. 50-219
Technical Specification Change Request No. 206

1.0. PROPOSED TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR)

GPUN requests that the following pages of the OCNGS Technical Specifications (Tech. Specs.) be replaced as indicated below:

Replace Pages: ii, and iii; 1.0-6; 3.6-2, 3.6-3, 3.6-4, 3.6-5, 3.6-6; 3.14-1; 3.15-1, 3.15-2 (new Table 3.15.2), 3.15-3; 4.6-1, and 4.6-2; 4.14-1; 4.15-1, and 4.15-2 (new Table 4.15-2); 4.16-1; 6-10, 6-11, 6-12, 6-13, 6-14, 6-15, 6-16, 6-17, and 6-18.

Add New Pages: 3.6-1; 6-19, and 6-20.

Delete Pages: 3.6-1a, 3.6-1b, 3.6-7a, 3.6-7b, 3.6-8, 3.6-9, and 3.6-10; 3.15-4 (Table 3.15.1), 3.15-5, 3.15-6 (old Table 3.15.2), 3.15-7, and 3.15-8; 4.6-3, 4.6-4 (Table 4.6.1), 4.6-5, 4.6-6, 4.6-7 (Table 4.6.2), 4.6-8, 4.6-9, and 4.6-10; 4.15-3, 4.15-4 (old Table 4.15.2), and 4.15-5; 4.16-2, 4.16-3, 4.16-4 (Table 4.16.1), 4.16-5 (Table 4.16.2), 4.16-6, and 4.16-7 (Table 4.16.3); 6-12a, and 6-12b.

2.0. DESCRIPTION OF CHANGES

The following changes relocate the Radiological Effluent Technical Specifications (RETS) and the Radioactive Effluent Monitoring Program (REMP) to the Offsite Dose Calculation Manual (ODCM) in accordance with Generic Letter (GL) 89-01. Pursuant to GL 89-01 and NUREG-1301 guidance (Reference 7.1), Revision 5 of the ODCM is included for reference. The sampling frequencies and locations are site based. Relocation of the procedural details of RETS to the Process Control Program (PCP), pursuant to GL 89-01, allows for concurrent implementation with the proposed TSCR.

p ii iii	The Table of Contents pages have been updated to reflect the administrative corrections necessary for title changes, deletions, and page numbering changes.
p 1.0-6	Tech. Spec. Definition 1.36, "Offsite Dose Calculation Manual (ODCM)" has been revised pursuant to GL 89-01, and conforms to 10 CFR 50.36(a)(2) for periodicity of reporting.
p 1-6	Tech. Spec. Definition 1.33, "Process Control Plan (PCP)" has been revised pursuant to GL 89-01.
p 3.6-1a	Tech. Spec. 3.6.A, "Radioactive Effluents - Reactor Coolant Radioactivity", Paragraph 3 Annual Reporting Requirement has been deleted and will now be governed by the ODCM, as revised.
p 3.6-1b	Tech. Spec. 3.6.B, "Radioactive Effluents - Liquid Radwaste Treatment," has been deleted and relocated to the ODCM.
P 3.6-2	Tech. Spec. 3.6.C.2, "Radioactive Liquid Storage," has been revised to conform to 10 CFR 50.36(a)(2) with respect to the reporting period being changed from 'semiannual' to Annual.

9310150016 931008
PDR ADOCK 05000219
P PDR

Oyster Creek Nuclear Generating Station (OCNGS)
Technical Specification Change Request No. 206

p 3.6-2	Tech. Spec. 3.6.D, "Radioactive Effluents - Condenser Offgas Treatment," has been deleted, and relocated to the ODCM, as revised.
p 3.6-3	Tech. Spec. 3.6.I, "Radioactive Effluents - Radioactive Concentration in Liquid Effluent," has been deleted and relocated to the ODCM, as revised.
p 3.6-4	Tech. Spec. 3.6.J, "Radioactive Effluents - Limit on Dose Due to Liquid Effluent," has been deleted and relocated to the ODCM, as revised.
p 3.6-4	Tech. Spec. 3.6.K, "Radioactive Effluents - Dose Rate Due to Gaseous Effluent," has been deleted and relocated to the ODCM, as revised.
p 3.6-5	Tech. Spec. 3.6.L, "Radioactive Effluents - Air Dose Due to Noble Gas in Gaseous Effluent," has been deleted and relocated to the ODCM, as revised.
p 3.6-5	Tech. Spec. 3.6.M, "Radioactive Effluents - Dose Due to Radioiodine and Particulates in Gaseous Effluent," has been deleted and relocated to the ODCM, as revised.
p 3.6-5 3.6-6	Tech. Spec. 3.6.N, "Radioactive Effluents - Annual Total Doses Due to Radioactive Effluents," has been deleted and relocated to the ODCM, as revised.
p 3.6-7b 3.6-8 3.6-9 3.6-10	Bases: Tech. Specs. 3.6.B, 3.6.D, 3.6.I, 3.6.J, 3.6.K, 3.6.L/M, and 3.6.N have been deleted and relocated to the ODCM, as 3.6-9 revised.
p 3.14-1	Tech. Spec. 3.14, "Solid Radioactive Waste," and Basis have been deleted, as it is governed by the Process Control Plan.
p 3.15-1 3.15-2 3.15-3	Portions of Tech. Spec. 3.15, "Radioactive Effluent Monitoring Instrumentation" and Basis have been deleted and relocated to the ODCM, as revised; and, the Tech. Spec. has been retitled as "Explosive Gas Monitoring Instrumentation" pursuant to the GL 89-01.
p 3.15-4 3.15-5	Table 3.15.1, "Radioactive Liquid Effluent Monitoring Instrumentation," has been deleted and relocated to the ODCM, as revised, together with the Table 3.15.1 Notations.
p 3.15-6 3.15-7 3.15-8	Table 3.15.2, "Radioactive Gaseous Effluent Monitoring Instrumentation," has been revised to reflect "Explosive Monitoring Instrumentation" only, pursuant to GL 89-01; the remainder of the information including Table 3.15.2 Notations (with the exception of the notations applicable to the Explosive Gas Monitoring Instrumentation) has been deleted and relocated to the ODCM, as revised.

Oyster Creek Nuclear Generating Station (OCNGS)
<u>Technical Specification Change Request No. 206</u>

p 4.6-1 4.6-2 4.6-3	Tech. Spec. 4.6, "Radioactive Effluent", Surveillance has been deleted and relocated to the ODCM, as revised.
p 4.6-4 4.6-5 4.6-6	Table 4.6.1, "Radwaste in Liquid Waste Sampling and Analysis Program", has been deleted together with its "Table Notations" and relocated to the ODCM, as revised.
p 4.6-7 4.6-8 4.6-9	Table 4.6.2, "Radioactive Gaseous Waste Sampling and Analysis Program", has been deleted together with its "Table Notations" and relocated to the ODCM, as revised.
p 4.14-1	Tech. Spec. 4.14, "Solid Radioactive Waste," has been deleted as it is governed by the Process Control Plan.
p 4.15-1 4.15-2 4.15-3	Tech. Spec. 4.15, "Radioactive Effluent Monitoring Instrumentation and Table 4.15.1, "Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements," and Table 4.15.1 "Table Notations" have been deleted and relocated to the ODCM, as revised.
p 4.15-1 4.15-4	New Tech. Spec. 4.15, "Explosive Gas Monitoring Instrumentation" and a Revised Table 4.15.2, "Explosive Gas Monitoring Instrumentation Surveillance Requirements" with revised Table 4.15.2 "Table Notations" replaces the Tech. Specs moved to the ODCM pursuant to the GL 89-01 recommendations.
p 4.16-1 4.16-2 4.16-3	Tech. Specs. 4.16.A, "Radiological Environmental Monitoring" Surveillance; 4.16.B, "Interlaboratory Comparison Program;" and, 4.16.C, "Land Use Survey;" and 4.16 Basis have been deleted and relocated to the ODCM, as revised.
p 4.16-4	Table 4.16.1, "Radiological Environmental Monitoring Program," has been deleted and relocated to the ODCM, as revised.
p 4.16-5 4.16-6	Table 4.16.2, "Detection Capabilities for Environmental Sample Analysis," and Table 4.16.2 "Notations" have been deleted and relocated to the ODCM, as revised.
p 4.16-7	Table 4.16.3, "Reporting Levels (RL) for Nonroutine Operating Events," has been deleted and relocated to the ODCM, as revised.
p 6-10	Tech. Spec. 6.8, "Procedures," is replaced in its entirety with revised Tech. Spec. "Procedures and Programs" wording which reflects the GL 89-01 recommendations and adding a new subsection 6.8.4 for the "Radioactive Effluent Controls Program" and the "Radiological Environmental Monitoring Program."

p 6-11 Tech. Spec. 6.9.1.d, "Semiannual Radioactive Effluent Release Report," reporting requirements for routine reports, has been simplified and re-worded pursuant to GL 89-01; and, has been revised to conform to 10 CFR 50.36(a)(2) with respect to the reporting period being changed from 'semiannual' to Annual. NOW LOCATED ON NEW TECHNICAL SPECIFICATION PAGE 6-14.

p 6-12 Tech. Spec. 6.9.1.e, "Annual Radiological Environmental Report," reporting requirements for routine reports, has been simplified and re-worded pursuant to GL 89-01; NOW LOCATED ON NEW TECHNICAL SPECIFICATION PAGE 6-14.

p 6-12b Tech. Spec. 6.9.1.e Basis has been deleted as reporting requirements for routine reports will be governed by the ODCM, as revised.

p 6-15 New Tech. Spec. 6.10.2.n, "Records Retention" for the reviews of the ODCM and PCP has been added pursuant to GL 89-01.

p 6-17 Tech. Spec. 6.18, "Process Control Plan," has been revised to reflect that which appears in GL 89-01, and has been revised to conform to 10 CFR 50.36(a)(2) with respect to the reporting period being changed from 'semiannual' to Annual.

p 6-17 Tech. Spec. 6.19, "Offsite Dose Calculation Manual," has been revised to reflect that which appears in GL 89-01, and has been revised to conform to 10 CFR 50.36(a)(2) with respect to the reporting period being changed from 'semiannual' to Annual.

p 6-18 Tech. Spec. 6.20, "Major Changes to Radioactive Waste Treatment Systems," and Basis have been deleted and relocated to the ODCM, as revised.

3.0. REASONS FOR CHANGE

These changes are consistent with GL 89-01 and, for the most part, NUREG-0473, Revision 3. Variances from NUREG-0473 reflect current monitoring practices, unique to Oyster Creek.

The requirement to shutdown the plant, if releases of liquid or gaseous effluent cannot be immediately restored to within Tech. Spec. limits, is deleted, as this is in accordance with NUREG-0473.

Other changes are administrative in nature and are required to conform the Technical Specifications and ODCM to 10 CFR 50.36(a)(2) for the periodicity of reporting, and editorial reformatting or repagination.

4.0 SAFETY EVALUATION JUSTIFYING CHANGE

These changes relocate the Oyster Creek RETS to either the ODCM or the PCP. This TSCR is consistent with the guidance contained in NRC Generic Letter No. 89-01, dated January 31, 1989 and incorporates line-item improvements in Technical Specifications.

Additionally, the proposed changes are, for the most part, consistent with the guidance of NUREG-0473.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATIONS

GPUN has determined that this Technical Specification Change Request involves no significant hazards consideration as defined by NRC in 10 CFR 50.92.

5.1 Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

The proposed amendment allows relocation of the Oyster Creek RETS to either the ODCM or PCP according to the guidance contained in GL 89-01, and conforms the Technical Specification to 10 CFR 50.36(a)(2). This proposal simplifies the RETS, meets the regulatory requirements for radioactive effluent and radiological environmental monitoring, and is provided as a line-item improvement of the Tech. Specs. Thus, this change does not increase the probability of occurrence or consequences of an accident previously evaluated.

5.2 Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposal relocates procedural details, currently included in the Tech. Spec. Amendment, on radioactive effluent, solid radioactive wastes, environmental monitoring, and associated reporting requirements to the ODCM or PCP, as appropriate. Future changes to these procedural details in the ODCM and the PCP will be handled under the administrative controls for changes to these documents. Therefore, this change has no effect on the possibility of creating a new or different kind of accident from any accident previously evaluated.

5.3 Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The procedural details being relocated to the ODCM or the PCP are consistent with the guidance provided in GL 89-01 and NUREG-0473. Therefore, it is concluded that operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

The Commission has provided guidelines on the application of the three standards by listing specific examples in 45 FR 14870. The proposed amendment is considered to be in the same category as example (vi) of amendments that are considered not likely to involve significant hazards consideration in that the proposed change conforms to changes provided in NRC Generic Letter 89-01. Implementation of the proposed amendment according to the guidance contained in NRC Generic Letter 89-01 incorporates a line-item improvement in Technical Specifications. Thus, operation of the facility in accordance with the proposed amendment involves no significant hazards considerations.

6.0 IMPLEMENTATION

It is requested that the amendment authorizing this change become effective within 60 days upon issuance to allow time for implementation of affected procedure changes.

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (cont'd)

*Issued by NRC Order dated 10-24-80

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 REPORTABLE EVENT

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

1.31 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE is that leakage which is collected in the primary containment equipment drain tank and eventually transferred to radwaste for processing.

1.32 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE is all measured leakage that is other than identified leakage.

1.33 PROCESS CONTROL PLAN (PCP)

The PROCESS CONTROL PLAN shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.34 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.

1.35 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.

1.36 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 3.6 AND 3.15, respectively; and, (2) descriptions of the information that should be included in the Annual Radioactive Effluent Release Report AND Annual Radiological Environmental Operating Report required by Specifications 6.9.1.d and 6.9.1.e, respectively.

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 primary coolant except during REFUEL MODE shall be limited to: Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT (D.E.) I-131.

Limiting Condition for Operation

1. Whenever an isotopic analysis shows reactor coolant activity exceeds 0.2 uCi/gram DOSE EQUIVALENT (D.E.) I-131, operation may continue for up to 48 hours. Additional analyses shall be done at least once per 4 hours until the specific activity of the primary coolant is restored to within its limit.

2. If the reactor coolant activity is greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram D.E. I-131, be in at least SHUTDOWN CONDITION within 12 hours.

3. Annual Reporting Requirement

   The results of specific activity analyses in which the reactor coolant exceeded the limits of Specification 3.6.A shall be reported on an annual basis. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded until after the radioiodine activity is reduced to less than the limit; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded until after the radioiodine activity is reduced to less than the limit; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and, (5) The time duration when specific activity of the primary coolant exceeded the radioiodine limit.

---

* If there are consecutive thermal power changes by more than 15% per hour, take sample and analyze at least one sample between 2 and 6 hours following the change and at least once per four hours thereafter, until the specific activity of the primary coolant is restored to within limits.

4. With the reactor mode switch in Run or Startup position, with:

   1. Thermal power changed by more than 15% of rated thermal power in one hour*, or

   2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

   3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

   take sample and analyze at least one sample, between 2 and 6 hours following the change in thermal power or off-gas level and at least once per four hours thereafter, until the specific activity of the primary coolant is restored to within limits.

3.6.B <u>Liquid Radwaste Treatment</u> - RELOCATED TO THE ODCM

3.6.C <u>Radioactive Liquid Storage</u>

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 Annual Effluent Release Report.

3. Specifications 3.0.A and 3.0.B do not apply.

3.6.D <u>Condenser Offgas Treatment</u> - RELOCATED TO THE ODCM

3.6.E <u>Main Condenser Offgas Radioactivity</u>

1. The gross radioactivity in noble gases discharged from the main condenser air ejector shall not exceed 0.21/E Ci/sec after the holdup line where 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 <u>Condenser Offgas Hydrogen Concentration</u>

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 <u>Radioactivity Concentration in Liquid Effluent</u>

RELOCATED TO THE ODCM

3.6.J <u>Limit on Dose Due to Liquid Effluent</u>

RELOCATED TO THE ODCM

3.6.K <u>Dose Rate Due to Gaseous Effluent</u>

RELOCATED TO THE ODCM

3.6.L <u>Air Dose Due to Noble Gas in Gaseous Effluent</u>

RELOCATED TO THE ODCM

3.6.M <u>Dose Due to Radioiodine and Particulates in Gaseous Effluent</u>

RELOCATED TO THE ODCM

3.6.N <u>Annual Total Dose Due to Radioactive Effluents</u>

RELOCATED TO THE ODCM

Basis:

3.6.A 10 CFR 100, as implemented by SRP Section 15.6.4, requires that the radiological consequences of failure of a main steam line outside containment be limited to small fractions of the exposure guidelines of 10 CFR 100. During Systematic Evaluation Program (SEP) for Oyster Creek, an independent assessment of the radiological consequences of a main steam line failure outside containment (SEP Topic XV-18) was performed by the NRC staff. The assessment determined that if the existing Oyster Creek Technical Specification limit for primary coolant iodine activity (8.0 uCi total iodine per gram) is used, the potential offsite doses would exceed the applicable dose limit. The staff recommended that Oyster Creek maintain the primary coolant radioiodine activity within the General Electric Standard Technical Specification (NUREG-0123) limit (0.2 uCi/gram DOSE EQUIVALENT I-131), which would meet the acceptance criteria.

However, the Staff's analyses for Oyster Creek showed that small-line failures are more limiting than the main steam line failure. 10 CFR 100, as implemented by SRP Section 15.6.2, requires that the radiological consequences of failure of small lines carrying primary coolant outside containment be limited to small fractions of the exposure guidelines of 10 CFR 100. During the evaluation of SEP Topic XV-16 "Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment" the Staff determined that Oyster Creek does not comply with current acceptance criteria. The Staff recommended that the General Electric Standard Technical Specification (NUREG-0123) limit (0.2 uCi/gram DOSE EQUIVALENT I-131) for reactor coolant radioiodine activity be adopted in order to ensure that the radiological consequences to the environment from a failure of small lines are acceptably low.

The LCO statement permitting power operation to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power. The reporting of cumulative operating time with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission to evaluate the circumstances.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3.6.B RELOCATED TO THE ODCM

3.6.C Restricting the quantity of radioactive material contained in the specific 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 mem/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.

3.6.D RELOCATED TO THE ODCM

3.6.E Some radioactive materials are 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 off-site 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 dose is due to the cloud passage over the receptor and not cloud submersion, in which the beta dose could be additive.

A subsequent licensee calculation, using ODCM methodology and based on representative 1989 and 1990 air ejector offgas - Noble gas concentrations, has established that a release rate of .34 Ci/sec would be within 10 CFR 20 limits, assuming a maximum projected $\bar{E}$ of 0.93 Mev/disintegration.

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

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.

3.6.G NOT USED

3.6.H NOT USED

3.6.I RELOCATED TO THE ODCM

3.6.J RELOCATED TO THE ODCM

3.6.K RELOCATED TO THE ODCM

3.6.L RELOCATED TO THE ODCM

3.6.M RELOCATED TO THE ODCM

3.6.N RELOCATED TO THE ODCM

3.14 Solid Radioactive Waste - DELETED

3.15 Explosive Gas Monitoring Instrumentation

Objective: The explosive gas monitoring instrumentation channels shown in Table 3.15.2 shall be OPERABLE with Alarm/Trip setpoints set to ensure that the limits of Specification 3.6.F are not exceeded.

Applicability: As shown in Table 3.15.2

Specification

A. Explosive Gas Instrumentation

1. With an explosive gas monitoring instrumentation channel Alarm/Trip setpoint less conservative than required by the Objective above declare the channel inoperable and take ACTION shown in Table 3.15.2.

2. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.15.2. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3.

3. The provisions of Specifications 3.0 and 3.1 are not applicable.

Basis:

A. The explosive gas monitoring instrumentation in Table 3.15.2 is provided 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 50. The offgas hydrogen monitor has an alarm which reports in the reactor Control Room. The offgas hydrogen monitor initiates a bypass of the Augmented Offgas System in the event the setpoint is exceeded.

TABLE 3.15.2

EXPLOSIVE 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

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) NOT USED

(c) During Augmented Offgas Treatment System operation.

(d) One hydrogen and one temperature sensor.

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.

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 reasonable achievable and within the limit of 10 CFR Part 20.106.

Specification:

A. Reactor Coolant

Reactor coolant shall be sampled and analyzed at least once every 72 hours for DOSE EQUIVALENT I-131 during RUN MODE, STARTUP MODE and SHUTDOWN CONDITION.

B. NOT USED.

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.

D. Main Condenser Offgas Treatment

RELOCATED TO THE ODCM

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 the thermal power level.

F. Condenser Offgas Hydrogen Concentration

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

G. NOT USED.

H. NOT USED.

I. Radioactivity Concentration in Liquid Effluent

RELOCATED TO THE ODCM

J. Dose due to Liquid Effluent

RELOCATED TO THE ODCM

K. Dose Rate Due to Gaseous Effluent

RELOCATED TO THE ODCM

L. NOT USED.

M. Dose Due to Radioiodine and Particulates in Gaseous Effluent

RELOCATED TO THE ODCM

N. Annual Total Dose Due to Radwaste Effluent

RELOCATED TO THE ODCM.

Basis:

A. The reactor water sample will be used to assure that the limit of Specification 3.6.A is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of several days. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant.

I. RELOCATED TO THE ODCM.

4.14 Solid Radioactive Waste - DELETED

4.15 Explosive Gas Monitoring Instrumentation

Applicability: States surveillance requirements for OPERABILITY of explosive gas monitoring instrumentation.

Objective: To demonstrate the OPERABILITY of explosive gas monitoring instrumentation.

Specification:

Gaseous Effluent Instrumentation

Each explosive gas effluent monitoring instrument channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.15.2.

TABLE 4.15.2

EXPLOSIVE GAS MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION(f)	FUNCTIONAL TEST	CHANNEL SURVEILLANCE REQUIRED(a)
1. Main Condenser Offgas Treatment System Hydrogen Monitor	D	N/A	Q(g)	M	(c)

Legend: D = once per 24 hrs; M = once per 31 days; Q = once per 92 days; N/A = Not Applicable.

TABLE 4.15.2 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.

(c) During main condenser offgas treatment system operation.

(f) The CHANNEL CALIBRATION shall be performed according to established station calibration procedures.

(g) A 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.

4.16 Radiological Environmental Surveillance

RELOCATED TO THE ODCM

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the items referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33 as referenced in the GPU Nuclear Operational Quality Assurance Program.

b. Surveillance and test activities of equipment that affects nuclear safety and radioactive waste management equipment.

c. Refueling Operations.

d. Security Plan Implementation.

e. Fire Protection Program Implementation.

f. Emergency Plan Implementation.

g. Process Control Plan Implementation

h. Offsite Dose Calculation Manual Implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1.

j. Plant Staff Overtime pursuant to Technical Specification 6.2.2.2(i), above.

6.8.2 Each procedure required by 6.8.1 above, and substantive changes thereto, shall be reviewed and approved as described in 6.5.1 prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1, above, may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of GPU Nuclear Management Staff qualified in accordance with 6.5.1.14 and knowledgeable in the area affected by the procedure. For changes which may affect the operational status of unit systems or equipment, at least one of these individuals shall be a member of unit management or supervision holding a Senior Reactor Operator's license on the unit.

c. The change is documented, reviewed and approved as described in 6.5.1 within 14 days of implementation.

6.8.4 The following programs shall be established, implemented and maintained:

a. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluent and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluent as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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

2. Limitations on the concentrations of radioactive material released in liquid effluent to UNRESTRICTED AREAS conforming to 10 CFR 20, Appendix B, Table II, Column 2.

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluent in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM.

4. Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluent released to UNRESTRICTED AREAS conforming to Appendix I of 10 CFR 50,

5. Determination of cumulative and projected dose contributions from radioactive effluent for the current calendar quarter and the current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in the 31 day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR 50,

7. Limitations on the dose rate resulting from radioactive materials released in gaseous effluent to areas beyond the EXCLUSION AREA boundary conforming to doses associated with 10 CFR 20, Appendix B, Table II, Column 1,

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents beyond the EXCLUSION AREA boundary conforming to Appendix I of 10 CFR 50,

9. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from I-131, I-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluent released beyond the EXCLUSION AREA boundary conforming to Appendix I of 10 CFR 50,

10. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from Uranium fuel cycle sources conforming to 40 CFR Part 190.

b. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

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

2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of 10 CFR, the following identified reports shall be submitted to the Administrator of the NRC Region I office unless otherwise noted.

6.9.1 ROUTINE REPORTS

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design of a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specified details required in license conditions based on other commitments shall be included in this report.

   Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Exposure Data Report. Routine exposure data reports covering the operation of the facility during the previous calendar year shall be submitted prior to March 1 of each year. Reports shall contain a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man rem exposure according to work and job functions (this tabulation supplements the requirements of 10 CFR 20.407), e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis which will include a narrative of operating experience, to the Director, Office of Management and Program Control, U.S. Nuclear Regulatory Commission, with a copy to the Regional Office, no later than the 15th of each month following the calendar month covered by the report.

d. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the operations of the unit during the previous 12 months of operation shall be submitted within 60 days after January 1, each year.

The Report shall include a summary of the quantities of radioactive liquid and gaseous effluent and solid waste released from the unit. The material provided shall be: (1) consistent with the objectives outlined in the ODCM and PCP; and, (2) in conformance with 10 CFR 50.36(a) and Section IV.B.1 of Appendix I to 10 CFR Part 50.

e. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in: (1) the ODCM; and, (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

f. CORE OPERATING LIMITS REPORT (COLR)

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

   a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.10.A

   b. The K_f core flow adjustment factor for Specification 3.10.C.

   c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.10.C

   d. The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) for Specification 3.10.B.

   and shall be documented in the COLR.

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

   a. GPU Nuclear (GPUN) Topical Report (TR) 020, Methods for the Analysis of Boiling Water Reactors Lattice Physics, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

   b. GPUN TR 021, Methods for the Analysis of Boiling Water Reactors Steady State Physics, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

c. GPUN TR 033, Methods for the Generation of Core Kinetics Data for RETRAN-02, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

d. GPUN TR 040, Steady-State and Quasi-Steady-State Methods Used in the Analysis of Accidents and Transients, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

e. GPUN TR 045, BWR-2 Transient Analysis Model Using the Retran Code, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

f. NEDE-31462P and NEDE-31462, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

g. NEDE-24011, General Electric Standard Application for Reactor Fuel, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

h. NEDE-24195, General Electric Reload Fuel Application for Oyster Creek, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

i. XN-75-55-(A); XN-75-55, Supplement 1-(A); XN-75-55, Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek Plant," April 1977

j. XN-75-36(NP)-(A); XN-75-36(NP), Supplement 1-(A), "Spray Cooling Heat Transfer Phase Test Results, ENC - 8x8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

4. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Basis: 6.9.1.e - RELOCATED TO THE ODCM

6.9.2 <u>REPORTABLE EVENTS</u>

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 removable contamination.

d. 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.

e.-j. Pursuant to the ODCM

k. Records of results of analyses required by the Radiological Environmental Monitoring Program.

l. Failures and challenges to Relief and Safety Valves which do not constitute an LER will be the subject of a special report submitted to the Commission within 60 days of the occurrence. A challenge is defined as any automatic actuation (other than during surveillance or testing) of Safety or Relief Valves.

m. Plans for compliance with standby liquid control Specifications 3.2.C.3(b) and 3.2.C.3(e)(1) or plans to obtain enrichment test results per Specification 4.2.E.5.

n. Inoperable high range radioactive noble gas effluent monitor (3.13H)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at 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. All Licensee Event 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 of Environmental Qualification which are covered under the provisions for paragraph 6.14.

l. Records of the service lives of all snubbers, including the date which the service life commences, and associated installation and maintenance records.

m. Records of results of analyses required by the Radiological Environmental Monitoring Program.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PLAN.

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 1,000 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 Director responsible for radiological controls.

6.13.2 Specification 6.13.1 shall also apply to each high radiation area in which the intensity of radiation is greater than 1,000 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 IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position of 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 preventative 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 of once every 24 months. 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:

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

*Areas requiring personnel access for establishing hot shutdown condition.

6.17 POST-ACCIDENT SAMPLING

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

A program has been established which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel in sampling and analysis.

b. Procedures for sampling and analysis.

c. Provisions for verifying operability of the System.

6.18 PROCESS CONTROL PLAN

a. GPU Nuclear Corporation initiated changes to the PCP:

   1. Shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:

      a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;

      b. a determination that the changes did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

      c. documentation that the changes have been reviewed and approved pursuant to Section 6.8.2.

   2. Shall become effective upon review and approval by GPU Nuclear Management.

6.19 OFFSITE DOSE CALCULATION MANUAL

a. The ODCM shall be approved by the Commission prior to implementation.

b. GPU Nuclear Corporation initiated changes to the ODCM shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:

   1. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;

   2. a determination that the changes did not reduce the accuracy or reliability of dose calculations or setpoint determination; and,

   3. documentation that the changes have been reviewed and approved pursuant to Section 6.8.2.

c. Change(s) shall become effective upon review and approval by GPU Nuclear Management.

6.20 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

DELETED.

Enclosure 4

OYSTER CREEK OFFSITE DOSE CALCULATION MANUAL
(2000-ADM-4532.04)

Revision 5, Errata

Page No.	Section No.	Description of Correction
p. 60	4.5.1.1	Underlines in equation missing, less than or equal to sign should be greater than or equal to.
p. 65	4.6.1.1.2.B	Referenced Sections should be 4.6.1.1.6 and 4.6.1.1.7, respectively, in place of: "4.6.1.1.5 or 4.6.1.1.6."
p. 66	4.6.1.1.4.A	Add: 10.0 mRem to any body organ during any calendar year.
p. 70	4.6.2.1.3.B	Referenced Section should be 4.6.1.1.3 in place of "4.6.1.1.7."
p. 71	4.7.1.1.1.A	Referenced Section should be 4.6.1.1.3.A in place of "4.6.1.1.4.A."
p. 74	4.7.1.2.2	Place a comma instead of a period before "Action 124"
p. 85	Att. 4532.04-16	Typo: Item 1.a. operable should be inoperable.

Enclosure 3

Offsite Dose Calculation Manual (ODCM)

Revision 5

GPU Nuclear	OYSTER CREEK POLICY AND PROCEDURE MANUAL	Number 2000-ADM-4532.04
Title Oyster Creek Offsite Dose Calculation Manual		Revision No. 5
Applicability/Scope All GPUN Employees		Responsible Office Environmental Controls 6635
This document is within QA plan scope _X_ Yes ___ No Safety Reviews Required _X_ Yes ___ No		Effective Date

Prior Revision _4_ incorporated the following Temporary Changes:

N/A

This Revision _5_ incorporates the following Temporary Changes:

N/A

List of Effective Pages

Page	Revision	Page	Revision	Page	Revision	Page	Revision
1.0	5	27.0	5	53.0	5	79.0	5
2.0	5	28.0	5	54.0	5	80.0	5
3.0	5	29.0	5	55.0	5	81.0	5
4.0	5	30.0	5	56.0	5	82.0	5
5.0	5	31.0	5	57.0	5	83.0	5
6.0	5	32.0	5	58.0	5	84.0	5
7.0	5	33.0	5	59.0	5	85.0	5
8.0	5	34.0	5	60.0	5	86.0	5
9.0	5	35.0	5	61.0	5	87.0	5
10.0	5	36.0	5	62.0	5	88.0	5
11.0	5	37.0	5	63.0	5	89.0	5
12.0	5	38.0	5	64.0	5	90.0	5
13.0	5	39.0	5	65.0	5	91.0	5
14.0	5	40.0	5	66.0	5	92.0	5
15.0	5	41.0	5	67.0	5	E1-1	4
16.0	5	42.0	5	68.0	5	E1-2	4
17.0	5	43.0	5	69.0	5	E1-3	5
18.0	5	44.0	5	70.0	5	E1-4	5
19.0	5	45.0	5	71.0	5	E1-5	5
20.0	5	46.0	5	72.0	5	E1-6	5
21.0	5	47.0	5	73.0	5	E1-7	4
22.0	5	48.0	5	74.0	5	E1-8	4
23.0	5	49.0	5	75.0	5	E1-9	4
24.0	5	50.0	5	76.0	5	E1-10	4
25.0	5	51.0	5	77.0	5	E1-11	4
26.0	5	52.0	5	78.0	5		

	Signature	Concurring Organization Element	Date
Originator	Paul E. Schwartz	Senior Environmental Scientist	18AUG93
Concurred By	James J. Vaughtas	Environmental Controls Manager (OC)	8/16/93
	[illegible]	Radiological Engineering Manager (OC)	8/16/93
	[illegible]	Manager Plant Chemistry (OC)	8/16/93
Approved By	[illegible]	Director, Radiological Controls (OC)	8/16/93

(0879P)

DOCUMENT HISTORY

REVISION	EFFECTIVE DATE	DESCRIPTION OF CHANGE	PREPARED BY: REVIEWED BY: APPROVED BY:
4	11/06/91	As a "PLAN/RELATED PROCEDURE" the ODCM needs to be formatted as per 1000-ADM-1218.01 with an appropriate ID.	P: R: A:
5	11/01/92	As per USNRC Generic Letter 89-01, the RETS and REMP sections of the OCNGS Technical Specifications have been written into the ODCM.	P: PES R: [illegible] A: RRS

Oyster Creek Offsite Dose Calculation Manual (ODCM)

Table of Contents

Oyster Creek Offsite Dose Calculation Manual (ODCM)

Table of Contents

Oyster Creek Offsite Dose Calculation Manual (ODCM)

Table of Contents

Oyster Creek Offsite Dose Calculation Manual (ODCM)

Table of Contents

LIST OF ATTACHMENTS

1.0 PURPOSE

The purpose of the Offsite Dose Calculation Manual (ODCM) is to provide information, methodology, and parameters to be used by the Oyster Creek Nuclear Generating Station (OCNGS) in the calculation of radiological doses offsite due to liquid and gaseous effluents. The ODCM is designed to provide computational methods of assessing compliance with the OCNGS operating Technical Specifications related to radioactive liquid and gaseous effluents and compliance with 10CFR20.106, 10CFR50 Appendix I and 40CFR190. Besides the areas of the ODCM mentioned above, the calculational methodology for determining liquid and gaseous effluent monitoring instrumentation alarm trip setpoints is described in detail. The pertinent information related to the Radiological Environmental Monitoring Program (REMP) including descriptions and sampling locations is also included in this manual.

2.0 APPLICABILITY/SCOPE

The Offsite Dose Calculation Manual, ODCM is applicable to those individuals responsible for assessing compliance with radioactive effluent and environmental technical specifications at the Oyster Creek Nuclear Generating Station. This manual provides the methods used in performance of dose assessment during routine situations where radioactive material has been or is predicted to be released to the environment.

3.0 DEFINITIONS

3.1 DOSE CONVERSION FACTOR (DCF)

A parameter calculated by the methods of internal dosimetry, which indicates the committed dose equivalent (to the whole body or organ) per unit activity inhaled or ingested. This parameter is specific to the isotope and the dose pathway. Dose conversion factors are commonly tabulated in units of mRem/hr per picocurie/m^3 in air or water. They can be found in the USNRC Regulatory Guidance appendices.

3.2 ELEVATED (STACK) RELEASE

An airborne effluent plume whose release point is higher than twice the height of the nearest adjacent solid structure and well above any building wake effects so as to be essentially unentrained. USNRC Regulatory Guide 1.111 is the basis of the definition of an elevated release. Elevated releases generally will not produce any significant ground level concentrations within the first few hundred yards of the source. Elevated releases generally have less dose consequence to the public due to the greater downwind distance to the ground concentration maximum compared to ground releases. All main stack releases at Oyster Creek are elevated releases.

3.3 FINITE PLUME MODEL

Atmospheric dispersion and dose assessment model which is based on the assumption that the horizontal and vertical dimensions of an effluent plume are not necessarily large compared to the distance that gamma rays can travel in air. It is more realistic than the semi-infinite plume model because it considers the finite dimensions of the plume, the radiation build-up factor, and the air attenuation of the gamma rays coming from the cloud. This model can estimate the dose to a receptor who is not submerged in the radioactive cloud. It is particularly useful in evaluating doses from an elevated plume or when the receptor is near the effluent source.

3.4 GROUND LEVEL (VENT) RELEASE

An airborne effluent plume which contacts the ground essentially at the point of release either from a source actually located at ground elevation or from a source well above the ground elevation which has significant building wake effects to cause the plume to be entrained in the wake and driven to the ground elevation. Ground

level releases are treated differently than elevated releases in that the X/Q calculation results in significantly higher concentrations at the ground elevation near the release point.

3.5 MAXIMUM PERMISSIBLE CONCENTRATION (MPC)

The maximum airborne concentration of radioactive material an individual can breathe 40 hours per week, 50 weeks per year, over a long period of time without exceeding five (5) REM/yr to the internal organs equivalent whole body exposure.

3.6 OFFSITE

The area that is beyond the exclusion area boundary where the land is neither owned, leased nor otherwise subject to control by GPU.

3.7 RAGEMS

A plant system that monitors gaseous effluent releases from monitored release points. There is a RAGEMS system for the main stack (RAGEMS I) and one for the turbine building (RAGEMS II). They monitor particulates, iodines and noble gases.

3.8 SEMI-INFINITE PLUME MODEL

Dose assessment model which is based on the assumption that the dimensions of an effluent plume are large compared to the distance that gamma rays can travel in air. The ground is considered to be an infinitely large flat plate and the receptor is located at the origin of a hemispherical cloud of infinite radius. The radioactive cloud is limited to the space above the ground plane. The semi-infinite plume model is limited to immersion dose calculations.

3.9 SOURCE TERM

The activity release rate, or concentration of an actual release or potential release. The common units for the source term are curies, curies per second, and curies per cubic centimeter, or multiples thereof (e.g, microcuries).

3.10 X/Q - ("CHI over Q")

The dispersion factor of a gaseous release in the environment calculated by a point source gaussian dispersion model. Normal units of X/Q are sec/m^3. The X/Q is used to determine environmental atmospheric concentrations by multiplying the source term, represented by Q (in units of uCi/sec or Ci/sec). Thus, the plume dispersion, X/Q (seconds/cubic meter) multiplied by the source term, Q (uCi/seconds) yields an environmental concentration, X (uCi/m^3). X/Q is a function of many parameters including wind speed, stability class, release point height, building size, and release velocity.

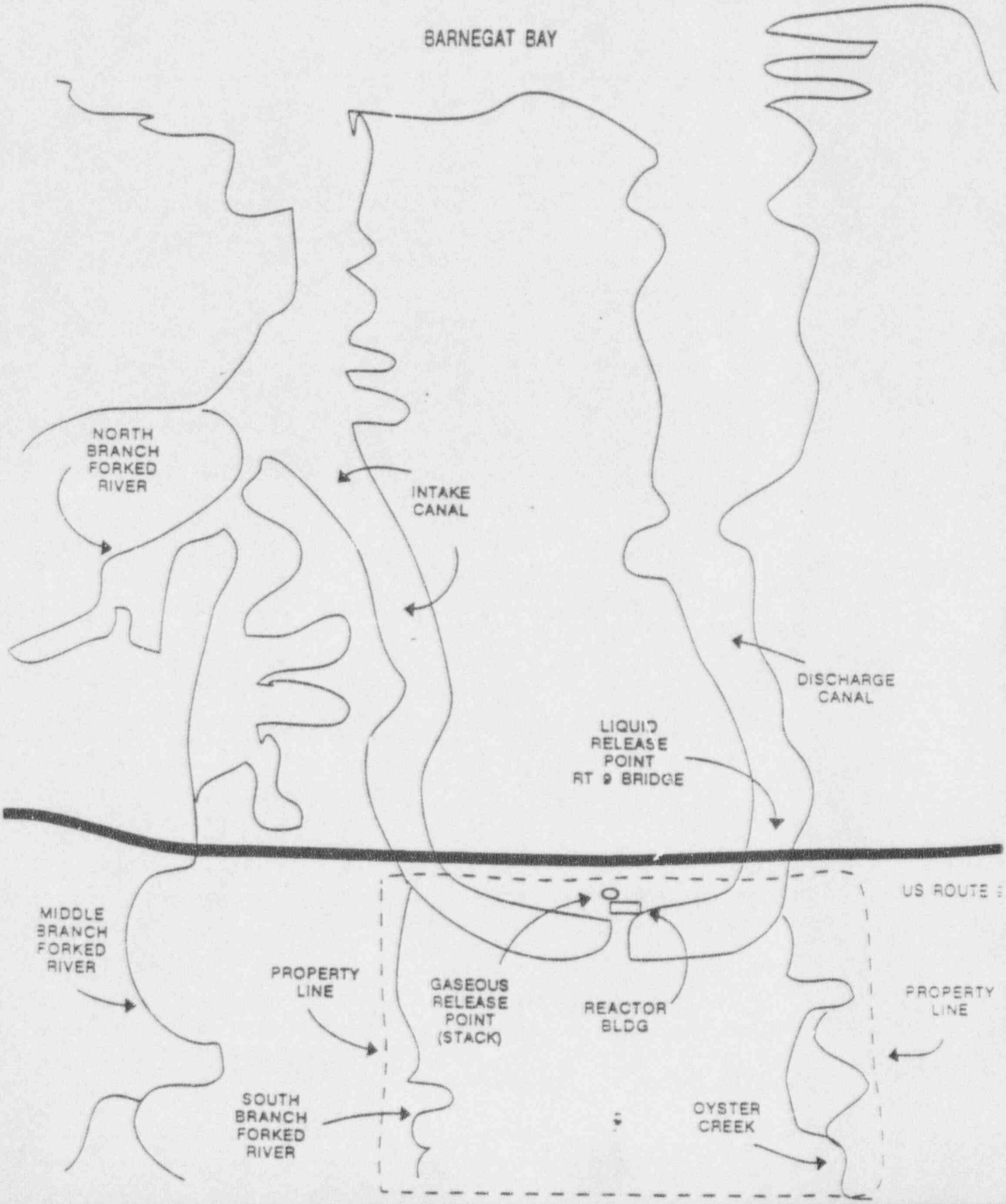
4.0 PROCEDURE

This section of the ODCM is divided into eight (8) sections, 4.1 Effluent Monitor Setpoints, 4.2 Radioactive Liquid Effluents, 4.3 Radioactive Gaseous Effluents, 4.4 Total Dose, 4.5 Radiological Environmental Monitoring Program (REMP), 4.6 Radioactive Effluents Monitoring Program, 4.7 Radioactive Effluents Monitoring Instrumentation, and 4.8 Reporting Requirements.

4.1 EFFLUENT MONITOR SETPOINTS

4.1.1 Aqueous Effluent Radioactivity Monitor Setpoints

Attachment 2000-ADM-4532.04-1
Site Boundary and Effluent Release Locations
Oyster Creek NGS



4.1.1.1 General

Section 4.7.1.1.1.A and 4.7.1.1.1.B require radioactive liquid effluent monitors to alarm/trip in the event the limit in Section 4.6.1.1.3.A is exceeded. The Reactor Building Service Water Effluent Line monitor causes an alarm in the event its setpoint is exceeded. The Liquid Radwaste Effluent Line monitor and the Turbine Building Sump no. 1-5 monitor are designed to cause a trip which halts the respective effluent discharge in the event its setpoint is exceeded.

Each monitor setpoint is derived independently to enable an alarm or trip in the event the concentration limit in 10 CFR Part 20 Appendix B, Table 2, Column 2 is exceeded at the boundary of the OCNGS restricted area, i.e., at the Route 9 bridge across the discharge canal. With prompt action to reduce the radioactive release following an alarm, GPU can be reasonably sure that radioactive material in aqueous effluent released to an unrestricted area will comply with the provisions of 10 CFR Part 20.106.

The alarm or trip setpoint of each effluent radioactivity monitor is calculated with the equation:

$$S = \frac{A}{FMPC} \cdot \frac{F_2}{F_1} \cdot g + BKG$$

where S = radiation monitor alarm setpoint (cpm)

A = counting rate (cpm/ml) or activity concentration (μCi/ml) of sample in laboratory (i.e., $A = \Sigma C_i$ or C_{gross}).

FMPC = fraction or multiple of unrestricted area MPC in aqueous effluent, based on sample analysis.

g = ratio of effluent radiation monitor counting rate to laboratory counting rate or activity concentration in a sample of liquid (cpm per cpm/ml or cpm per μCi/ml).

F_1 = flow in the batch release line (gal/min).* Value not greater than the discharge line flow alarm maximum setpoint.

F_2 = flow in the discharge canal (gal/min).* Value not less than the discharge canal minimum flow.

BKG = Monitoring instrument background (cpm)

The term, A/FMPC, represents the counting rate of a solution having the same radionuclide distribution as the sample and having the maximum permissible concentration of that mixture.

---

*Any suitable but identical units of flow (volume/time)

The term, FMPC, is the ratio of the radioactivity concentration in effluent to the unrestricted area maximum permissible concentration (MPC). It is calculated with the equation:

$$FMPC = \sum_i \frac{C_i}{MPC_i}$$

where Ci = concentration of radionuclide i in effluent, i.e., in a liquid radwaste sample tank, in reactor building service water, or in Sump no. 1-5 (μCi/ml).

MPCi = unrestricted area MPC of radionuclide i, i.e., 10 CFR Part 20, Appendix B, Table 2, Column 2 quantity for radionuclide i (μCi/ml).

In the event gross radioactivity analysis alone is used to determine the radioactivity in an effluent stream or batch, FMPC is:

$$FMPC = \frac{C}{1 \times 10^{-7}}$$

where C = the gross radioactivity in the effluent stream or batch (μCi/ml).

$1x10^{-7}$ = the unrestricted area MPC for unidentified radionuclides (μCi/ml).

If the gross activity concentration, C, is below the lower limit of detection for gross activity, the value, $1x10^{-7}$ μCi/ml, or the equivalent counting rate (cpm/ml) may be substituted for the factor A/FMPC:

$$(\text{i.e., } \frac{A}{FMPC} = 1 \times 10^{-7}\ \mu\text{Ci/ml}).$$

Usually, when the concentration of specific radionuclides is determinable in a sample(s), i.e., greater than the LLD, the alarm/trip setpoint of each liquid effluent radioactivity monitor is based upon measurement, in accord with Section 4.6.2.1.3 of radioactive material in a batch of liquid to be released or in a continuous aqueous discharge. Alternatively, a radionuclide distribution that represents the distribution expected to be in the effluent if the concentration were high enough to be detectable, i.e., greater than the LLD, may be assumed. The representative distribution may be based upon past measurements of the effluent stream or upon a computed distribution.

4.1.1.2 Batch Release

A sample of each batch of liquid radwaste is analyzed for I-131 and other principal gamma emitters or for gross beta or gross gamma activity before release. The result of the analysis is used to calculate the trip setpoint of the radioactivity monitor on the liquid radwaste effluent line to apply to release of the batch (Exhibit 2000-ADM-4532.04-5).

4.1.1.3 Continuous Release

The reactor building service water effluent and the turbine building Sump no. 1-5 are each sampled and analyzed weekly for I-131 and other principal gamma emitters. Results of analyses for the preceding week or for a period as long as the preceding 3 months are used to calculate the alarm/trip setpoint of the corresponding effluent radioactivity monitor in order to determine a representative value.

In each case, whether batch or continuous, the monitor alarm/trip setpoint may be set at lower activity concentration than the calculated setpoint.

4.1.2 Gaseous Effluent Monitor Setpoints

4.1.2.1 General

Sections 4.7.1.1.2.A and 4.7.1.1.2.B require radioactive gaseous effluent monitors to initiate an alarm in the event the dose rate beyond the site boundary due to radioactive noble gas in gaseous effluent exceeds a limit in Section 4.6.1.1.5.A. With prompt action to reduce radioactive

effluent following an alarm, the unrestricted area concentration provisions in 10 CFR Part 20.106 should not be exceeded.

Each radioactive noble gas effluent monitor setpoint is derived either on the basis of dose equivalent rate or on the basis of the 10 CFR Part 20, Appendix B, Table 2, Column 1 limit for radioactive noble gases in the unrestricted area beyond the exclusion area boundary. The monitor may be set to cause the alarm to activate at a lower concentration than the calculated setpoint.

For the purpose of deriving a setpoint, the distribution of radioactive noble gases in an effluent stream may be determined in one of the following ways:

a. Preferably, the radionuclide distribution is obtained by gamma spectrum analysis of identifiable noble gases in effluent gas samples. Results of the analyses of one or more samples may be averaged to obtain a representative spectrum.

b. In the event a representative distribution is unobtainable from recent measurements by the radioactive gaseous waste sampling and analysis program, it may be based upon past measurements or on a computed spectrum appearing in Attachment 2000-ADM-4532.04-2 herein.

c. Alternatively, the total activity concentration of radioactive noble gases may be assumed to be Xenon-133 as found in USNRC Regulatory Guideline 1.97.

Attachment 2000-ADM-4532.04-2

Distribution Of Radioactive Noble Gases in Gaseous Effluent

Radionuclide	Effluent Discharge Point					
	Stack		Turbine Building Vent		AOG Building Vent	
	(Ci/yr)	Fraction	(Ci/yr)	Fraction	(Ci/yr)	Fraction
Kr83m	205	5.40E-03	*	*	*	*
Kr85m	5620	1.50E-01	68	1.90E-02	3	1.90E-02
Kr85	140	3.70E-03	*	*	*	*
Kr87	400	1.00E-02	190	5.40E-02	3	1.90E-02
Kr88	6200	1.60E-01	230	6.60E-02	3	1.90E-02
Kr89	650	1.70E-02	*	*	*	*
Xe131m	62	1.60E-03	*	*	*	*
Xe133m	110	2.90E-03	*	*	*	*
Xe133	20000	5.20E-01	280	8.00E-02	66	4.10E-01
Xe135m	760	2.00E-02	650	1.90E-01	46	2.80E-01
Xe135	1200	3.10E-02	630	1.80E-01	34	2.10E-01
Xe137	790	2.10E-02	*	*	*	*
Xe138	2000	5.20E-02	1440	4.10E-01	7	4.30E-02
Ar41	25	6.60E-05	*	*	*	*
TOTAL	38162		3488		162	

Reference: United States Nuclear Regulatory Commission - Off. Std. Dev., April 1976, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR GALE Code), NUREG-0016.

4.1.2.2 Setpoint Based on Dose Equivalent Rate

The alarm setpoint of a radioactive noble gas effluent monitor may be calculated on the basis of whole body dose equivalent rate offsite. A setpoint of a monitor of an elevated release, e.g., from the stack, may be calculated with the equation:

$$S = 1.06 \frac{h}{f} \sum_i \frac{C_i}{(C_i \cdot DFS_i)} + BKG$$

The setpoint of a monitor of a ground-level or split-wake release, e.g., from the turbine building vent or the AOG building, may be calculated with the equation:

$$S = 1.06 \frac{h}{f \frac{X}{Q}} \sum_i \frac{C_i}{(C_i \cdot DFV_i)} + BKG$$

where S = the alarm setpoint (cpm) or (mR/hr)

h = monitor response to activity concentration of effluent being monitored, $\frac{cpm}{\mu Ci/cm^3}$ or $\frac{mR/hr}{\mu Ci/cm^3}$

C_i = relative concentration of noble gas radionuclide i in effluent at the point of monitoring ($\mu Ci/cm^3$)

X/Q = atmospheric dispersion from point of ground-level or split-wake release to the location of potential exposure (sec/m^3)

1.06 = $\frac{\text{500mrem/year (Sect. 4.3.1.1)}}{\text{472 (conversion of cfm to cc/sec)}}$

DFS_i = factor converting elevated release rate of radionuclide i to total body dose equivalent rate at the location of potential exposure (Attachment 2000-ADM-4532.04-3).

$$\frac{\text{mrem}}{\text{yr.} \frac{\mu\text{Ci}}{\text{sec}}}$$

DFV_i = factor converting ground-level or splitwake release of radionuclide i to the total body dose equivalent rate at the location of potential exposure (Attachment 2000-ADM-4532.04-3).

$$\frac{\text{mrem}}{\text{yr.} \frac{\mu\text{Ci}}{\text{m3}}}$$

f = flow of gaseous effluent stream being monitored, i.e., stack flow, vent flow, etc. (ft^3/min).

BKG = Monitoring instrument background (cpm or mR/hr)

Each monitoring channel has a unique response, h, which is determined by the instrument calibration. The concentration of each noble gas radionuclide i in a gaseous effluent is determined as discussed already (see Attachment 2000-ADM-4532.04-2).

The atmospheric dispersion, X/Q, and the dose conversion factor, DFS_i, depend upon local conditions. For the purpose of calculating radioactive noble gas effluent monitor alarm setpoints appropriate for the OCNGS, the locations of maximum potential offsite exposure and the reference atmospheric dispersion factors applicable to the derivation of setpoints are:

Discharge Point	Receptor Location Sector	Receptor Location Distance(m)	Atm. Dispersion (sec/m3)
Ground-level or vent	SE	522	2.15 E-5
Stack	SE	522	N/A

The applicable dose conversion factors, DFS_i and DFV_i, for deriving setpoints are in Attachment 2000-ADM-4532.04-3.

Attachment 2000-ADM-4532.04-3

Dose Conversion Factors for Deriving Radioactive Noble Gas Effluent Monitor Setpoints

Radionuclide	Factor DFSi for Stack Release* (mRem-second / uCi-year)	Factor DFVi for Ground-level or Split-Wake Release** (mRem-cubic meters / uCi-year)
Kr83m	1.47E-09	7.56E-02
Kr85m	9.12E-05	1.17E+03
Kr85	1.47E-06	1.61E+01
Kr87	4.80E-04	5.92E+03
Kr88	1.18E-03	1.47E+04
Kr89	1.17E-03	1.66E+04
Kr90	*******	1.56E+04
Xe131m	2.10E-05	9.15E+01
Xe133m	1.64E-05	2.51E+02
Xe133	1.57E-05	2.94E+02
Xe135m	2.77E-04	3.12E+03
Xe135	1.51E-04	1.81E+03
Xe137	1.06E-04	1.42E+03
Xe138	7.63E-04	8.83E+03
Xe139	1.44E-04	5.02E+03
Ar41	9.11E-04	8.84E+03

* Based on reference meteorology applicable at 522 meters SE of stack.

** For exposure to a semi-infinite cloud of noble gas.

4.1.2.3 Setpoint Based on Concentration

Alternatively, the alarm setpoint of an effluent noble gas monitor may be calculated on the basis of the unrestricted area concentration limit for radioactive noble gases. The equation used to calculate a setpoint on this basis is:

$$S = \frac{MPC \cdot h}{(4.7 \times 10^{-4})\ (f)\ (X/Q)} + BKG$$

where

- S = alarm counting rate setpoint (cpm) or (mR/hr)
- h = effluent noble gas monitor counting rate response $\frac{cpm}{\mu Ci/cm^3}$ or calibration $\frac{mR/hr}{\mu Ci/cm^3}$
- f = discharge rate of gaseous effluent (ft^3/min)
- X/Q = atmospheric dispersion from release point to unrestricted area ($\mu Ci/M^3$ per μCi/sec)
- 4.7×10^{-4} = conversion constant $\frac{1\ m^3}{35.31\ ft^3} \cdot \frac{1\ min}{60\ sec}$
- MPC = unrestricted area maximum permissible concentration for the effluent noble gas mixture, i.e., 10 CFR Part 20, Appendix B, Table 2, Column 1 limit for a mixture ($\mu Ci/cm^3$)

The MPC of noble gas is then calculated from the distribution with the equation:

$$MPC_i = \sum C_i \text{ divided by } \frac{Ci}{MPC_i}$$

where C_i = relative concentration of noble gas radionuclide i in gaseous release ($\mu Ci/cm^3$)

MPC_i = 10 CFR Part 20 Appendix B, Table 2, Column 1 value.

Note that this is simply the aggregate of the concentrations of radionuclides i in a sample divided by the sum of fractions of MPC constituted by radionuclides i in the same sample. In the event the distribution of radioactive noble gases is based on a computed distribution appearing in Attachment 2000-ADM-4532.04-2, the MPC will be:

MPC = 8×10^{-8} $\mu Ci/cm^3$ for noble gases released via the stack

MPC = 1×10^{-7} $\mu Ci/cm^3$ for noble gases released via the Turbine Building vent.

MPC = 2×10^{-7} $\mu Ci/cm^3$ for noble gases released via the AOG Building vent.

Alternatively, the total activity concentration of the noble gases may be used with the MPC value of Xe-133 (2×10^{-8} $\mu Ci/cm^3$) for the purpose of conservatively determining an activity concentration of noble gases that will be less than the 10 CFR 20 Appendix B, Table 2, Column 1 limit. If this approach is used, the value of MPC is simply 2×10^{-8} $\mu Ci/cm^3$. The value of atmospheric dispersion used to derive a setpoint based on concentration is the reference atmospheric dispersion value from the discharge point to the location of maximum potential exposure offsite. The applicable reference values from Exhibit 2000-ADM-4532.04-3 are:

Locations of Maximum X/Q Offsite

Discharge Point	Receptor Location Sector	Receptor Location Distance (m)	Atm Dispersion (sec/m^3)
Ground-level or vent	SE	522	2.15 E-5
Stack	SE	522	3.07 E-8

4.2 RADIOACTIVE LIQUID EFFLUENT

4.2.1 Concentration - 10 CFR 20.106 Limits

The concentration of a known mixture as a fraction of the allowable limit should be derived by determining, for each radionuclide in the mixture, the sum of the ratios of the concentration in the mixture present in the unrestricted area and the limits otherwise established in 10 CFR Part 20, Appendix B, Table II, Column 2 for the specific radionuclide when not in a mixture. The following equation is used to calculate the fraction of the maximum permissible concentration in the discharge canal at the Route 9 bridge as required in Section 4.6.1.1.3.

$$MPC_{frac} = \sum_{i=1}^{n} (C_i/MPC_i)^* \; F_1/F_2$$

where:

F_1 = flow in the discharge line (where the concentration discharged is C_i) (gallons/minute)*.

F_2 = flow in the discharge canal (gallons/minute)*.

C_i = the concentration of radionuclide, i, in the liquid effluent discharged (μCi/cc), before dilution.

MPC_i = the maximum permissible concentration of the ith radionuclide in an unrestricted area according to 10 CFR 20, Appendix B, Table II, Column 2.

MPC_{frac} = the ratio of the concentration present in the unrestricted area to the Appendix B limit for the mixture of radionuclides.

4.2.2 Liquid Effluents - 10 CFR 50 Appendix I Limits

Radiation dose from liquid effluent may be received through the ingestion of fish, shellfish, and from direct shoreline exposure. Personal radiation exposure via other aqueous pathways is negligible at Oyster Creek. The method used to assess compliance with Section 4.6.1.1.4 for exposure via fish and shellfish consumption and irradiation by shoreline deposits is described hereafter.

---

* any suitable but identical units (vol/time)

The dose due to eating fish and shellfish is calculated with the equation:

$$DOSE_{aij} = 10^{9} \frac{F_1}{F_2 A} \Delta T \sum_i C_i \exp(-\gamma_i t_p) (Bf_i \cdot Uf_a + Bs_i \cdot Us_a) D_{aipj}.$$

The dose due to irradiation by shoreline deposits is calculated with the equation:

$$DOSE_{aij} = 10^{11} \frac{F_1}{F_2 A} \Delta T \, W \sum_i C_i \, H_i \exp(-\gamma_i t_p)[1-\exp(-\gamma_i t_b)] \, Uh \cdot D_{aipj}$$

where:

$DOSE_{aij}$ = radiation dose commitment to organ, j, including total body, of a person in the most restrictive age group, a (mrem).

C_i = concentration of radionuclide, i, in the undiluted liquid waste stream or batch discharged (μCi/ml).

Bf_i, Bs_i = the equilibrium bioaccumulation factor for radionuclide, i, in fish or shellfish, expressed as the ratio of the concentration in the biota (pCi/Kg) to the radionuclide concentration in water (pCi/l), (i.e., liters/kg), USNRC Reg. Guide 1.109, Table A-1.

D_{aipj} = the dose conversion factor, specific to a given age group, a, radionuclide, i, pathway, p, and organ, j. It is used to calculate the radiation dose from an intake of a radionuclide (mRem/pCi), USNRC Reg. Guide 1.109, Tables E-7 through E-14, or from the exposure to a given concentration of a radionuclide in sediment. The latter is expressed as the ratio of the dose rate (mRem/hr) to the area radionuclide concentration (pCi/m^2), USNRC Reg. Guide 1.109, Table E-6.

F_1 = release rate of undiluted liquid waste stream or batch (vol/time)*.

F_2 = flow in discharge canal (vol/time)*.

A = factor of dilution from discharge canal to vicinity of shoreline or location of catching fish or shellfish (unitless). This factor is applicable at all three receptor points where dose is determined and is based on the ratio of the percent of non-discharge dilution to the recirculation factor:

From the discharge canal to U.S.Route 9 bridge, A = .266
From the discharge canal to the Barnegat Bay, A = 6.6
From the discharge canal to the Atlantic Ocean, A = 26.6.

ΔT = duration of release of effluent having concentration C_i, (yr).

$10^9 = 10^6 \frac{pCi}{\mu Ci} \cdot 10^3 \frac{milliliters}{liter}$

$10^{11} = 10^6 \frac{pCi}{\mu Ci} \cdot 10^3 \frac{milliliters}{liter} \cdot 10^2 \frac{liter}{m^2}$

γ_i = radioactive decay constant (hr^{-1})

t_b = the period of time for which sediment or soil is exposed to the contaminated water (hours).

H_i = the half-life of the radionuclide, i (years).

t_p = the average transit time required for nuclides to reach the point of exposure. For internal dose, t_p is the total time elapsed between radionuclide release and ingestion of food or water (hours).

Uf_a, Us_a, Uh = the usage factor that specifies the exposure time or intake rate for an individual of age group, a, associated with fish, shellfish, and shoreline irradiation (kg/yr and hr/yr), Exhibit 2000-ADM-4532.04-4, herein.

---

* any suitable but identical units of flow.

W = shore width factor (dimensionless); ref. Reg. Guide 1.109, Table A-2.

At the intake or discharge canal, W = 0.1

Along Barnegat Bay shores, W = 0.25

Along Atlantic Ocean shores, W = 1.0

The factor of dilution, A, incorporates a factor of 3.76 to account for recirculation of water through the intake and discharge canal (Reference 6.5.8). A percentage of dilution of non-discharge water is used at each receptor. For example, 25 percent dilution is assumed in the Barnegat Bay while one-hundred percent dilution is assumed in the Atlantic Ocean. The factor of dilution, A, is the ratio of the dilution to the recirculation factor. Dilution at the U.S. Route 9 bridge is assumed to be minimal and, A, is simply the inverse of the recirculation factor.

Liquid effluent concentration values that are less than LLD values are excluded from the dose calculations.

For the purpose of assessing compliance with Section 4.6.1.1.4, the most exposed person is assumed to be a member of the public who is exposed to direct shoreline exposure and who consumes fish and shellfish taken from the canal.

4.2.3 Alternative Calculation Parameters

In the case where the values of the site specific parameters are not applicable, the following parameters from USNRC Reg. Guide 1.109, Rev. 1 are used in lieu of the site specific data:

4.2.3.1. Usage Factor - Maximum age factor for ingestion of fish and seafood* and shoreline exposure (Table E-5).

4.2.3.2 Shore Width Factor - Table A-2 of Regulatory Guide USNRC Reg. Guide 1.109, and described in Section 4.2.2.

* Includes all shellfish and fish.

4.2.4 Projected Dose - Liquid

When aqueous effluent is released from the liquid radwaste management system without treatment during a calendar quarter, the equation used subsequently to project the radiation dose to a member of the public due to aqueous effluent is:

$$Dose_{PROJ} = \frac{X}{Y} * D_{PREC}$$

where X = the number of days during the period of projection.

Y = the number of days during which D_{PREC} applies.

D_{PREC} = the dose during a past time period judged to represent anticipated releases (mrem).

The preceding time period on which actual release and dose is based may be the same quarter to date or may be an earlier period which is judged to represent anticipated releases.

4.3 RADIOACTIVE GASEOUS EFFLUENT

4.3.1 Dose Rate - 10 CFR Part 20

4.3.1.1 Noble Gases

The dose rate limit beyond the Site Boundary (ref. FSAR Fig. II-2-2) resulting from radioactive noble gases in airborne effluent from the OCNGS is 500 mrem/yr to the total body and 3000 mrem/yr to skin. Compliance is monitored by radioactive gaseous effluent monitors with their alarm setpoint set to cause automatic alarm when or before a limit in Section 4.6.1.1.5.A is exceeded, thus further calculation is not required to demonstrate compliance. The setpoint is calculated in accordance with Section 4.1.2 herein and the whole body dose rate limit is used as being most conservative. Thus further calculation is not required to demonstrate compliance as long as a noble gas effluent monitor does not alarm.

In the event a noble gas effluent monitor exceeds its alarm setting and alarms, an assessment of compliance with Section 4.6.1.1.5.A is performed by either

(1) observing whether the radioactivity recorded by the noble gas monitor exceeded its derived setpoint (limit),

or

(2) by calculating the noble gas radiation-to-dose equivalent rate with the following equations.

$$Q_i = \frac{Q \cdot f_i}{T}$$

where Q = quantity of gas released as measured by gross noble gas radioactivity monitor (μCi)

f_i = fraction of noble gas radionuclide i within the quantity release

T = averaging time of noble gas discharge (hr)

Gross activity, Q, measured by the noble gas effluent monitor during recording time interval, T, normally one hour, represents the quantity of noble gas released. The noble gas distribution, f_i, is estimated as described in Section 4.1.2.1.

4.3.1.2 Total Body Dose Rate - The total body dose equivalent rate from radioactive noble gases discharged from an elevated point (stack above building wake) is calculated with the equation

$$DG = \sum_i Q_i \cdot P\gamma s_i$$

From a ground-level release (building vent) the total body dose equivalent rate is

$$DG = \frac{X}{Q_v} \sum_i Q_i \cdot P\gamma v_i$$

where DG = total body dose equivalent rate due to irradiation by radioactive noble gas (mrem/hr)

Q_i = average discharge rate of noble gas radionclide i released during the averaging time (μCi/hr)

$P\gamma V_i$ = factor converting time integrated, ground-level concentration of noble gas nuclide i to total body dose $\frac{\text{mrem}}{(\mu\text{Ci-sec})/\text{m}^3}$.

See Attachment 2000-ADM-4532.04-4.

$\frac{X}{Q_v}$ = atmospheric dispersion factor from the Plant to the offsite location of interest (sec/m^3)

$P\gamma S_i$ = factor converting unit noble gas nuclide i stack release to total body dose at ground level received outdoors from the overhead plume (mrem/μCi). See Attachment 2000-ADM-4532.04-4.

4.3.1.3 Skin Dose Rate - The dose equivalent rate to skin from radioactive noble gases is calculated by assuming a person at ground level is immersed in and irradiated by a semi-infinite cloud of the noble gases originating in airborne effluent. It is calculated for each air effluent discharge point with the equation

$$DB = \frac{X}{Q} \sum_i Q_i \,(SB_i + 1.11 \cdot Ay\gamma V_i)$$

where DB = dose rate to skin from radioactive noble gases (mrem/hr)

$\frac{X}{Q}$ = atmospheric dispersions from gaseous effluent discharge point to ground-level location of interest (sec/m^3)

Q_i = discharge rate of noble gas radionuclide i (μCi/hr)

SB_i = factor converting time integrated ground-level concentration of noble gas radionuclide i to skin dose from beta radiation $\frac{\text{mrem}}{\mu\text{Ci} \cdot \frac{\text{sec}}{m^3}}$

$A\gamma V_i$ = factor for converting time integrated, semi-infinite concentration of noble gas radionuclide i to air dose from its gamma radiation $\frac{\text{mrad}}{\mu\text{Ci} \cdot \frac{\text{sec}}{m^3}}$

The dose equivalent rate due to irradiation by effluent noble gases is normally evaluated at the same location and with the same reference meteorology data as used to calculated noble gas effluent monitor setpoints (sec. 4.1.2.2). Alternatively, meteorological dispersion data during the noble gas discharge being examined and the consequent location of most exposure may be used as the basis for evaluation.

The noble gas plume gamma-to-total body dose factors, $P\gamma S_i$ at designated locations are derived from meteorological dispersion data with the USNRC RABFIN software computer code or similar computer program implementing Regulatory Guide 1.109, Appendix B. The noble gas semi-infinite cloud gamma-to-total body dose factors, $P\gamma S_i$, are derived from NRC Regulatory Guide 1.109, Revision 1, Table B-1, Column 5.

The noble gas beta radiation-to-skin-dose factors, SB_i and the noble gas gamma-to-air dose factors, $A\gamma V_i$, are derived from NRC Regulatory Guide 1.109, Revision 1, Table B-1, columns 3 and 4 respectively. A tabulation of these factors used to compute noble gas-to-dose equivalent rate at 522 meters SE of the OCNGS is in Attachment 2000-ADM-4532.04-4.

The dose equivalent rate is calculated with the following meteorological dispersion data:

Discharge Point	Receptor Location Sector	Distance (m)	Atm. Dispersion (sec/m^3)
Ground Level or vent	SE	522	2.15 E-5
Stack	SE	522	3.07 E-8

Attachment 2000-ADM-4532.04-4

Noble Gas Radionuclide-to-Dose Equivalent Rate Factors*

Radionuclide	P Si** (mRem / uCi)	P Vi*** (mRem-m3 / uCi-sec)	A Vi*** (mRad-m3 / uCi-sec)	SBi*** (mRem-m3 / uCi-sec)
Kr83m	4.66E-17	2.40E-09	6.13E-07	*****
Kr85m	2.89E-12	3.71E-09	3.90E-05	4.63E-05
Kr85	4.66E-14	5.11E-07	5.46E-07	4.25E-05
Kr87	1.52E-11	1.88E-04	1.96E-04	3.09E-04
Kr88	3.73E-11	4.67E-04	4.83E-04	7.52E-05
Kr89	3.70E-11	5.27E-05	5.49E-04	3.21E-04
Kr90	*	4.95E-04	5.17E-04	2.31E-04
Xe131m	6.65E-13	2.90E-06	4.95E-06	1.51E-05
Xe133m	5.20E-13	7.97E-06	1.04E-05	3.16E-05
Xe133	4.97E-13	9.33E-06	1.12E-05	9.71E-06
Xe135m	8.78E-12	9.90E-05	1.07E-04	2.26E-05
Xe135	4.78E-12	5.75E-05	6.10E-05	5.90E-05
Xe137	3.36E-12	4.51E-05	4.79E-05	3.87E-04
Xe138	2.42E-11	2.80E-04	2.92E-04	1.31E-04
Xe139	4.56E-12	*****	*****	*****
Ar41	2.89E-11	2.81E-04	2.95E-04	8.54E-05

* All of these dose factors apply out-of-doors.

** Based on reference meteorology at 522 meters SE of effluent stack.

*** Derived from USNRC Regulatory Guide 1.109, Revision 1, Table B-1.

4.3.1.4 Radioactive Iodines and Particulates

The dose rate offsite due to the airborne release of I-131, I-133, tritium, and particulates with half-lives greater than 8 days is limited to no more than 1500 mrem/yr to any organ. Evaluation of compliance with Section 4.6.1.1.5.B is based on the sampling and analyses specified in Attachment 2000-ADM-4532.04-15. Since the dose rate cannot be resolved within less than the sample integration or compositing time, the contribution of each radionuclide to the calculated dose rate will be averaged no more than 3 months for H-3, Sr-89, Sr-90, and alpha-emitting radionuclides and no more than 31 days for other radionuclide. These are their usual sample integration or compositing times.

The equation used to assess compliance of radioiodine, tritium, and radioactive particulate releases with the dose rate limit is:

$$DR_p = \sum_e \sum_{i=1}^{n} TA_{eai} \cdot Q_{ei} \, (\overline{X/Q})_e$$

where:

DR_p = the average dose rate to an organ via exposure pathway, p (mrem/yr).

TA_{eai} = inhalation dose factors due to intake of radionuclide i via pathway e (mrem/yr per $\mu Ci/m^3$) to the most restrictive age group a (USNRC Reg. Guide 1.109).

$\overline{(X/Q)}_e$ = annual average relative airborne concentration at an offsite location due to a release from either the Stack or a vent, i.e. release point, e (sec/m^3).

Q_{ei} = release rate of radionuclide i from release point, e during the period of interest ($\mu Ci/sec$).

The location of the maximum ground-level concentration originating from a vent release will differ from the maximum ground-level concentration from a stack release. When assessing compliance with Section 4.6.1.1.5.B for tritium, iodine, and particulate, the air dispersion (X/Q) value for a vent release is used. The appropriate location and corresponding air dispersion factor are found in Exhibit 2000-ADM-4532.04-3.

Location of Maximum Exposure Rate by Inhalation

Discharge Point	Receptor Location		Atm. Dispersion
	Sector	Distance (m)	(sec/m3)
Ground Level or vent	SE	522	2.15 E-5
Stack	SE	522	3.07 E-8

Alternatively, inhalation exposure to effluent from the stack may be evaluated at the closest hypothetical individual located at:

Stack	SE	966	2.39 E-8

4.3.2 DOSE

Doses resulting from the release of noble gases, and radioiodines and particulates must be calculated to show compliance with Appendix I of 10CFR50. Calculations will be performed at least monthly for all gaseous effluents as stated in Sections 4.6.2.1.6 and 4.6.2.1.7 to verify that the dose to air is kept below the limits specified in Section 4.6.1.1.6 and the dose to members of the public is maintained below the limits specified in Section 4.6.1.1.7.

4.3.2.1 Noble Gases

The method used to calculate the air dose at the critical location due to noble gas is described by the following equations. The limits are provided in Section 4.6.1.1.6 for air dose offsite due to gamma and beta radiations from effluent noble gas.

a) For Gamma Radiation

$$\text{Dose } \gamma = \sum_{i=1}^{n} 3.17E\text{-}8 \quad A\gamma V_i \ \overline{(X/Q)}_v Q_{vi} + A\gamma S_i \ Q_{si}$$

b) For Beta Radiation

$$\text{Dose } \beta = 3.17E\text{-}8 \sum_{e} \sum_{i=1}^{n} A\beta_i \ \overline{(X/Q)}_e Q_{ei}$$

where:

Dose γ = the gamma dose during any specified time period (mRad).

Dose β = the beta dose during any specified time period (mRad).

$A\gamma V_1$ = the air dose factor due to ground level gamma emissions for each identified noble gas radionuclide, i; from USNRC Reg. Guide 1.109 Table B-1 (mRad/yr per $\mu Ci/m^3$).

$A\gamma S_1$ = the factor for air dose at ground level due to irradiation for an airborne plume resulting from a Stack release (mrad per μCi), Attachment 2000-ADM-4532.04-5.

$A\beta_1$ = the air dose factor due to beta emissions for each identified noble gas radionuclide, i from USNRC Reg. Guide 1.109, Table B-1 (mRad/yr per $\mu Ci/m^3$).

$\overline{X/Q}_e$ = the annual average relative concentration for areas at or beyond the exclusion area boundary for releases from either the Stack or ground release at the critical location (sec/m^3), Exhibit 2000-ADM-4532.04-3.

Q_{vi} = amount of radionuclide i released from vents (μCi).

Q_{si} = amount of radionuclide i released from the Stack (μCi).

Q_{ei} = amount of radionuclide i released from release point e(μCi).

3.17E-8 = The inverse of the number of seconds in a year.

Noble gases are continuously released from the Reactor Building vent and Stack. The quantity of

noble gas radionuclides released will be determined from the continuous noble gas monitors and periodic isotopic analyses. Noble gases discharged from either the stack or from building vents and dispersed by reference meteorology (Exhibit 2000-ADM-4532.04-3) produces its maximum gamma radiation dose-to-air offsite at 522 meters SE of the OCNGS. Values of $A\gamma S_i$ depend upon the meteorological conditions and the location of exposure and are calculated using the NRC RABFIN code or similar one in accordance with Reg. Guide 1.109, Rev. 1, Appendix B, Section 1. $A\gamma V_i$ and $A\beta_i$ are derived from USNRC Reg. Guide 1.109, Rev. 1, Table B-1 for a semi-infinite cloud, independent of meteorology or location. Values of $A\gamma S_i$, $A\gamma V_i$ and $A\beta_i$ used to calculate the noble gas radiation dose to air at 522 meters SE of the OCNGS are in Attachment 2000-ADM-4532.04-5. Reference atmospheric dispersion from the OCNGS to 522 meters SE is:

Discharge Point	Receptor Location		Atm. Dispersion
	Sector	Distance (m)	(sec/m3)
Ground Level or vent	SE	522	2.15 E-5
Stack	SE	522	3.07 E-8

Attachment 2000-ADM-4532.04-5

Air Dose Conversion Factors for Effluent Noble Gas*

Radionuclide	$A_{\gamma}Si$** mRad / uCi	$A_{\gamma}Vi$*** mRad-m3 / uCi-sec	ABi*** mRad-m3 / uCi-sec
Kr83m	9.35E-15	6.13E-07	9.14E-06
Kr85m	3.03E-12	3.90E-05	6.25E-05
Kr85	4.94E-14	5.46E-07	6.19E-05
Kr87	1.60E-11	1.96E-04	3.27E-04
Kr88	3.93E-11	4.83E-04	9.30E-05
Kr89	3.90E-11	5.49E-04	3.37E-04
Kr90	*	5.17E-04	2.49E-04
Xe131m	7.62E-13	4.95E-06	3.52E-05
Xe133m	5.86E-13	1.04E-05	4.70E-05
Xe133	5.45E-13	1.12E-05	3.33E-05
Xe135m	9.32E-12	1.07E-04	2.35E-05
Xe135	6.18E-12	6.10E-05	7.81E-05
Xe137	3.55E-12	4.79E-05	4.03E-04
Xe138	2.54E-11	2.92E-04	1.51E-04
Xe139	4.82E-12	*****	*****
Ar41	3.03E-11	2.95E-04	1.04E-04

* To balance time units, when using $A_{\gamma}Vi$ or ABi from the above table, omit the 3.17E-8 time conversion from the associated formulae.

** Based on reference meteorology at 522 meters SE of effluent stack

*** Derived from USNRC Regulatory Guide 1.109, Revision 1, Table B-1.

4.3.2.2 Radioiodine, Particulates, and Other Radionuclides

The maximum dose to an individual from radioiodines and radioactive particulates with half-lives of greater than eight days in gaseous effluents released to unrestricted areas is determined with the equation:

$$Dose_{1P} = \sum_{e} \sum_{i=1}^{n} Q_{ei} \overline{X/Q}_e R_{ii} + 3.17E\text{-}8\ Q_{ei} \overline{D/Q}_e (R_{gi}+R_{vi})$$

where:

$Dose_{1P}$ = the calculated dose to an individual from radioiodines and radioactive particulates with half-lives of greater than eight days (mRem).

R_{ii} = the dose factor for each identified radionuclide i, other than noble gases for the Inhalation pathway (mRem/sec per $\mu Ci/m^3$), derived from USNRC Reg. Guide 1.109, Tables E-7 through E-10.

R_{gi} = the dose factor for each identified radionuclide, i, other than noble gases for the Ground-plane pathway (m^2-mRem/yr per μCi/sec) derived for USNRC Reg. Guide 1.109, Table E-6.

R_{vi} = the dose factor for each identified radionuclide, i, other than noble gases for the <u>Vegetation pathway</u> (m^2-mRem/yr per µCi/sec) derived from USNRC Reg. Guide 1.109, Tables E-11 through E-14.

$\overline{X/Q}_e$ = the annual average relative concentration for areas at or beyond the site boundary for releases from either the Stack or vent release at the critical location (sec/m^3), Exhibit 2000-ADM-4532.04-3 or the annual average for the 12-month period of evaluation.

$\overline{D/Q}_e$ = the annual average relative deposition for areas at or beyond the site boundary for releases from either the Stack or vent release at the critical location on the ground (m^{-2}), Exhibit 2000-ADM-4532.04-3 or the annual average for the 12-month period of evaluation.

Q_{ei} = the quantity of each radionuclide, i, released from release point, e (µCi).

3.17E-8 = The inverse of the number of seconds in a year.

As per Reg. Guide 1.109, Rev. 1, only half of the iodine released is considered elemental and is used for deposition calculations.

Environmental pathways that radioiodine and particulates in airborne effluent follow to the member of the public experiencing the most confirmed exposure as determined by the annual land use survey and reference meteorology will be evaluated. The seasonality of exposure pathways may be considered. For instance, if the most exposed receptor has a garden, fresh and stored vegetables are assumed to be harvested and eaten during April through October. Fresh vegetables need not be considered as an exposure pathway during November through March.

Alternatively, the dose may be calculated to correspond with residence or exposure at an acceptable location where calculated exposures are unlikely to underestimate those experienced by the most exposed member of the public. That is, conditions assumed in the dose assessment may be realistic or may be more conservative than actual conditions.

To assess compliance with Section 4.6.1.1.7.A, the dose due to radioactive iodine and particulates in airborne effluent is calculated to a person residing 966 meters SE of the OCNGS where reference atmospheric dispersion and deposition factors (Exhibit 2000-ADM-4532.04-3) are:

Discharge Point	Depleted Atm Dispersion Xd/Q (sec/m3)	Deposition D/Q (1/m2)
Ground Level or vent	1.16E-5*	1.37E-8
Stack	2.39E-8*	6.38E-9

* Xd/Q assumed to equal X/Q

The environmental pathways of exposure to be evaluated are identified in Attachment 2000-ADM-4532.04-6.

4.3.3 Projection of Doses

The method used to project radiation dose from the Ventilation Exhaust Treatment System on releases to offsite environs during the preceding quarter is:

$$Dose_{PROJ} = \frac{X}{Y} * D_{PREC}$$

where:

X = the number of days during the period of projection.

Y = the number of days during which D_{PREC} applies.

D_{PREC} = the dose during a past time period that is judged to be representative of anticipated releases (mrem).

The projection of dose is not necessary for continuous releases from the Stack because Section 4.6.1.1.2 requires the gaseous radwaste treatment system (offgas holdup system) to be operated at all times and batch releases are not performed.

4.4 TOTAL DOSE - COMPLIANCE WITH 40 CFR 190.10a

4.4.1 General

Section 4.6.1.1.8 requires that in the event the calculated doses associated with the effluent releases exceed twice the limits of either Specification:

3.6.J Dose due to Liquid Effluent

3.6.L Air dose due to Noble Gas in Gaseous Effluent, or

3.6.M Dose due to Radioiodine and Particulates in Gaseous Effluent.

OCNGS will assess compliance with the limit on annual dose to a Member of the Public stated in Section 4.6.1.1.8.A.

To perform this assessment, the total body and organ doses resulting from liquid effluents from the OCNGS will be summed with the doses resulting from releases of radioiodines and particulates in airborne effluents. The total body dose attributable to (external) irradiation by noble gases or directly from the Station will be added with the doses due to internal exposure. These dose estimates will be based on releases from the OCNGS during the calendar year to date including the calendar quarter in which twice the limit of Section 4.6.1.1.4.A, or 4.6.1.1.7.A was exceeded.

Dose estimates to assess compliance with 40 CFR Part 190.10a are also required annually by Section 4.6.2.1.8.

Contributions to the dose due to liquid and gaseous effluent are calculated as described by the equations for:

a) total body and maximally exposed organ doses via shoreline exposure and eating fish and shellfish caught in the discharge canal at the Route 9 bridge as in Section 4.2.2.

b) total body and maximally exposed organ doses due to gaseous effluent other than noble gases as in Section 4.3.2.2.[*]

c) total body and organ doses due to gaseous tritium effluent as in Section 4.4.4.

d) total body and skin doses due to noble gases are gaseous effluent as described in Section 4.4.3.

e) total body irradiation from the Station, including outside storage tanks as described in Section 4.4.2.

[*] Based on I-131, I-133, and radionuclides in particulate form having half lives greater than eight days.

The receptor of the dose is a member of the public selected on the basis of the combination of applicable pathways of exposure to gaseous effluent identified in the annual land use survey and the minimum annual atmospheric dispersion from the OCNGS to his residence. Alternatively, the dose may be calculated to correspond with residence or exposure at an acceptable location where calculated exposures are unlikely to underestimate those experienced by the most exposed member of the public. That is, conditions assumed in the dose assessment may be realistic or may be more conservative than actual conditions. For the purpose of assessing compliance with Sections 4.6.1.1.8 or 4.6.2.1.8, the exposure of the most exposed member of the public is assumed to occur at the locations described in Attachment 2000-ADM-4532.04-6.

When assessing compliance with 40 CFR Part 190.10a, aqueous and atmospheric dispersion of radioactive material are based on annually-averaged conditions. Either data from the reference year or from the most recent calendar year are used to estimate dilution, dispersion, deposition, and elevated plume gamma exposure.

Environmental Measurements: When assessing compliance with 40 CFR 190.10a, Radiological Environmental Monitoring Program results may be used to indicate actual radioactivity levels in the environment attributable to the Station as an alternative to calculating the concentrations from radioactive effluent measurements. The measured environmental activity levels may thus be used to supplement the evaluation of doses to a member of the public.

4.4.2 Irradiation: The dose to a member of the public due to irradiation (external gamma radiation exposure) from the OCNGS and Station gaseous effluents will be estimated with the aid of environmental TLD, PIC, or similar environmental dosimetry. This will be done by examining the annual dosimetry data for a statistical difference between measurements near the Station and background measurements.

4.4.3 Noble Gas Dose Calculation: As an alternative to environmental dosimetry, the gamma dose to a member of the public by irradiation by noble gases in gaseous effluent from the Station may be calculated. In that event, the total body gamma dose is calculated with the equation:

$$D\gamma = SF \sum_i Q_{si} \quad P\gamma S_i + SF \sum_i Q_{vi} \; (X/Q)_v \, P\gamma V_i$$

where:

$D\gamma$ = noble gas gamma dose to total body (mrem).

SF = 0.7 is a shielding factor to account for the shielding by housing

Q_{si} = amount of radionuclide i released via the Stack (μCi).

Q_{vi} = amount of radionuclide i released via vent, v (μCi).

$P\gamma S_i$ = factor converting unit release of radionuclide i via the Stack to total body dose at ground level from the overhead plume (mrem/μCi).

$P\gamma V_i$ = factor converting time integrated, ground-level concentration of noble gas nuclide, i to total body dose (mRem/μCi per sec/m^3)

$(X/Q)_v$ = relative atmospheric dispersion from a vent release (sec/m^3).

Values of $P\gamma S_i$ and $P\gamma V_i$ which are used to compute the dose to the most exposed member of the public appear in Attachment 2000-ADM-4532.04-7. Similarly, the skin dose to a person due to noble gas in gaseous effluent may be calculated with the equation:

$$D\beta = \sum_i Q_{si}(X/Q)_s S\beta_i + \sum_i Q_{vi}(X/Q)_v S\beta_i + SF \sum_i 1.11 A\gamma S_i Q_{si} + SF \sum_i 1.11 A\gamma V_i (X/Q)_v Q_{vi}$$

where:

$D\beta$ = skin dose due to beta and gamma irradiation from noble gas in air (mrem).

X/Q_s = relative atmospheric dispersion from a Stack release (sec/m^3).

$S\beta_i$ = factor converting time-integrated ground level concentration of noble gas radionuclide i to skin dose from beta irradiation [mrem/(μCi-sec)/m^3].

Values of $S\beta_i$ for noble gases are included in Attachment 2000-ADM-4532.04-7.

1.11 = Ratio of tissue dose equivalent to air dose in a radioactive field (mRem/mRad).

SF = 0.7 a shielding factor to account for shielding by housing.

$A\gamma S_i$ = factor converting an elevated unit noble gas radionuclide i release to air dose at ground-level at a designated location from gamma radiation it emits (mrad/μCi).

$A\gamma V_i$ = factor for converting time integrated, semi-infinite concentration of noble gas radionuclide i to air dose from its gamma radiation [mRad/(μCi-sec)/m^3]

Noble gas plume gamma radiation-to-ground level air dose factors, $A\gamma S_i$, are calculated in accordance with Reg. Guide 1.109, Rev. 1, Appendix B models. The noble gas semi-infinite cloud gamma radiation-to-air dose factors, $A\gamma V_i$, and the noble gas semi-infinite cloud beta radiation-to-skin dose factors, $S\beta_i$, are derived from Reg. Guide 1.109, Rev. 1, Table B-1. Values of $A\gamma S_i$, $A\gamma V_i$, and $S\beta_i$ are used to assess compliance with 40CFR190.10a and are tabulated in

Attachment 2000-ADM-4532.04-7.

4.4.4 <u>Tritium Dose Calculation</u>: The total body and organ doses due to tritium in gaseous effluent are calculated with the equation:

$$DOSE = \sum_{p} \sum_{e} Q_e (X/Q)_e \, R_p$$

where:

DOSF = dose due to tritium released in gaseous effluent (mrem).

Q_e = amount of tritium released via release point e (μCi).

$(X/Q)_e$ = relative atmospheric dispersion from release point e to receptor at ground level (sec/m^3).

R_p = atmospheric concentration to dose factor for tritium via environmenta'. pathway, p (mrem/sec per $\mu Ci/m^3$).

Attachment 2000-ADM-4532.04-6

Locations Associated with Maximum Exposure of a Member of the Public*

Effluent	Location	
	Distance (meters)	Direction (to)
Liquid	U.S. Route 9 Bridge At Discharge Canal	
Airborne Iodine and Particulates	966	SE
Tritium	966	SE
Noble Gases	966	SE
Irradiation by OCNGS	Site Boundary	ALL

Note: the nearby resident experiencing the maximum exposure to airborne effluent and to gamma radiation directly from the Station is located 966 meters SE of the OCNGS. The most exposed member of the public is assumed to be exposed by irradiation from the OCNGS, by inhaling airborne effluent, by irradiation by the airbor.,e effluent, by irradiation by the airborne plume of the noble gas, by radionuclides deposited onto the ground, by irradiation by shoreline deposits, and by eating fish and shellfish caught in the discharge canal.

* The age group of the most exposed member of the public is based on USNRC Reg. Guide 1.109, Revision 1.

Attachment 2000-ADM-4532.04-7

Critical Receptor Noble Gas Dose Conversion Factors*

Radionuclide	$P_\gamma Si$** mRem/uCi	$P_\gamma Vi$*** mRem-m3/uCi-sec	$A_\gamma Vi$*** mRad-m3/uCi-sec	$A_\gamma Si$** mRad/uCi	SBi*** mRem-m3/uCi-sec
Kr83m	3.76E-17	2.40E-09	6.13E-07	9.66E-15	*****
Kr85m	1.68E-12	3.71E-05	3.90E-05	1.75E-12	4.63E-05
Kr85	2.60E-14	5.11E-07	5.46E-07	2.75E-14	4.25E-05
Kr87	8.37E-12	1.88E-04	1.96E-04	8.81E-12	3.09E-04
Kr88	2.08E-11	4.67E-04	4.83E-04	2.18E-11	7.52E-05
Kr89	1.83E-11	5.27E-04	5.49E-04	1.93E-11	3.21E-04
Kr90	*	4.95E-04	5.17E-04	*	2.31E-04
Xe131m	3.99E-13	2.90E-06	4.95E-06	4.44E-13	1.51E-05
Xe133m	3.10E-13	7.97E-06	1.04E-05	3.58E-13	3.16E-05
Xe133	3.11E-13	9.33E-06	1.12E-05	3.42E-13	9.71E-06
Xe135m	4.71E-12	9.90E-05	1.07E-04	5.01E-12	2.26E-05
Xe135	2.73E-12	5.75E-05	6.10E-05	2.87E-12	5.90E-05
Xe137	1.65E-12	4.51E-05	4.79E-05	1.75E-12	3.87E-04
Xe138	1.33E-11	2.80E-04	2.92E-04	1.40E-11	1.31E-04
Xe139	*	*****	*****	1.61E-12	*****
Ar41	1.58E-11	2.81E-04	2.95E-04	1.66E-11	8.54E-05

* All of these dose factors apply out-of-doors.

** Based on reference meteorology at 966 meters SE of effluent stack.

*** Derived from USNRC Regulatory Guide 1.109, Revision 1, Table B-1.

Attachment 2000-ADM-4532.04-8

The Individual Maximum Hypothetical Exposed Member of the Public at the OCNGS

Effluent	Location	
	Distance (meters)	Direction (to)
Liquid	U.S. Route 9 Bridge At Discharge Canal	
Noble Gas Y-Air Dose	Site Boundary	ALL
Noble Gas B-Air Dose	966	SE
Iodine and Particulates	966	SE

Note: The Beta Air Dose, Noble Gas and Iodine and Particulate receptor is dependent on the highest annual-average X/Q values as described in this ODCM. For liquids, the maximum pathway is via consumption of fish and shellfish from the OCNGS Discharge Canal and direct shoreline exposure. The nearest individual resident is 966 meters to the SE of the OCNGS for assessing compliance with Sections 4.6.1.1.5, 4.6.1.1.6 and 4.6.1.1.7.

4.5 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP)

4.5.1 Radiological Environmental Surveillance - the environmental surveillance of radiation and radioactive effluent from the OCNGS. The objective of the REMP is the measurement and assessment of radiation and radioactive material in the environment which was discharged from the OCNGS.

4.5.1.1 Environmental samples shall be collected and analyzed according to Attachment 2000-ADM-4532.04-9. Analytical techniques shall be used such that the detection capabilities indicated in Attachment 2000-ADM-4532.04-10 are achieved. Locations from which radiological environmental samples are intended to be collected shall be identified in Section 4.5.2, Radiological Environmental Implementation.

4.5.1.2 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.

4.5.1.3 In the event an environmental sample required by Attachment 2000-ADM-4532.04-9 is not collected and analyzed in accordance with the provisions of the attachment, the deviation shall be documented in the Annual Radiological Environmental Report.

4.5.1.4 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.

4.5.1.5 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 Section 4.5.2 in accordance with Specification 6.19 of the OCNGS Technical Specifications.

4.5.1.6 If a confirmed[1] measured radionuclide concentration in an

---

[1] A confirmatory re-analysis of the original, 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 Attachment 2000-ADM-4532.04-11 are detected and are due to plant effluent, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objectives of 10CFR50, Appendix I. This report may include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

Attachment 2000-ADM-4532.04-9

Radiological Environmental Monitoring Program

Medium Sampled	Minimum Number of Sampling Locations	Sampling and Collection Frequency	Analysis Type
Airborne (Particulate)	4 Indicator / 1 Background	Biweekly Quarterly	Gross Beta Gamma Isotopic
Airborne (Iodine)	4 Indicator / 1 Background	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

Attachment 2000-ADM-4532.04-10

Detection Capabilities For Environmental Sample Analysis

Lower Limit of Detection*

Isotope	Water (pCi/liter)	Air (pCi/m3)	Food Products (pCi/kg WET)	Sediments/Soils (pCi/kg DRY)	Aquatic Biota (pCi/kg WET)
H-3	2000**	-	-	-	-
	3000***	-	-	-	-
Mn-54	15		-	-	130
Fe-59	30	-	-	-	260
Co-58	15	-	-	-	130
Co-60	15	-	-	-	130
Zn-65	30	-	-	-	260
Zr-95	30	-	-	-	-
Nb-95	15	-	-	-	-
Cs-134	15	5E-02	60	150	130
Cs-137	18	6E-02	80	180	150
La-140	15	-	-	-	-
Ba-140	60	-	-	-	-
I-131	1000***	7E-02	60	-	-
Gross Beta	4	1E-02	-	-	-

Attachment 2000-ADM-4532.04-10 Notations

* 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 five 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:

$$LLD = \frac{4.66\ S_b}{E*V*2.22E6\gamma*\exp(-\lambda\Delta t)}$$

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)

2.22E6 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

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

V is the sample size (mass or volume).

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

** For a sample of drinking water. See Section 4.5.1.4.5.

*** For a sample of water not used as a source of drinking water. See Section 4.5.1.4.5.

environmental sampling medium averaged over any quarter sampling period exceeds the reporting level given in Attachment 2000-ADM-4532.04-11, 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 effluent (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 Attachment 2000-ADM-4532.04-11 are detected in the medium, the reporting level shall have been exceeded if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots\ldots\ldots \leq 1$$

4.5.1.2 Interlaboratory Comparison Program Specifications

4.5.1.2.1 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.

4.5.1.2.2 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.

4.5.1.2.3 The provisions of 3.0.A, 3.0.B and 6.9.2 of the OCNGS Technical Specifications are not applicable.

4.5.1.3 Land-Use Survey Specification

4.5.1.3.1 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)[2], 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 OCNGS. If it is learned from this survey that the milk animals or gardens are present at a location which yields a calculated

---

[2] 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.

Attachment 2000-ADM-4532.04-11

Reporting Levels (RL) For Nonroutine Operating Reports

Analysis	Water (pCi/liter)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/kg WET)	Milk (pCi/l)	Broad Leaf Vegetation (pCi/kg WET)
H-3	2E+04**	-	-	-	-
Mn-54	1E+03	-	3E+04	-	-
Fe-59	4E+02	-	1E+04	-	-
Co-58	1E+03	-	3E+04	-	-
Co-60	3E+02	-	1E+04	-	-
Zn-65	3E+02	-	2E+04	-	-
Zr-Nb-95	4E+02	-	-	-	-
I-131	2	0.9	-	3	1E+02
Cs-134	30	10	1E+03	60	1E+03
Cs-137	50	20	2E+03	70	2E+03
Ba-La-140	2E+02	-	-	3E+02	-

* Well Water Only - See Section 4.5.1.4.5

thyroid dose at least 20 percent greater than those previously sampled, or if the survey results in changes in the location used in Section 4.5.2, 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.

4.5.1.4 Basis

4.5.1.4.1 It should be noted that in addition to the sampling and analysis required herein, 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.

4.5.1.4.2 The detection capability stated in Attachment 2000-ADM-4532.04-10 and the reporting level stated in Attachment 2000-ADM-4532.04-11 for each radionuclide in environmental samples are derived from the USNRC Branch Technical Position on Radiological Environmental Monitoring, Revision 1, Tables 2 and 4, November, 1979.

4.5.1.4.3 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.

4.5.1.4.4 Recent on site research conducted by GPUN (final report, March, 1984) has demonstrated that the groundwater pathway is not a potential pathway to man from the OCNGS. This recent site research also installed new sampling wells on site.

4.5.1.4.5 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.

4.5.1.4.6 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 these specifications were developed, limited quantities of locally grown vegetables were available for sampling.

4.5.1.4.7 The source for Attachment 2000-ADM-4532.04-10 and Attachment 2000-ADM-4532.04-11 is the USNRC Branch Technical Position on Radiological Environmental Monitoring, Revision 1, dated November 1979.

4.5.2 Radiological Environmental Program Description and Sampling Locations

4.5.2.1 Air Particulate/Air Iodine: Currently, GPUN intends to maintain air particulate and air iodine sampling stations as follows: Azimuths and distances are estimated from present conditions and may change slightly.

a) One indicator station in the SSE sector at approximately 1.5 to 2.0 miles from the OCNGS, generally along azimuth of 157.
b) One indicator station in the SE sector within 7 miles of the OCNGS, generally along the azimuth of 126.
c) One indicator station in the ESE sector within 7 miles of the OCNGS, generally along the azimuth of 110.
d) One indicator station in the N or NNE sector within 3 miles of the OCNGS, generally along the azimuth of 22.5.
e) One background station in the vicinity of Cookstown, N.J. (NW sector, approximately 24 miles from the OCNGS, along the azimuth of 306.

4.5.2.2 Thermoluminescent Dosimetry (TLD): Currently, GPUN intends to deploy 32 thermoluminescent dosimeters as follows:

a) Sixteen stations within 1/4 mile of the site boundary, one per sector, or as close as reasonable highway access permits.
b) Eleven stations, one per sector, within 5 miles of the site boundary (5 sectors are over water at 5 miles).
c) Three areas of special interest at this time, Waretown, N.J. (NNW, N, or NNE) vicinity, and the Garden State Parkway vicinity are being considered.

4.5.2.3 Groundwater: See Section 4.5.1.4.4. The indicator station will be located .1 miles to the southwest on the OCNGS site. The background station will be in a suitably representative but relatively unaffected area around the OCNGS.

4.5.2.4 Surface Water: See Section 4.5.1.4.5. The indicator station for surface water will be located in the OCNGS discharge canal, east of the U.S. Route 9 highway bridge generally along the azimuth of 113. The background station will be in a suitably representative but relatively unaffected section of the Barnegat Bay system.

4.5.2.5 Aquatic Sediment: The sediment sampling locations will be the same as for the surface water (see Attachment 2000-ADM-4532.04-9 for quantity of samples as well as collection frequency). The background station will be in a suitably representative but relatively unaffected section of the Barnegat Bay system.

4.5.2.6 Fish: The sampling locations for fish may be affected by adverse weather, availability of fish, and other variables. Specific locations will be reported in the Annual Radiological Environmental Monitoring report. Where possible and practical, established stations will be regularly sampled. Also, it should be noted that fish samples will consist of the edible portions of available fish that are consumed by man. Seasonal availability and naturally occurring fishery population fluctuations will determine the precise species that are sampled.

4.5.2.7 Shellfish: The sampling locations for shellfish will be the same locations as for the surface water (see Attachment 2000-ADM-4532.04-9 for quantity of samples as well as collection frequency). The background station will be in a suitably representative but relatively unaffected section of the Barnegat Bay system. Samples will consist of the edible portions of approximately one to three dozen hard clams (as limited by seasonal availability or naturally occurring population fluctuations).

4.6 RADIOACTIVE EFFLUENTS MONITORING PROGRAM

4.6.1 Radioactive Effluent Requirements

This section applies to the radioactive effluent of the facility. It's objective is 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 10CFR part 20.106 and 40CFR Part 190.10(a).

4.6.1.1 Sections

4.6.1.1.1 Liquid Radwaste Treatment

This section applies to liquid radwaste batches before discharge as aqueous effluent.

A. 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 batch exceeds 0.001 µCi/ml.

B. 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 Section 4.8.3, a Special Report that includes the following information:

(1) Identification of any inoperable equipment or subsystems, and the reason for the inoperability.

(2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and a

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

4.6.1.1.2 Condenser Offgas Treatment

This applies 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.

A. 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.

B. If gaseous effluent is released without treatment for more than 30 consecutive days and either Sections 4.6.1.1.5 or 4.6.1.1.6 exceeded, in lieu of any other report, submit a Special Report pursuant to Section 4.8.3 to the NRC within 30 days from the end of the quarter during which the release occurred which includes the following information:

(1) Identification of the inoperable equipment or subsystem and the reason for inoperability; and

(2) Action(s) taken to restore the inoperable equipment to OPERABLE status and to prevent a recurrence.

4.6.1.1.3 Radioactivity Concentration in Liquid Effluent

A. The concentration of radioactive material, other than noble gases, in liquid effluent in the discharge canal at the U.S. Route 9 bridge (see Attachment 2000-ADM-4532.04-1) shall not exceed the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2.

B. The concentration of noble gases dissolved or entrained in liquid effluent in the discharge canal at the U.S. Route 9 bridge shall not exceed 2.0E-4 μCi/ml.

C. In the event the concentration of radioactive material in liquid effluent released into the Offsite area beyond the U.S. Route 9 bridge exceeds either the concentration limit in 4.6.1.1.3.A or 4.6.1.1.3.B, reduce the release rate without delay to bring the concentration below the limit.

4.6.1.1.4 Limit on Dose Due to Liquid Effluent

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

1.5 mRem to the Total Body during any calendar qtr.

5.0 mRem to any body organ during any calendar qtr.

3.0 mRem to the Total Body during any calendar yr.

When the calculated dose from the release of radioactive materials in liquid effluent 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 Section 4.8.3, 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.

4.6.1.1.5 Dose Rate Due To Gaseous Effluent

A. The dose equivalent rate outside the EXCLUSION AREA (see Attachment 2000-ADM-4532.04-1) 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.

B. The dose equivalent rate outside the EXCLUSION AREA due to tritium (H-3), I-131, I-133, and to radioactive material in particulate form 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 no more than 31 days.

C. In the event the dose equivalent rate exceeds any of the limits in 4.6.1.1.5.A or 4.6.1.1.5.B, 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.

4.6.1.1.6 Air Dose Due to Noble Gas in Gaseous Effluent

A. The air dose outside of the EXCLUSION AREA (see Attachment 2000-ADM-4532.04-1) 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.

B. If the calculated air dose due to noble gas released in gaseous effluent exceeds any limit in Section 4.6.1.1.6.A, 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 Section 4.8.3, 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.

4.6.1.1.7 Dose Due to Radioiodine and Particulates in Gaseous Effluent

A. The dose due to a MEMBER OF THE PUBLIC from I-131, I-133, and from radioiodines in particulate from having half-lives of 8 days or more in gaseous effluent, outside of the EXCLUSION AREA (see Attachment 2000-ADM-4532.04-1) shall not exceed 7.5 mRem to any body organ per calendar quarter or 15 mRem to any body organ per calendar year.

B. 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 Section 4.6.1.1.7.A, 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 Section 4.8.3, 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.

4.6.1.1.8 Annual Total Dose Due to Radioactive Effluent

A. The annual dose to a MEMBER OF THE PUBLIC due to radioactive material in effluent from the OCNGS outside of the EXCLUSION AREA (see Attachment 2000-ADM-4532.04-1) shall not exceed 75 mRem to his/her thyroid or 25 mRem to his/her total body or to any other organ.

B. In the event the calculated dose due to radioactive material released in liquid or gaseous effluent exceeds twice the limits of Section 4.6.1.1.4, 4.6.1.1.6, or 4.6.1.1.7, perform an assessment of compliance with Section 4.6.1.1.8 in accordance with methodology within this ODCM.

C. In the event an assessment shows that Section 4.6.1.1.8 to have been exceeded, prepare and submit a Special Report to the Commission within 30 days, pursuant to Section 4.8.3 and in lieu of any other report. The report shall include information specified in 10CFR20.405(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 40CFR Part 190. In that event, the request is timely and a variance is granted until NRC action on the request is complete.

4.6.1.2 Basis

4.6.1.2.1 This specification implements the requirements of 10CFR50.36a related to operation of radioactive waste treatment equipment to keep radioactive material in effluent to unrestricted areas as low as reasonably achievable. Radioactive liquid wastes generated at the OCNGS 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 μCi/ml, it is not cost-beneficial to treat a batch of aqueous waste for the purpose of reducing potential radiation exposure offsite. Hence Section 4.6.1.1.1 implements 10CFR Part 50 Appendix I provisions for cost-beneficial treatment 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.

4.6.1.2.2 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 10CFR Part 50 Appendix A Criterion 60.

The appropriate portions of this system provide reasonable assurance that the releases of radioactive materials in gaseous effluent will be kept "as low as 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.

4.6.1.2.3 The purpose of Section 4.6.1.1.3 is to require that concentrations of radioactive material in aqueous effluent 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.

4.6.1.2.4 The purpose of Section 4.6.1.1.4 is to require compliance with 10CFR Part 50 Appendix I, Section IV.A to assure that radioactive material in liquid effluent is kept as low as reasonably achievable and to permit operating flexibility under unusual operating conditions.

4.6.1.2.5 The purpose of Section 4.6.1.1.5 is to require that the concentrations of radioactive material in airborne effluents to OFFSITE areas comply with 10CFR 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 to those Members of the Public to be less than the equivalent annual limit on radiation exposure to a Member of the Public incurred OFFSITE.

4.6.1.2.6 The purpose of Section 4.6.1.1.6 is to require compliance with 10CFR Part 50 Appendix I, Section IV.A and to provide operating flexibility under unusual conditions as permitted in 10CFR 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 incurred Offsite.

4.6.1.2.7 See Basis 4.6.1.2.6.

4.6.1.2.8 Sections 4.6.1.1.8 and 4.6.2.1.8 implement the provisions of 40 CFR 190.10a as incorporated into 10CFR Part 20.405(c). It is unlikely that the dose to any Member of the Public will exceed the limits of 40CFR Part 190.102 as long as the exposure remains within the limits of Sections 4.6.1.1.4, 4.6.1.1.6 and 4.6.1.1.7. 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.

4.6.2 <u>Radiological Effluent Monitoring Requirements</u>

These requirements apply to monitoring of gaseous and liquid radioactive effluent of the Station during release of effluent 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. The objective is to measure radioactive effluent adequately to verify that radioactive effluent are as low as is reasonably achievable and within the limit of 10CFR Part 20.106.

4.6.2.1 <u>Sections</u>

4.6.2.1.1 See Sections 4.6.2.1.3.

4.6.2.1.2 <u>Main Condenser Offgas Treatment</u>

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.

4.6.2.1.3 Radioactivity Concentration in Liquid Effluent

A. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program in Attachment 2000-ADM-4532.04-14.

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 1E-7 μCi/ml in the discharge canal at the U.S. Route 9 bridge is applied.

B. The alarm or trip setpoint of each radioactivity monitoring channel in Attachment 2000-ADM-4532.04-12 shall be determined on the basis of sampling and analyses results obtained according to Attachment 2000-ADM-4532.04-14 and setpoint method in this document and set to alarm or trip before exceeding the limits of Section 4.6.1.1.7.

4.6.2.1.4 Dose Due to Liquid Effluent

An assessment shall be performed in accordance with this procedure at least once a month to determine compliance with Section 4.6.1.1.4.

4.6.2.1.5 Dose Rate Due To Gaseous Effluent

Radioactive noble gaseous effluent shall be monitored in accordance with Section 4.7.1.1.2. Radioactive noble gas monitors named in Attachment 2000-ADM-4532.04-13 shall be set to cause automatic alarm when the monitor setpoint, determined as specified in this procedure, is exceeded.

4.6.2.1.6 Not Used.

4.6.2.1.7 Dose Due to Radioiodine and Particulates in Gaseous Effluent

An assessment shall be performed in accordance with this procedure 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 Section 4.6.1.1.7.

4.6.2.1.8 Annual Total Dose Due to Radioactive Effluent

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

4.6.2.2 Basis

4.6.2.2.1 Not Used.

4.6.2.2.2 Not Used.

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

4.6.2.2.4 Not Used.

4.6.2.2.5 Not Used.

4.6.2.2.6 Not Used.

4.6.2.2.7 Not Used.

4.6.2.2.8 Not Used.

4.7 RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION

4.7.1 Radioactive Effluent Monitoring Instrumentation Requirements

This section applies to instrumentation whose function is to monitor aqueous and airborne radioactive effluent from the Station. Its objective is to assure that instrumentation to monitor radioactive effluent is OPERABLE when effluent is discharged or that means of measuring effluent is provided.

4.7.1.1 Sections

4.7.1.1.1 Liquid Effluent Instrumentation

A. The radioactive effluent monitoring channels listed in Attachment 2000-ADM-4532.04-12 shall be OPERABLE with their alarm/trip setpoints set to initiate alarm/trip in the event the limit of Specification 4.6.1.1.4.A is exceeded.

B. The alarm or trip setpoint of these channels shall be determined and set in accordance with the method described in Section 4.1.

C. 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 effluent 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 functions(s).

D. When less than the minimum number of radioactive liquid effluent monitoring instrumentation channels are OPERABLE, take the ACTION shown in Attachment 2000-ADM-4532.04-12. Make every reasonable effort to restore the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

4.7.1.1.2 Gaseous Effluent Instrumentation

A. Each radioactive effluent noble gas monitoring channel listed in Attachment 2000-ADM-4532.04-13 shall be OPERABLE with its alarm setpoint set to cause automatic alarm in the event a limit of Section 4.6.1.1.5.A is exceeded.

B. The alarm or trip setpoint of these channels shall be determined and set in accordance with the method described in Section 4.1.

C. When a radioactive effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required by Section 4.7.1.1.2.A without delay suspend the release of radioactive gaseous effluent monitored by the affected channel or declare the channel inoperable or change the setpoint so it is acceptably conservative.

D. When less than the minimum number of radioactive gaseous monitoring instrumentation channels are OPERABLE, take the ACTION shown in Attachment 2000-ADM-4532.04-13. Make every reasonable effort to restore the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

4.7.1.2 Basis

4.7.1.2.1 The radioactive liquid effluent instrumentation is provided to monitor and control, if applicable, the releases of radioactive materials in liquid effluent during actual or potential releases of liquid effluent. The use of this instrumentation is consistent with the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. Radioactivity monitors on the liquid radwaste effluent line and in the Turbine Building Sump 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 Section 4.1 to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20.106.

4.7.1.2.2 The radioactive gaseous effluent instrumentation in Attachment 2000-ADM-4532.04-13 is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluent during releases of gaseous effluent. The alarm/trip setpoint for each of the noble gas monitors is calculated and adjusted in accordance with methodology and parameters in Section 4.1 to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20.106. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. 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 Stack and the Turbine Building exhaust ventilation effluent air are monitored by a radioactive gaseous effluent monitoring system. It can sample effluent for radioactive particulates, iodine, and noble gases. It can measure the gross concentration of radioactive noble gases. It can measure the gross concentration of radioactive noble gases. A grab sample of the effluent will be taken at least once per month and analyzed for the principal noble gas radionuclides (Reference: Attachment 2000-ADM-4532.04-15).

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 would also cause an alarm in the reactor control room. Although flow data may be collected by a computer the sample flow and the sampled steam flow can also be observed at a display located near the monitoring instrument (in which case the channel continues to serve its essential function and remains OPERABLE). If the noble gas activity concentration display and the associated alarm become inoperable in the reactor control room, then OCNGS will perform the appropriate action according to Attachment 2000-ADM-4532.04-13.

The gross gamma activity concentration of noble gas in Turbine Building exhaust ventilation effluent is displayed locally in the Turbine RAGEMS Building. Noble gas activity concentration, sample flow and sampled stream flow may be collected by a computer and may also be observed on displays near the monitoring instruments (in which case the channels continue to serve their essential functions and remain OPERABLE). If the noble gas activity concentration display and the associated alarm become inoperable in the Turbine RAGEMS Building, then OCNGS will perform the appropriate action according to Attachment 2000-ADM-4532.04-13.

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 Attachment 2000-ADM-4532,04-13. Action 124 that is applied during purging is not imposed during drywell venting.

4.7.2 Radioactive Effluent Monitoring Instrumentation Surveillance

This section states surveillance requirements for **OPERABILITY** of radioactive effluent monitoring instrumentation. Its objective is to demonstrate the **OPERABILITY** of radioactive effluent monitoring instrumentation.

4.7.2.1 Section

4.7.2.1.1 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 Attachment 2000-ADM-4532.04-16.

4.7.2.1.2 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 Attachment 2000-ADM-4532.04-17.

Attachment 2000-ADM-4532.04-12

Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	Minimum Channels Operable[a]	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 Number 1-5	1	b	114
2. FLOW MEASUREMENT DEVICES			
a. Liquid Radwaste Effluent Line	1	b	113

ATTACHMENT 2000-ADM-4532.04-12 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 Section 4.6.2.1.3.A, 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 1E-6 uCi/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 Section 4.6.2.1.3.A.

2. Qualified personnel determine and independently verify the acceptable release rate.

Attachment 2000-ADM-4532.04-13

Radioactive Gaseous Effluent Monitoring Instrumentation

Instrument	Minimum Channels Operable[a]	Essential Function	Applicability	Action
1. Not Used	*	*	*	*
2. Stack Monitoring System				
a. Radioactive Noble Gas Monitor (low range)	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 (low range)	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 tilation 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

ATTACHMENT 2000-ADM-4532.04-13 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.

e. Monitor/sampler or an alternate shall be OPERABLE to monitor/sample Stack effluent whenever the drywell is being purged.

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 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.

Attachment 2000-ADM-4532.04-14

Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)
A. Batch Waste Release Tanks	P Each Batch[b]	P[c] Each Batch	Principal Gamma Emitters	1E-06
			I-131	1E-06
	P One Batch/M[b]	M	Dissolved and Entrained Gases (Gamma Emitters)	1E-05
	P Each Batch[b]	M Composite[d]	H-3	1E-05
			Gross Alpha	1E-07
	P Each Batch[b]	Q C[illegible]	Sr-89, Sr-90	5E-08
			Fe-55	1E-06
B. Reactor Building Service Water Effluent and Turbine Bldg. Sump No. 1-5	W Grab Sample[e]	W	Principal Gamma Emitters	1E-06
			I-131	1E-06
	(note f)	M Composite[g]	H-3	1E-05
			Gross Alpha	1E-07
	(note f)	Q Composite[g]	Sr-89, Sr-90	5E-08
			Fe-55	1E-06

Legend:

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

Attachment 2000-ADM-4532.04-14 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 analyses program.

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

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

Where:

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

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

E is the counting efficiency.

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

2.22E-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 the 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 t for the radionuclides Mn-54, Fe-59, Co-58, Co-60, Zn-65, Ce-141, Cs-134, Cs-137; and LLD of 1E-5 µCi/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.

Attachment 2000-ADM-4532.04-14 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 sample. The background count rate of a GeLi 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, 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 be identified and reported. The LLD for Mo-99 and Ce-144 is 1E5 µCi/ml. Nuclides which are below the LLD for the analysis should be not 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 be analyzed for principal gamma emitters afterwards.

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 1E-6µCi/ml of I-131 and principal gamma emitters or in the event the effluent radioactivity monitor indicates more than 1E-6 µCi/ml radioactivity in the effluent, as applicable, sample Reactor Building Service Water effluent daily or sample Turbine Building Sump 1-5 each discharge until analysis confirms the activity concentration in the effluent does not exceed 1E-6 µCi/ml.

g. A composite sample is produced combining grab samples, each having a defined volume, collected routinely from the sump or stream being sampled.

Attachment 2000-ADM-4532.04-15

Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)
Stack	Q Grab Sample[f]	Q	H-3	1E-06
Stack: Turbine Building Exhaust Vents; Offgas Building Vent	M Grab Sample c,d,f	M	Principal Gamma Emitters[b] (Noble Gases)	1E-04
	Continuous[f]	W Charcoal Sample	I-131	1E-12
			I-133	1E-10
	Continuous[f]	W Particulate Sample	Principal Gamma Emitters[b] (particulates)	1E-11
	Continuous[f]	M[e] Composite Particulate Sample	Gross Alpha	1E-11
	Continuous	Q[e] Composite Particulate Sample	Sr-89,Sr-90	1E-11
	Continuous	Noble Gas Monitor	Noble Gases Gamma Radioactivity	1E-06

Legend:

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

Attachment 2000-ADM-4532.04-15 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 analyses program.

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

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

Where:

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

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

E is the counting efficiency.

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

2.22E-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 the 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 t for the radionuclides Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, and Cd-141; an LLD of 1E-5 µCiml should typically be achieved for Mo-99 and Ce-144. Occasionally background fluctuations, interfering radionuclides, or other uncontrollable circumstances may render these LLDs 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 GeLi 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.

Attachment 2000-ADM-4532.04-15 NOTATIONS
(continued)

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 Annual Radioactive Material Release Report. The LLD for Mo-99 and Ce-144 is 1E-10 μCi/ml whereas the LLD for other principal gamma emitting particulates is 1E-11 μCi/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 a 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.

Attachment 2000-ADM-4532.04-16

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[f]	R[e]	Q[c]	b
b. Reactor Building Service Water System Effluent Line	D	M	R[e]	Q[d]	b
c. Turbine Building Sump No. 1-5	D	M	R[e]	Q[d]	b
2. Flow Rate Measurement Devices					
a. Liquid Radwaste Effluent Line	D[g]	N/A	R	Q	b

Legend:

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

Attachment 2000-ADM-4532.04-16 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 operable.

b. During releases via this pathway.

c. 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.

d. 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.

e. The **CHANNEL CALIBRATION** shall be performed according to established station calibration procedures.

f. On any day during which a release is made, a **SOURCE CHECK** shall be made at least once per day, before the first release.

g. **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.

h. The **CHANNEL FUNCTIONAL TEST** shall also demonstrate that Control Room alarm annunciator occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Instrument indicates a downscale failure.

Attachment 2000-ADM-4532.04-17

Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test	Surveillance Required [a]
1. Not Used	*	*	*	*	*
2. Main Stack Monitoring System					
a. Radioactive Noble Gas Monitor (Low Range)	D	M	1/24 [f]	Q [h]	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	1/24	Q	b
e. Sample Flow Measuring Device	D	N/A	R	Q	b
3. Turbine Building Ventilation Monitoring System					
a. Radioactive Noble Gas Monitor (Low Range)	D	M	1/24 [f]	Q [i]	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	1/24	Q	b
e. Sample Flow Measuring Device	D	N/A	N/A	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. Sample Flow Measuring Device	D	N/A	R	N/A	b

Legend:

S = once per 12 hours
M = once per 31 days
R = once per 18 months
P = completed before each release
D = once per 24 hours
Q = once per 92 days
S/U = before each reactor startup
N/A = Not Applicable
W = once per 7 days
SA = once per 184 days
1/24 = once per 24 months

Attachment 2000-ADM-4532.04-17 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.

e. The **CHANNEL FUNCTIONAL TEST** shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

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.

h. 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 low count rate/monitor failure.
3. Switch cover alarm shall be verified to alarm when the cover is opened; and clear when the cover is closed after the faceplate switches are verified in their correct positions.

i. 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 low count rate/monitor failure.

4.8 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of 10 CFR, the following identified reports shall be submitted to the Administrator of the NRC Region I office unless otherwise noted.

4.8.1 Annual Radioactive Effluent Release Report - A report of radioactive materials released from the Station during the preceding 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:

4.8.1.1 A summary by calendar quarter and by radionuclide of the quantities of radioactive liquid and gaseous effluent from the Station.

4.8.1.2 A summary of radioactive solid waste shipped from the Station including:

4.8.1.2.1 physical description of the waste
4.8.1.2.2 classification of the waste, per 10 CFR Part 61
4.8.1.2.3 solidification agent (if solidified)
4.8.1.2.4 total volume shipped
4.8.1.2.5 total quantity of radioactive material shipped (curies)
4.8.1.2.6 best knowledge of identity of principal radionuclides.

4.8.1.3 A description of any changes to the Process Control Plan (PCP) or ODCM.

4.8.1.4 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.

4.8.2 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 Section 4.5.1 for radiological environmental surveillance:

4.8.2.1 A summary description of the radiological environmental monitoring program.

4.8.2.2 A map and a table of distances and directions (compass azimuth) of locations of sampling stations from the reactor.

4.8.2.3 Results of analyses of samples and of radiation measurements. (In the event some results are not available, the reasons shall be explained int he report. In the event the missing results are obtained, they shall be reported to the NRC as soon as is reasonable.)

4.8.2.4 Deviation(s) from the environmental sampling schedule in Attachment 2000-ADM-4532.04-10.

4.8.2.5 Identification of environmental samples analyzed when instrumentation was not capable of meeting detection capability in Attachment 2000-ADM-4532.04-10.

4.8.2.6 A summary of the results of the land use survey.

4.8.2.7 A summary of results of licensee participation in an NRC approved inter-laboratory crosscheck program for environmental samples.

4.8.2.8 Results of dose evaluations to demonstrate compliance with 40 CFR Part 190.10a.

4.8.3 Unique Reporting Requirements

Special reports shall be submitted to the appropriate NRC 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) Dose due to radioactive liquid effluent exceeding Section 4.6.1.1.4.A

b) Air dose due to radioactive noble gas in gaseous effluent exceeding Section 4.6.1.1.6.A

c) Dose due to radioiodine and particulates exceeding Section 4.6.1.1.7.A

d) Annual total dose due to radioactive effluent exceeding Section 4.6.1.1.8.A

e) Records of results of analyses required by the Radiological Environmental Monitoring Program.

f) Liquid radwaste batch discharge exceeding Section 4.6.1.1.1.A

g) Main condenser offgas discharge without treatment per Section 4.6.1.1.2.A

4.8.4 Basis

4.8.4.1 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.

5.0 RESPONSIBILITIES

5.1 **RADIOLOGICAL CONTROLS DIRECTOR** (Oyster Creek) - Responsible for this procedure and changes to it, as per 1000-ADM-1291.01, "Oyster Creek Review and Approval Matrix (Exhibit 2)".

5.2 **PLANT CHEMISTRY** (Oyster Creek) - Responsible for:

5.2.1 Compliance with specifications regarding routine dose assessment.

5.2.2 Concurring with this procedure and any Divisional procedures which address interdivisional responsibilities.

5.3 **PLANT OPERATIONS** (Oyster Creek) - Responsible for compliance with specifications regarding instrumentation surveillance.

5.4 **PLANT ENGINEERING** (Oyster Creek) - Responsible for compliance with specifications regarding setpoint implementation.

5.5 **TECH FUNCTIONS** (Oyster Creek) - Responsible for compliance with specifications regarding setpoint determination.

5.6 **ENVIRONMENTAL CONTROLS** - (Oyster Creek) - Responsible for:

5.6.1 Radiological Environmental Monitoring Program, its maintenance and adherence to specifications contained herein.

5.6.2 Technical consultation and review of Environmental Control-related items in the document.

5.7 **RADIOLOGICAL ENGINEERING** (Oyster Creek) - Responsible for technical consultation and review of Radiological items contained within the document.

6.0 REFERENCE

6.1 Procedures

6.1.1 1000-ADM-1218.01, GPU Nuclear Policy, Plan and Procedure System

6.1.2 1000-ADM-1291.01, Safety Review Process

6.2 Oyster Creek Updated Final Safety Analysis Report, Volume 1

6.3 Oyster Creek Final Design Safety Analysis Report, Volume 1

6.4 Oyster Creek Technical Specification and Bases

6.5 Other References

6.5.1 Boegli, J.S., W.L. Britix, R. R. Bellamy, and R. L. Waterfield, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", NUREG-0133, October, 1978.

6.5.2 Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I, USNRC Regulatory Guide 1.109 (Rev 1), October, 1977.

6.5.3 Licensing Application, Amendment 13, Meteorological Radiological Evaluation for the Oyster Creek Nuclear Power Station Site.

6.5.4 Licensing Application, Amendment 11, Question IV-8.

6.5.5 Evaluation of the Oyster Creek Nuclear Generating Station to Demonstrate Conformance to the Design Objectives of 10CFR50, Appendix I, May, 1976, Table 3-10.

6.5.6 Meteorological Information and Diffusion Estimates t Conform with Appendix I Requirements: Oyster Creek, July, 1976, Table 1.3-11B.

6.5.7 Hydrological Information and Liquid Dilution Factors Determination to Conform with Appendix I Requirements: Oyster Creek, correspondence from T. Potter, Pickard, Lowe and Garrick, Inc. to Oyster Creek, July, 1976.

6.5.8 Carpenter, J. J. "Recirculation and Effluent Distribution for Oyster Creek Site", Pritchard-Carpenter Consultants, Baltimore, Maryland, 1964.

6.5.9 Nuclear Regulatory Commission, Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section and Relocation of the Procedural Details of RETS to the ODCM or PCP", January, 1989.

6.5.10 USNRC Regulatory Guide 4.8, Branch Technical Position, November, 1979.

6.5.11 Ground Water Monitoring System (Final Report), Woodward-Clyde Consultants, March, 1984.

6.5.12 Meteorology and Atomic Energy, Department of Energy, 1981.

7.0 EXHIBITS

- 2000-ADM-4532.04-1 Atmospheric Dispersion Data
- 2000-ADM-4532.04-2 Assumed Distance to OCNGS Site Boundary in Each Sector
- 2000-ADM-4532.04-3 OCNGS Dispersion Parameters
- 2000-ADM-4532.04-4 OCNGS Usage Factors for Individual Dose Assessment
- 2000-ADM-4532.04-5 Radionuclides in Liquid Effluent Used to Assess Compliance With Technical Specifications
- 2000-ADM-4532.04-6 Radionuclides in Gaseous Effluent Used to Assess Compliance With Compliance with Technical Specifications.

EXHIBIT 2000-ADM-4532.04-1

ATMOSPHERIC DISPERSION DATA

Centerline X/Q and D/Q values were obtained for a sector width of 22.5 degrees. On-site meteorological data for the period January 1, 1989 through December 31, 1990 were used. Data were collected and atmospheric and stability classes were defined on the basis of vertical temperature difference methodology in conformance with NRC Regulatory Guide 1.23.

METEROLOGICAL ANALYSES

Exhibit 2000-ADM-4532.04-2	Site Boundary from Reactor/Stack
Exhibit 2000-ADM-4532.04-3	Stack - Elevated Release - X/Q
Exhibit 2000-ADM-4532.04-3	Stack - Elevated Release - D/Q
Exhibit 2000-ADM-4532.04-3	Reactor Building Vent - Ground Release - X/Q
Exhibit 2000-ADM-4532.04-3	Reactor Building Vent - Ground Release - D/Q

EXHIBIT 2000-ADM-4532.04-2

Assumed Distance to OCNGS Site Boundary

In Each Sector*

Direction Sector	Distance (Meters)
N	640
NNE	460
NE	410
ENE	400
E	405
ESE	430
SE	522
SSE	540
S	487
SSW	530
SW	760
WSW	1760
W	1900
WNW	990
NW	617
NNW	590

* Distances are measured from a point equidistant from the reactor building and the stack. The distance between the two is 16 meters.

EXHIBIT 2000-ADM-4532.04-3

OCNGS Dispersion Parameters (X/Q) For Elevated Releases
Standard Distances (sec/m3)

Sector	522m .32 miles	805 m .5 miles	966m .62 miles	3218 m 2 miles	4827 m 3 miles	6436 m 4 miles	8045 m 5 miles	16090 m 10 miles	38120 m 20 miles	48270 m 30 miles	64360 m 40 miles	80450 m 50 miles
N	8.61E-09	1.01E-08	1.07E-08	2.05E-08	1.77E-08	1.45E-08	1.18E-08	5.88E-09	2.27E-09	1.73E-09	1.24E-09	9.52E-10
NNE	7.94E-09	9.69E-09	1.03E-08	2.51E-08	2.28E-08	2.08E-08	1.69E-08	8.32E-09	3.13E-09	2.37E-09	1.69E-09	1.29E-09
NE	5.62E-09	7.52E-09	8.10E-09	1.38E-08	1.40E-08	1.26E-08	1.09E-08	6.07E-09	2.54E-09	1.97E-09	1.44E-09	1.13E-09
ENE	1.24E-08	1.42E-08	1.39E-08	1.42E-08	1.34E-08	1.16E-08	9.91E-09	5.34E-09	2.22E-09	1.72E-09	1.26E-09	9.86E-10
E	1.38E-08	1.53E-08	1.53E-08	1.51E-08	1.38E-08	1.19E-08	1.00E-08	5.36E-09	2.23E-09	1.74E-09	1.28E-09	1.00E-09
ESE	2.22E-08	2.27E-08	2.21E-08	1.85E-08	1.61E-08	1.35E-08	1.12E-08	5.65E-09	2.25E-09	1.73E-09	1.26E-09	9.77E-10
SE	3.07E-08	2.56E-08	2.39E-08	1.87E-08	1.64E-08	1.38E-08	1.14E-08	5.89E-09	2.37E-09	1.83E-09	1.34E-09	1.04E-09
SSE	1.32E-08	1.35E-08	1.31E-08	1.29E-08	1.18E-08	1.01E-08	8.44E-09	4.42E-09	1.79E-09	1.39E-09	1.01E-09	7.90E-10
S	3.02E-09	5.36E-09	6.05E-09	1.27E-08	1.13E-08	9.44E-09	7.83E-09	4.05E-09	1.65E-09	1.27E-09	9.33F-10	7.27E-10
SSW	4.69E-09	5.44E-09	5.68E-09	1.85E-08	2.04E-08	1.51E-08	1.18E-08	5.50E-09	2.03E-09	1.53E-09	1.09E-09	8.35E-10
SW	1.15E-08	1.19E-08	1.19E-08	2.48E-08	2.45E-08	1.78E-08	1.36E-08	6.27E-09	2.25E-09	1.69E-09	1.20E-09	9.09E-10
WSW	1.64E-08	1.56E-08	1.55E-08	1.91E-08	1.93E-08	1.44E-08	1.12E-08	5.66E-09	1.97E-09	1.47E-09	1.03E-09	7.80E-10
W	1.08E-08	1.30E-08	1.33E-08	1.58E-08	1.59E-08	1.33E-08	1.03E-08	5.38E-09	1.88E-09	1.40E-09	9.90E-10	7.50E-10
WNW	9.48E-09	1.13E-08	1.15E-08	1.89E-08	1.51E-08	1.27E-08	9.76E-09	5.04E-09	1.78E-09	1.34E-09	9.44E-10	7.16E-10
NW	1.12E-08	1.30E-08	1.31E-08	2.39E-08	1.92E-08	1.44E-08	1.13E-08	5.74E-09	2.01E-09	1.50E-09	1.06E-09	8.02E-10
NNW	7.94E-09	9.43E-09	9.92E-09	1.82E-08	1.75E-08	1.36E-08	1.08E-08	5.06E-09	1.87E-09	1.41E-09	1.01E-09	7.67E-10

Period of Record -> 01JAN89 through 31DEC90

SOURCE: SEEDS Version 90.20 / Forked River Meteorological Tower

EXHIBIT 2000-ADM-4532.04-3

OCNGS Dispersion Parameters (X/Q) For Ground Releases
Standard Distances (sec/m3)

Sector	522m .32 miles	805 m .5 miles	966m .62 miles	3218 m 2 miles	4827 m 3 miles	6436 m 4 miles	8045 m 5 miles	16090 m 10 miles	38120 m 20 miles	48270 m 30 miles	64360 m 40 miles	80450 m 50 miles
N	1.18E-05	1.07E-05	6.36E-06	9.75E-07	5.87E-07	4.03E-07	2.84E-07	1.13E-07	4.09E-08	3.09E-08	2.17E-08	1.74E-08
NNE	1.84E-05	1.35E-05	9.94E-06	1.28E-06	7.65E-07	5.25E-07	3.64E-07	1.44E-07	5.12E-08	3.83E-08	2.68E-08	2.14E-08
NE	2.37E-05	1.46E-05	1.28E-05	1.40E-06	8.38E-07	5.76E-07	4.01E-07	1.59E-07	5.68E-08	4.24E-08	2.96E-08	2.37E-08
ENE	3.07E-05	1.32E-05	1.66E-05	1.24E-06	7.44E-07	5.12E-07	3.57E-07	1.42E-07	5.07E-08	3.80E-08	2.66E-08	2.12E-08
E	3.26E-05	1.36E-05	1.76E-05	1.27E-06	7.62E-07	5.25E-07	3.68E-07	1.47E-07	5.28E-08	3.95E-08	2.77E-08	2.21E-08
ESE	2.50E-05	1.28E-05	1.35E-05	1.17E-06	7.02E-07	4.83E-07	3.38E-07	1.35E-07	4.85E-08	3.64E-08	2.56E-08	2.05E-08
SE	2.15E-05	1.39E-05	1.16E-05	1.27E-06	7.63E-07	5.26E-07	3.70E-07	1.48E-07	5.36E-08	4.03E-08	2.84E-08	2.27E-08
SSE	1.95E-05	1.17E-05	1.05E-05	1.06E-06	6.41E-07	4.42E-07	3.13E-07	1.25E-07	4.55E-08	3.43E-08	2.41E-08	1.93E-08
S	1.16E-05	1.03E-05	6.24E-06	9.29E-07	5.62E-07	3.88E-07	2.75E-07	1.10E-07	4.02E-08	3.05E-08	2.15E-08	1.72E-08
SSW	7.55E-06	6.20E-06	4.08E-06	5.66E-07	3.39E-07	2.32E-07	1.62E-07	6.39E-08	2.30E-08	1.73E-08	1.21E-08	9.71E-09
SW	8.11E-06	7.64E-06	4.38E-06	6.95E-07	4.15E-07	2.84E-07	1.95E-07	7.68E-08	2.75E-08	2.06E-08	1.44E-08	1.15E-08
WSW	6.85E-06	5.64E-06	3.70E-06	5.29E-07	3.13E-07	2.13E-07	1.45E-07	5.65E-08	1.99E-08	1.48E-08	1.03E-08	8.20E-09
W	5.07E-06	5.14E-06	2.74E-06	4.66E-07	2.80E-07	1.92E-07	1.34E-07	5.30E-08	1.91E-08	1.44E-08	1.01E-08	8.09E-09
WNW	4.96E-06	4.47E-06	2.65E-06	4.02E-07	2.41E-07	1.66E-07	1.16E-07	4.60E-08	1.66E-08	1.25E-08	8.82E-09	7.05E-09
NW	5.92E-06	5.46E-06	3.20E-06	4.93E-07	2.95E-07	2.03E-07	1.41E-07	5.59E-08	2.02E-08	1.52E-08	1.07E-08	8.54E-09
NNW	8.65E-06	7.92E-06	4.67E-06	7.14E-07	4.29E-07	2.95E-07	2.07E-07	8.23E-08	2.99E-08	2.26E-08	1.59E-08	1.28E-08

Period of Record -> 01JAN89 through 31DEC90

SOURCE: SEEDS Version 90.20 / Forked River Meteorological Tower

EXHIBIT 2000-ADM-4532.04-3

OCNGS Dispersion Parameters (D/Q) For Elevated Releases
Standard Distances (m-2)

Sector	522m .32 miles	805 m .5 miles	966m .62 miles	3218 m 2 miles	4827 m 3 miles	6436 m 4 miles	8045 m 5 miles	16090 m 10 miles	38120 m 20 miles	48270 m 30 miles	64360 m 40 miles	80450 m 50 miles
N	8.62E-09	5.18E-09	4.25E-09	5.54E-10	2.78E-10	1.86E-10	1.39E-10	6.81E-11	2.06E-11	1.31E-11	7.62E-12	4.86E-12
NNE	1.12E-08	7.06E-09	5.90E-09	7.90E-10	3.99E-10	2.76E-10	2.13E-10	1.14E-10	3.50E-11	2.21E-11	1.26E-11	7.87E-12
NE	6.14E-09	3.80E-09	3.11E-09	4.19E-10	2.25E-10	1.78E-10	1.56E-10	1.05E-10	3.36E-11	2.10E-11	1.18E-11	7.25E-12
ENE	8.18E-09	4.80E-09	3.81E-09	4.98E-10	2.61E-10	1.93E-10	1.60E-10	9.70E-11	3.07E-11	1.94E-11	1.10E-11	6.90E-12
E	9.27E-09	5.40E-09	4.23E-09	5.52E-10	2.87E-10	2.06E-10	1.65E-10	9.53E-11	2.99E-11	1.90E-11	1.09E-11	6.88E-12
ESE	1.48E-08	8.52E-09	6.61E-09	8.56E-10	4.35E-10	2.94E-10	2.22E-10	1.11E-10	3.42E-11	2.20E-11	1.29E-11	8.28E-12
SE	1.45E-08	8.25E-09	6.38E-09	8.26E-10	4.20E-10	2.83E-10	2.13E-10	1.06E-10	3.26E-11	2.10E-11	1.23E-11	7.94E-12
SSE	7.18E-09	4.26E-09	3.38E-09	4.45E-10	2.30E-10	1.64E-10	1.31E-10	7.43E-11	2.32E-11	1.47E-11	8.43E-12	5.31E-12
S	4.69E-09	2.90E-09	2.46E-09	3.23E-10	1.65E-10	1.18E-10	9.38E-11	5.35E-11	1.66E-11	1.04E-11	5.91E-12	3.68E-12
SSW	4.68E-09	2.98E-09	2.54E-09	3.38E-10	1.67E-10	1.09E-10	7.96E-11	3.70E-11	1.09E-11	6.94E-12	3.99E-12	2.53E-12
SW	9.25E-09	5.83E-09	4.86E-09	6.45E-10	3.15E-10	1.98E-10	1.37E-10	5.45E-11	1.56E-11	1.00E-11	5.91E-12	3.84E-12
WSW	9.29E-09	5.65E-09	4.57E-09	6.01E-10	2.94E-10	1.83E-10	1.26E-10	4.79E-11	1.37E-11	8.90E-12	5.35E-12	3.53E-12
W	6.00E-09	3.59E-09	2.86E-09	3.75E-10	1.85E-10	1.17E-10	8.20E-11	3.35E-11	9.78E-12	6.36E-12	3.80E-12	2.50E-12
WNW	4.73E-09	2.88E-09	2.31E-09	3.05E-10	1.50E-10	9.52E-11	6.66E-11	2.72E-11	7.90E-12	5.12E-12	3.05E-12	1.99E-12
NW	6.82E-09	4.07E-09	3.25E-09	4.24E-10	2.10E-10	1.35E-10	9.56E-11	4.07E-11	1.20E-11	7.78E-12	4.63E-12	3.02E-12
NNW	6.75E-09	4.07E-09	3.31E-09	4.34E-10	2.16E-10	1.41E-10	1.02E-10	4.67E-11	1.39E-11	8.93E-12	5.23E-12	3.37E-12

Period of Record -> 01JAN89 through 31DEC90

SOURCE: SEEDS Version 90.20 / Forked River Meteorological Tower

EXHIBIT 2000-ADM-4532.04-3

OCNGS Dispersion Parameters (D/Q) For Ground Releases
Standard Distances (m-2)

Sector	522m .32 miles	805 m .5 miles	966m .62 miles	3218 m 2 miles	4827 m 3 miles	6436 m 4 miles	8045 m 5 miles	16090 m 10 miles	38120 m 20 miles	48270 m 30 miles	64360 m 40 miles	80450 m 50 miles
N	2.31E-08	1.23E-08	8.69E-09	1.12E-09	5.10E-10	3.27E-10	2.29E-10	6.57E-11	1.54E-11	9.98E-12	5.74E-12	3.74E-12
NNE	3.76E-08	1.98E-08	1.41E-08	1.80E-09	8.23E-10	5.28E-10	3.70E-10	1.06E-10	2.48E-11	1.61E-11	9.26E-12	6.03E-12
NE	3.26E-08	1.67E-08	1.23E-08	1.52E-09	6.95E-10	4.45E-10	3.12E-10	8.94E-11	2.09E-11	1.36E-11	7.82E-12	5.09E-12
ENE	3.61E-08	1.59E-08	1.36E-08	1.45E-09	6.64E-10	4.26E-10	2.98E-10	8.54E-11	2.00E-11	1.30E-11	7.47E-12	4.86E-12
E	3.60E-08	1.60E-08	1.35E-08	1.46E-09	6.66E-10	4.27E-10	2.99E-10	8.57E-11	2.00E-11	1.30E-11	7.49E-12	4.88E-12
ESE	3.71E-08	1.97E-08	1.40E-08	1.80E-09	8.22E-10	5.27E-10	3.70E-10	1.06E-10	2.47E-11	1.61E-11	9.25E-12	6.03E-12
SE	3.52E-08	1.89E-08	1.37E-08	1.72E-09	7.88E-10	5.05E-10	3.54E-10	1.01E-10	2.37E-11	1.54E-11	8.87E-12	5.77E-12
SSE	2.50E-08	1.25E-08	9.38E-09	1.14E-09	5.22E-10	3.35E-10	2.35E-10	6.71E-11	1.57E-11	1.02E-11	5.87E-12	3.82E-12
S	1.50E-08	9.02E-09	5.62E-09	8.21E-10	3.76E-10	2.41E-10	1.69E-10	4.83E-11	1.13E-11	7.35E-12	4.23E-12	2.75E-12
SSW	1.24E-08	6.80E-09	4.65E-09	6.20E-10	2.83E-10	1.82E-10	1.27E-10	3.65E-11	8.52E-12	5.54E-12	3.19E-12	2.08E-12
SW	1.91E-08	1.08E-08	7.19E-09	9.80E-10	4.48E-10	2.87E-10	2.02E-10	5.77E-11	1.35E-11	8.77E-12	5.04E-12	3.28E-12
WSW	1.70E-08	9.66E-09	6.39E-09	8.80E-10	4.02E-10	2.58E-10	1.81E-10	5.18E-11	1.21E-11	7.87E-12	4.53E-12	2.95E-12
W	1.18E-08	6.53E-09	4.44E-09	5.94E-10	2.72E-10	1.74E-10	1.22E-10	3.50E-11	8.18E-12	5.32E-12	3.06E-12	1.99E-12
WNW	9.22E-09	5.30E-09	3.47E-09	4.83E-10	2.21E-10	1.42E-10	9.93E-11	2.84E-11	6.64E-12	4.32E-12	2.48E-12	1.62E-12
NW	1.52E-08	7.76E-09	5.70E-09	7.06E-10	3.23E-10	2.07E-10	1.45E-10	4.16E-11	9.72E-12	6.32E-12	3.63E-12	2.37E-12
NNW	1.74E-08	8.63E-09	6.54E-09	7.86E-10	3.59E-10	2.30E-10	1.62E-10	4.62E-11	1.08E-11	7.03E-12	4.04E-12	2.63E-12

Period of Record -> 01JAN89 through 31DEC90

SOURCE: SEEDS Version 90.20 / Forked River Meteorological Tower

GPU Nuclear

EXHIBIT 2000-ADM-4532.04-4

OCNGS Usage Factors For Individual Dose Assessment[a]

Liquid Ingestion Parameters	Usage Factor
Fraction Of Produce From Local Garden	7.6E-1
Soil Density In Plow Layer (Kg/m^2)	2.4E+2
Fraction Of Activity Retained On Spray Vegetation	2.5E-1
Shielding Factor For Residential Structures	7.0E-1
Period Of Buildup Of Activity In Soil (hr)	1.31E+5
Period of Pasture Grass Exposure to Activity (hr)	7.2E+2
Period Of Crop Exposure to Activity (hr)	1.44E+3
Delay Time For Ingestion Of Stored Feed By Animals (hr)	2.16E+3
Delay Time For Ingestion Of Leafy Vegetables By Man (hr)	2.4E+1
Delay Time For Ingestion Of Other Vegetables By Man (hr)	1.44E+3
Transport Time Milk-Man (hr)	4.8E+1
Time Between Slaughter and Consumption of Meat Animal (hr)	4.8E+2
Grass Yield Wet Weight (Kg/m^2)	7.0E-1
Other Vegetation Yield Wet-Weight (Kg/m^2)	2.0
Weathering Rate Constant For Activity on Veg. (hr^{-1})	2.1E-3
Milk Cow Feed Consumption Rate (Kg/day)	5.0E+1
Goat Feed Consumption Rate (Kg/day)	6.0
Beef Cattle Feed Consumption Rate (Kg/day)	5.0E+1
Milk Cow Water Consumption Rate (L/day)	6.0E+1
Goat Water Consumption Rate (L/day)	8.0
Beef Cattle Water Consumption Rate (L/day)	5.0E+1
Environmental Transit Time For Water Ingestion (hr)	1.2E+1
Environmental Transit Time For Fish Ingestion (hr)	2.4E+1
Environmental Transit Time For Invertebrate Ingestion (hr)	2.4E+1

[a] From USNRC Reg. Guide 1.109, Tables E-5 and E-15.

EXHIBIT 2000-ADM-4532.04-4 (Continued)

OCNGS Usage Factors For Individual Dose Assessment

Liquid Ingestion Parameters	Usage Factor
Environmental Transit Time For Shore Exposure (hr)	0
Water Ingestion (L/yr)	
a. Adult	7.3E+2
b. Teen	5.1E+2
c. Child	5.1E+2
d. Infant	3.3E+2
Shore Exposure (hr/yr)	
a. Adult	1.2E+1
b. Teen	6.7E+1
c. Child	1.4E+1
d. Infant	0
Salt Water Sport Fish Ingestion (Kg/yr)	
a. Adult	2.1E+1
b. Teen	1.6E+1
c. Child	6.9
d. Infant	0
Salt Water Commiecial Fish Ingestion (Kg/hr)	
a. Adult	2.1E+1
b. Teen	1.6E+1
c. Child	6.9
d. Infant	0
Salt Water Invertebrate Ingestion (Kg/hr)	
a. Adult	5.0
b. Teen	3.8
c. Child	1.7
d. Infant	0
Irrigated Leafy Vegetable Ingestion (Kg/yr)	
a. Adult	6.4E+1
b. Teen	4.2E+1
c. Child	2.6E+1
d. Infant	0
Irrigated Other Vegetable Ingestion (Kg/yr)	
a. Adult	5.2E+2
b. Teen	6.3E+2
c. Child	5.2E+2
d. Infant	0
Irrigated Root Vegetable Ingestion (Kg/yr)	
a. Adult	5.2E+2
b. Teen	6.3E+2
c. Child	5.2E+2
d. Infant	0

EXHIBIT 2000-ADM-4532.04-4 (Continued)

OCNGS Usage Factors For Individual Dose Assessment

Liquid Ingestion Parameters	Usage Factor
Environmental Transit Time For Shore Exposure (hr)	0
Irrigated Cow and Goat Milk Ingestion (L/yr)	
a. Adult	3.1E+2
b. Teen	4.0E+2
c. Child	3.3E+2
d. Infant	3.3E+2
Irrigated Beef Ingestion (Kg/yr)	
a. Adult	1.1E+2
b. Teen	6.5E+1
c. Child	4.1E+1
d. Infant	0

EXHIBIT 2000-ADM-4532.04-5

RADIONUCLIDES IN LIQUID EFFLUENT USED TO

ASSESS COMPLIANCE WITH TECHNICAL SPECIFICATIONS

H-3	Ru-103
Kr-85m	Ru-106
Xe-131m	Ag-110m
Xe-133m	Sb-124
Xe-133	Sb-125
Xe-135	Te-132
Cr-51	I-131
Mn-54	I-133
Fe-59	I-135
Co-58	Cs-134
Co-60	Cs-136
Zn-65	Cs-137
Sr-89	Ba-140
Sr-90	La-140
Nb-95	Ce-141
Zr-95	Ce-143
Mo-99	Ce-144
	Np-239

EXHIBIT 2000-ADM-4532.04-6

RADIONUCLIDES IN GASEOUS EFFLUENT USED TO

ASSESS COMPLIANCE WITH TECHNICAL SPECIFICATIONS

Kr-85m	Co-58
Kr-85	Co-60
Kr-87	Zn-65
Kr-88	Sr-89
Kr-89	Sr-90
Xe-131m	Zr-95
Xe-133m	Ru-103
Xe-133	Ru-106
Xe-135m	Ag-110m
Xe-135	Sb-124
Xe-137	Sb-125
Xe-138	Cs-134
H-3	Cs-136
I-131	Cs-137
I-133	Ba-140
Mn-54	Ce-141
Fe-59	Ce-144