OFFSITE DOSE CALCULATION MANUAL
FOR
GASEOUS AND LIQUID EFFLUENTS
FROM THE
TURKEY POINT PLANT UNITS 3 & 4

Florida Power and Light Company

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1.0 Introduction

This Manual describes acceptable methods of calculating radioactivity concentrations in the environment and the potentially resultant doses* offsite** that are associated with liquid and gaseous effluents from the Turkey Point Nuclear Plant. The radioactivity concentrations and dose estimates are used to demonstrate compliance with Technical Specifications required by 10 CFR 50.36. The methodology stated in this Manual is acceptable for use in demonstrating operational compliance with 10 CFR 20.106, CFR 50 Appendix I, and 40 CFR 190. Only the dose attributable to the Turkey Point Units 3 and 4 is considered in demonstrating compliance with 40 CFR 190 since no other nuclear facility exists within 50 miles of the Plant.

Monthly calculations are made to guide the management of station effluents and to verify that potential radioactivity concentrations and doses offsite satisfy the Technical Specifications. The receptor is described such that the exposure of any resident near the plant is unlikely to be underestimated. Even more conservative conditions (e.g. location and/or exposure pathways expected to yield higher computed doses) than appropriate for the maximally exposed person may be assumed when calculating the concentration or dose.

Monthly calculations made to assure that air dose and dose commitment specifications are not exceeded are based on atmospheric dispersion and deposition of gaseous effluents derived from reference meteorological conditions.*** Calculations made to assess the radioactive noble gas dose to air are based on the location offsite that could be occupied by a person where the maximum air dose is expected.

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* Dose is commonly used to mean personal dose equivalent commitment.

** Offsite means outside the exclusion area.

*** Reference meteorological conditions are annual averaged conditions during years 1976 and 1977.

Calculations of dose committed from radioactive releases over extended time (3 and 12 months) are also made for the purpose of verifying compliance with regulatory limits on offsite dose. For these calculations the receptor is selected on the basis of the combination of applicable exposure pathways identified in the land use census and the maximum ground level X/Q at a residence, or on the basis of more conservative conditions such that the dose to any resident near the Plant is unlikely to be underestimated.

2.0 Liquid Effluent

2.1 Radioactivity Concentration In Liquid Waste

The concentration of radionuclides in liquid waste is determined by sampling and analysis in accordance with Table 3.9-1 of the Technical Specifications. When a radionuclide concentration is below the lower limit of detection (LLD) for the analysis, it is not reported as being present in the sample.

2.2 Radioactivity Concentration in Water at the Restricted Area Boundary

Technical Specifications 3.9.1.a.1 and 3.9.1.a.2 require measured radioactivity concentrations in liquid releases to be used to calculate the fraction of the unrestricted area maximum permissible concentration (MPC) (10 CFR 20, Appendix B, Table 2, Column 2). For compliance with these specifications, the restricted area boundary is assumed to be at the end of the condenser cooling water mixing basin where water enters the system of cooling canals.

Radioactive material in liquid effluent is diluted by condenser cooling water from fossil units 1 and 2 and from nuclear units 3 and 4. When estimating the unrestricted area activity concentration in water, the total condenser cooling water flow into the condenser cooling water mixing basin from operating condenser cooling water pumps at the four units is assumed for dilution.

2.2.1 Aqueous Concentration. The diluted concentration of radionuclide i in the condenser cooling water mixing basin and its outflow is estimated with the equation

$$C_{zi} = C_i \cdot \frac{F_1}{F_2} \qquad (1)$$

where C_i = concentration of radionuclide i in liquid radwaste released (μCi/ml)

C_{zi} = concentration of radionuclide i in the water in the condenser cooling water mixing basin and its outflow (μCi/ml)

F_1/F_2 = dilution

F_1 = flow in radioactive liquid discharge line (gal/min)*

F_2 = minimum total condenser cooling water flow (gal/min)* Value not greater than the rated total condenser cooling water flow from operating condenser cooling water pumps at the four units.

2.2.2 Batch Release. A sample of each batch of liquid radwaste is analyzed before release for I-131 and other principal gamma emitters, or for total activity concentration. The fraction of the unrestricted area MPC present in the batch tank, FMB_b is derived either from isotopic analyses or from gross β-γ analysis.

With the activity concentration in a batch sample based on cumulative or gross β-γ activity alone, the fraction of the unrestricted area MPC in the batch is estimated by

$$FMB_b = \frac{C_{bp}}{3 \times 10^{-8}} \qquad (2)$$

where C_{bp} = activity concentration in batch sample measured by gross β-γ analysis or cumulative total of isotopic analysis (μCi/ml)

3×10^{-8} = unrestricted area MPC for unidentified radionuclides in water (μCi/ml)

Alternately, when the fraction of the unrestricted area MPC is derived from isotopic analyses identifying ioding and principal gamma emitters, FMB_b is calculated with the equation

$$FMB_b = \left[\sum_i \left(\frac{C_{bpi}}{MPC_i} \right)_{\text{identified}} \right] \div E_b \qquad (3)$$

*F_1 and F_2 may have any suitable but identical units of flow (volume/time).

where C_{bpi} = concentration of radionuclide i (including I-131 and principal gamma emitters) in a batch sample measured prior to release (μCi/ml)

MPC_i = activity concentration limit in water of radionuclide i according to 10 CFR 20, Appendix B, Table 2, column 2 (μCi/ml)

$$E_b = \left[\frac{\text{Quarterly average of the fraction of MPC in the batch tank due to I-131 and principal gamma emitters}}{\text{Quarterly average of the fraction of MPC in the batch tank due to all radionuclides measured}}\right]$$

E_b is an adjustment to account for radionuclides not measured prior to release but measured in the monthly and quarterly sample per Technical Specification Table 3.9-1. The value of E_b has been determined based on past operating data and is

$$E_b = 0.8$$

The fraction of the unrestricted area MPC present in the condenser cooling water mixing basin outflow due to a batch release, $FMPC_b$ may be calculated with the equation

$$FMPC_b = FMB_b \cdot \frac{F_{1s}}{F_2} \qquad (4)$$

where FMB_b = fraction of the unrestricted area MPC present in the batch tank from equation (2) if cumulative or gross activity if used or from equation (3) if isotopic analysis is used.

F_{1s} = flow in the batch release line (gal/min).* Value not less than the rated or measured pumping rate through the batch discharge line.

*F_1 and F_2 may have any suitable but identical units of flow (volume/time).

2.2.3 Continuous Release. Continuous aqueous discharges are sampled and analyzed according to the schedule in Technical Specifications Table 3.9-1. The fraction of the unrestricted area MPC present in a continuously discharged radioactive stream, FC_c, is derived either from isotopic analyses or from gross β-γ analysis. With the activity concentration in a continuous radioactive release stream based on the cumulative or gross β-γ activity alone, the fraction of the unrestricted area MPC in the waste stream is estimated with

$$FC_c = \frac{C_c}{3 \times 10^{-8}} \qquad (5)$$

where C_c = activity concentration in continuous release measured by gross β-γ analysis or cumulative total of isotopic analysis (μCi/ml)

Alternately, when the fraction of the unrestricted area MPC is derived from isotopic analyses, FC_c, is estimated with the equation

$$FC_c = \left[\sum_i \left(\frac{C_{cwi}}{MPC_i}\right)_{identified}\right] \div E_c \qquad (6)$$

where C_{cwi} = concentration of radionuclide i (including I-131 and principal gamma emitters) measured in weekly sample of continuous discharge stream

$$E_c = \left[\frac{\text{Quarterly average fraction of MPC due to I-131 and principal gamma emitters measured in weekly samples of continuous releases during the quarter}}{\text{Quarterly average fraction of MPC due to all radionuclides measured in samples of continuous releases}}\right]$$

E_c is an adjustment to account for radionuclides measured in monthly and quarterly composite samples but not in weekly samples of continuous releases. The value of E_c has been determined based on past operating data and is

$$E_c = 0.9$$

The fraction of the unrestricted area MPC present in the condenser cooling water basin outflow due to continuous aqueous discharges, $FMPC_c$, may be estimated with the equation

$$FMPC_c = FC_c \cdot \frac{F_{1c}}{F_2} \quad (7)$$

where FC_c = fraction of unrestricted area MPC present in each continuously discharged radioactive stream from equation (5) if cumulative or gross activity use or from equation (6) if isotopic analysis used.

F_{1c} = flow of continuously released aqueous radioactive discharge (gal/min)

2.3 Method of Establishing Alarm and Trip setpoints

The alarm/trip setpoint for each liquid effluent radiation monitor is derived from the concentration limit provided in 10 CFR Part 20, Appendix B, Table 2, Column 2 applied in the condenser cooling water mixing basin outflow. That is, the alarm setpoint is based on a concentration limit in the mixing basin outflow. Radiation monitoring and isolation points are located in the steam generator blowdown line, R 19, and the liquid waste disposal system line, R 18, through which radioactive waste effluent is eventually discharged into the canal basin.

The alarm setpoint for each liquid effluent monitor is based upon the measurements performed according to Technical Specification Table 3.9-1, of radioactivity in a batch of liquid to be released or in the continuous aqueous discharge. Alternately, the alarm setpoint may be based upon cumulative or gross β-γ activity of the liquid waste.

2.3.1 Setpoint for a Batch Release. The liquid radwaste effluent line radiation monitor alarm setpoint is determined with the equation

$$S = \frac{A}{FMB_b} \cdot \frac{F_2}{F_{1s}} \cdot g \qquad (8)$$

or a method which gives a lower setpoint value.

where S = radiation monitor alarm setpoint (cpm)

A = counting rate (cpm/ml) or activity concentration (μCi/ml) in laboratory of sample from batch tank

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

F_{1s} = flow in the batch release line (gal/min).* Value not less than the rated or measured pumping rate through the batch discharge line.

F_2 = minimum total condenser cooling water flow (gal/min).* Value not greater than the rated total condenser cooling water pumping rate from pumps operating at the 4 units.

FMB_b = is determined as described in § 2.2.2.

2.3.2 Setpoint for a Continuous Release. The alarm setpoint of the radiation monitor on a continuous radioactive discharge line is determined with the equation

$$S = \frac{A}{FC_c} \cdot \frac{F_2}{F_{1c}} \cdot g \qquad (9)$$

or by a method which gives a lower setpoint value.

where A = activity concentration (μCi/ml) or counting rate (cpm/ml) = in laboratory of weekly sample.

F_{1c} = flow in the radioactive liquid continuous discharge line (ml/sec).* Value not less than the rated or measured pumping rate through the discharge line

F_2 = total condenser cooling water flow (ml/sec).* Value not greater than the rated total condenser cooling water pumping rate from operating pumps at the 4 units.

FC_c = is determined as described in § 2.2.3.

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*F_1 and F_2 may have any convenient units of flow (i.e., volume/time) provided the units are identical.

2.4 Accumulated Personal Dose

Technical Specification 3.9.1.b,2 requires the dose or dose commitment to a person offsite due to radioactive material released in liquid effluent to be calculated on a cumulative basis at least once every 31 days. The requirements is satisfied by evaluating the accumulated dose commitment to a hypothetical adult exposed by eating fish and shellfish taken from the cooling canals.

The model that is used to evaluate doses due to radioactivity in liquid effluents is

$$D_{nk} = \sum_i A_{in}^{fish} \cdot C_{ik} \cdot F_{1k} \cdot t_k \cdot \frac{}{V \cdot \lambda_i^e} + \sum_i A_{in}^{shellfish} \cdot C_{ik} \cdot F_{1k} \cdot t_k \cdot \frac{}{V \cdot \lambda_i^e} \qquad (10)$$

where D_{nk} = the dose commitment (mrem) to organ n due to the radionuclides identified in sample analysis k where the analyses are those required by technical specifications Table 3.9-1. (The contribution to the dose from gamma emitters become available on a batch basis for batch releases and on a weekly basis for continuous releases. Similarly the contributions from H-3 are available on a monthly basis and the contributions from Sr-89 and Sr-90 become available on a quarterly basis.)

A_{in} = transfer factor relating a unit aqueous concentration of radionuclide i (μCi) to dose commitment rate to organ n or total body of an exposed person tabulated in the ODCM Appendix A (mrem/hr per μCi/ml)

C_{ik} = the concentration of radionuclide i in the undiluted liquid waste to be discharged (μCi/ml)

t_k = period of time (hours) during which liquid waste represented by analysis k is discharged

F_{1k} = liquid waste discharge flow during release represented by sample k (gal/min)

λ_i^e = effective decay constant ($minute^{-1}$) for nuclide i, $(\lambda_i + F_3/V)$ where λ_i is the radioactive decay constant

F_3 = canal-ground water interchange flow, approximately 2.25×10^5 gal/min

V = cooling canal effective volume, approximately 3.75×10^9 gallons

Summation over releases represented by the various samples analyzed gives the total dose to each organ

$$D_n = \sum_k D_{nk} \qquad (11)$$

Where D_n = the dose commitment to organ n, including total body, of the maximally exposed person during the quarter to date (mrem)

For the quarterly dose assessments to be included in the Annual Radiological Environmental Monitoring Report required by Specification 6.9.4.b, doses will be calculated with equations (10) and (11) for all age groups and organs on the basis of radionuclides measured in liquid radioactive effluent according to the sampling and analyses required in Technical Specification Table 3.9-1.

Based on an evaluation of the radionuclide distribution typical in liquid radioactive effluents, the calculated doses to individuals have been determined to be dominated by the radionuclides, Co-58, Co-60, Nb-95, Ag-110m, Cs-134, and Cs-137. These 6 nuclides typically contribute 90% or more of the adult's total body dose and the adult's GI-LLI dose, which is the critical organ and critical age group. Therefore, the dose commitment due to radioactivity in liquid effluents may be reasonably evaluated by limiting the dose calculational process to these radionuclides for the total body and the GI-LLI. Equation (10) can be simplified in the following ways.

1) The transfer factor ($A_{in}^{pathway}$) and the effective decay constant (λ_i^e) can be combined into a single factor $\left(\frac{A_i^{pathway}}{\lambda_i^e}\right)$

2) The concentration (C_{ik}), liquid discharge flow (F_{lk}), and time of the release (t_k) can be combined to provide the total release for a specified period of time (eg, Ci/calendar quarter).

These combinations provide the following simplified equation

$$D_n = \frac{1}{0.8 \cdot V} \sum_i Q_i \cdot \left(\frac{A_{in}^{fish} + A_{in}^{shellfish}}{\lambda_i^e} \right) \qquad (12)$$

where Q_i = the total release of radionuclide i for the specified period of time, e, monthly

0.8 = a conservatism factor to allow for variability in radionuclide distribution (only include if using simplified approach).

Refer to Appendix C for a detailed evaluation and explanation of this simplified approach.

Monthly dose calculations required by Specifications 3.9.2.b.1 and 3.9.2.~~b.2~~ c.1 may be done with either equations 10 and 11 or equation 12.

When equation 12 is used, the doses should be calculated for the adult total body and adult GI-LLI from the cumulative release for the radionuclides Co-58, Co-60, Nb-95, Ag-110m, Cs-134, and Cs-137.

2.5 Projected Personal Dose

Technical Specification 3.9.1.d.1 requires the doses to a person offsite due to radioactive material released in liquid effluent to be projected over a quarter at least one time during each month.

This requirement is satisfied by calculating the total body and organ dose commitments to a hypothetical adult exposed by eating fish and shellfish taken from the cooling canals. Section 2.4 includes the method for doing this calculation. Appendix F presents the technical bases for using this projection to determine liquid radwaste equipment operation. The dose from liquid effluents is projected by extrapolating the dose commitment to date during the current quarter to include the entire quarter.

On the basis of total activity released, dose commitment is projected with the relation

$$P = \frac{91 \cdot D_n}{X} \qquad (13)$$

where P = the projected total body or critical organ dose commitment to the maximally exposed person (mrem)

91 = number of days in a calendar quarter

X = number of days in current quarter represented by available radioactive discharge data

3.0 Gaseous Effluent

3.1 Introduction

Units 3 and 4 discharge gaseous effluent through the plant vent, Unit 3 Spent Fuel Pit vent, air ejector vents, and steam generator blowdown vents. These gaseous effluent streams, radioactivity monitoring points, and effluent discharge points are illustrated schematically in Figure 3-1.

3.2 Radioactivity in Gaseous Effluent

For the purpose of estimating offsite radionuclide concentrations and radiation doses, measured radionuclide concentrations in gaseous effluent and in ventilation air exhausted from the Plant are relied upon. Table 3.9-3 in the Technical Specifications identifies specific radionuclides in gaseous discharges for which sampling and analysis is done.

When a radionuclide concentration is below the LLD for the analysis, it is not reported as being present in the sample.

Noble Gases. The distribution of radioactive noble gases in a gaseous effluent stream is determined by gamma spectrum analysis of identifiable radionuclides in effluent gas sample(s). Results of one or more previous analyses may be averaged to obtain a representative spectrum. In the event the distribution is unobtainable from measured data, the distribution of radioactive noble gases appearing in Table 3-2 herein may be assumed.

Some gaseous effluents from both Units 3 and 4, whose sources are identified in Table 3-1, discharge in common through the Plant Vent. When needed to assure that the effluents are within allowable limits on a per reactor bases, the measured release from the Plant Vent is apportioned to each unit on a ratio equal to the ratio of primary coolant radioactivity concentration in the two reactors during the quarter.

When calculating atmospheric dispersion of gaseous effluent, gaseous discharges from Units 3 and 4 are treated as a mixed mode release from a single composite vent.

3.3 Effluent Noble Gas Monitor Alarm Setpoint

Instrumentation is provided to monitor gamma radiation from radioactive materials released from the Plant in gaseous effluents. Each monitor includes an alarm that is set to report at or below the level at which radioactive noble gas in gaseous effluent from a monitored stack or vent exceeds a rate calculated to cause a noble gas concentration offsite equal to that specified in 10 CFR 20, Appendix B, Table 2, Column 1 for the mixture. Setting effluent noble gas monitors to trigger an alarm at or below the concentration limit assures that action can be taken to ensure that the unrestricted area concentration specified by 10 CFR Part 20.106 and the corresponding dose rate limits in Specification 3.9.2.a are not exceeded.

The gross activity concentration of noble gas corresponding to the 10 CFR Part 20, Appendix B, Table 2, Column 1 limit is calculated from the distribution determined in § 3.2 with the equation

$$MPC = C \div \sum_i \frac{C_i}{MPC_i} \qquad (14)$$

where MPC = gross activity concentration of noble gas mixture corresponding to 10 CFR Part 20 Appendix B, Table 2 column 1 limit ($\mu Ci/cm^3$)

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

$C = \sum C_i$ = activity concentration of noble gas mixture released ($\mu Ci/cm^3$)

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

Note that this is simply the aggregate of the concentrations of radio nuclides in a sample divided by the fraction or multiple of MPC constituted by radionuclides in the same sample. For purposes of simplifying evaluations of MPC, the total activity concentration of the noble gases may be assumed to be Kr-88, which has the most restrictive MPC value (2×10^{-8} $\mu Ci/ml$) of the noble gases.

The alarm setpoint for the effluent noble gas monitor is then calculated with the equation

$$S = \frac{MPC \cdot h}{4.7 \times 10^{-4} \cdot F \cdot \frac{X}{Q}} \qquad (15)$$

or by a method that gives a lower setpoint

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

h = effluent noble gas monitor counting rate response $\left(\frac{\text{cpm}}{\mu\text{Ci/cm}^3}\right)$ or calibration $\left(\frac{\text{mR/hr}}{\mu\text{Ci/cm}^3}\right)$ for noble gas gamma radiation

F = discharge rate of gaseous effluent (ft^3/min)

X/Q = atmospheric dispersion for mixed mode release from release point to unrestricted area ($\mu\text{Ci/m}^3$ per $\mu\text{Ci/sec}$).

4.7×10^{-4} = conversion constant $\left(\frac{1\ \text{m}^3}{35.31\ \text{ft}^3} \cdot \frac{1\ \text{min}}{60\ \text{sec}}\right)$

MPC = maximum permissible concentration according to 10 CFR Part 20, Appendix B, Table 2, Column 1 ($\mu\text{Ci/cm}^3$)

The value of X/Q adopted in a setpoint calculation will be based either on prevailing meteorological conditions or on reference meteorological conditions. Minimum atmospheric dispersion offsite derived from reference meteorological conditions* at the site boundary 1950 meters SSE of the Plant are:

$$\frac{X}{Q} = 5.8 \times 10^{-7}\ \text{sec/m}^3$$

3.4 Noble Gas Gamma Radiation Dose Accumulated in Air

Technical Specification 3:9.2.b requires that the offsite air dose from noble gas gamma radiation not exceed 5 mrad per reactor during any calendar quarter. Specification 3.9.2.b.2 requires an evaluation be performed monthly to verify that the accumulated air dose does not exceed the limit.

The quantity of radioactive noble gas discharged during an interval of time is determined by integrating the release rate measurement of each effluent noble gas monitor identified in Figure 3-1. The total measured radioactivity discharged via a stack or vent during a counting interval is determined by the relation

$$Q_j = \frac{N_j \cdot F}{3.53 \times 10^{-5} \cdot h} \qquad (16)$$

*Reference meteorological data are tabulated in Tables 3.5, 3.6 and 3.7. Their derivation is described in Revised Radiological Effluent Technical Specifications: Gaseous Effluent Dilution Factors, Florida

where Q_j = total measured gaseous radioactivity release via a stack or vent during counting interval j (μCi)

3.53×10^{-5} = conversion constant (ft^3/cm^3)

N_j = counts accumulated during counting interval j

F = discharge rate of gaseous effluent stream (ft^3/min)

h = effluent noble gas monitor calibration or counting rate response for noble gas gamma radiation $\left(\frac{cpm}{\mu Ci/cm^3}\right)$

The distribution of radioactive noble gases in gaseous releases is determined by gamma spectrum analysis of gaseous effluent samples in accord with Technical Specification Table 3.9-3. In the event the radioactive noble gas distribution is not obtainable from sample(s) taken during the current period the distribution will be obtained from the most recently available data or from Table 3-2.

If f_i represents the fraction of radionuclide i in a given effluent stream, then the quantity of radionuclide i released in a given gaseous effluent stream during counting interval j is estimated by the relation

$$Q_{ij} = Q_j \cdot f_i \qquad (17)$$

The gamma radiation dose to air offsite as a consequence of noble gas discharged from each unit can be calculated with the equation

$$D_\gamma = \frac{1}{0.8} \cdot \frac{X}{Q} \cdot A_{\gamma_{eff}} \cdot \sum_j Q_j \qquad (18)$$

where D_γ = noble gas gamma dose to air due to effluent from mixed-mode release (mrad)

$A_{\gamma_{eff}}$ = radionuclide distribution weighted (effective) factor converting time-integrated, ground-level, total activity concentration of radioactive noble gas to air dose due to gamma radiation $\left(\frac{mrad}{(\mu Ci \cdot sec)/m^3}\right)$

X/Q = atmospheric dispersion factor for a mixed-mode discharge (sec/m^3)

0.8 = a conservatism factor which, in effect, increases the estimated dose to compensate for variability in radionuclide distribution.

An effective gamma air dose factor, $A_{\gamma_{eff}}$, has been derived from noble gas radionuclide distributions in routine operational releases. Refer to Appendix D for a detailed explanation. The effective gamma air dose factor derived is

$$A_{\gamma_{eff}} = 1.4 \times 10^{-5} \left(\frac{\text{mrad}}{(\mu\text{Ci} \cdot \text{sec})/\text{m}^3} \right)$$

Alternately, the gamma air dose may be calculated with the equation

$$D_\gamma = \frac{X}{Q} \cdot \sum_j \sum_i Q_j \cdot f_1 \cdot A_{\gamma i} \qquad (19)$$

where D_γ = noble gas gamma dose to air due to effluent from mixed mode release (mrad)

$A_{\gamma i}$ = factor converting time integrated, ground-level concentration of noble gas i to air dose from gamma radiation, listed in Table 3-3 $\left(\frac{\text{mrad}}{(\mu\text{Ci sec})/\text{m}^3} \right)$

(X/Q) = atmospheric dispersion factor for a mixed mode discharge (sec/m^3)

Specification 3.9.2.b.2 is satisfied by calculating the noble gas gamma radiation dose to air at the offsite location identified in Figure 3-2. At that location, 1950 meters SSE of the Plant; the reference atmospheric dispersion factor to be used is

$$\frac{X}{Q} = 5.8 \times 10^{-7} \text{ sec/m}^3$$

3.5 Noble Gas Beta Radiation Dose Accumulated in Air

Technical Specification 3.9.2.b requires that the offsite air dose from noble gas beta radiation not exceed 10 mrad during any calendar quarter. The quantity and radionuclide distribution of radioactive noble gases discharged as airborne effluents are determined as described in §3.4 herein.

Technical Specification 3.9.2.b.1/ requires a monthly determination of whether cummulative noble gas releases cause a beta radiation dose to air offsite in excess of the limit stated in Specification 3.9.2.b. This determination can be made by using the equation

$$D_\beta = \frac{1}{0.8} \cdot \frac{X}{Q} \cdot A_{\beta_{eff}} \cdot \sum_j Q_j \qquad (20)$$

where D_β = noble gas beta dose to air due to a mixed-mode release (mrad)

$A_{\beta_{eff}}$ = radionuclide distribution weighted (effective) factor converting time-integrated, ground-level, total activity concentration of radioactive noble gas to air dose due to beta radiation $\left(\frac{\text{mrad}}{(\mu\text{Ci} \cdot \text{sec})/\text{m}^3}\right)$

X/Q = atmospheric dispersion factor for a mixed mode discharge (sec/m^3)

An effective beta air dose factor, $A_{\beta_{eff}}$, has been derived from noble gas radionuclide distributions in routine operational releases. Refer to Appendix D for a detailed explanation. The effective beta air dose factor derived is

$$A_{\beta_{eff}} = 3.4 \times 10^{-5} \left(\frac{\text{mrad}}{(\mu\text{Ci} \cdot \text{sec})/\text{m}^3}\right)$$

Alternately, Specification 3.9.2.b.1/ may be satisfied by calculating the beta radiation dose to air offsite with the equation

$$D_\beta = \frac{X}{Q} \sum_j \sum_i Q_j \cdot f_i \cdot A_{\beta_i} \qquad (21)$$

where A_{β_i} = factor converting time-integrated, ground-level concentration of noble gas radionuclide i to air dose from beta radiation, listed in Table 3-3. $\left(\frac{\text{mrad}}{(\mu\text{Ci sec})/\text{m}^3}\right)$

Specification 3.9.2.b.1 is satisfied by calculating the noble gas beta radiation dose to air at the location identified in Figure 3-2. At that location, 1950 meters SSE of the Plant, the reference atmospheric dispersion factor to be used is

$$\frac{X}{Q} = 5.8 \times 10^{-7} \text{ sec/m}^3$$

3.6 Dose Due to Iodine, Tritium, and Particulates in Gaseous Effluents

Technical Specification 3.9.2.c requires that ~~radioiodines~~ I-131, tritium, ~~I-133, and I-135~~ and radioactive material in particulate form having a half-life greater than 8.0 days, in gaseous effluents released to the area offsite cause a dose to any organ or the total body of a member of the public no more than 7.5 mrem during a calendar quarter.

Radionuclides other than noble gases in gaseous effluents that are measured by the radioactive gaseous waste sampling and analysis program described in Technical Specification Table 3.9-3 are used as the release term in dose calculations. Airborne releases are discharged either via a stack above the top of the containment building or via building vents and are treated as a mixed mode release from a single location. For each of these release combinations, samples are analyzed weekly, monthly, quarterly, or for each batch release according to Table 3.9-3.

Each sample provides a measure of the concentration of specific radionuclides, C_i, in gaseous effluent discharged at flow, F_a, during a time increment Δt. Thus, each release is quantified according to the relation

$$Q_{ik} = C_{ik} \cdot \sum_j F_{aj} \cdot \Delta t_j \qquad (22)$$

where Q_{ik} = the quantity of radionuclide i released in a given effluent stream based on analysis k (μCi)

C_{ik} = concentration of radionuclide i in gaseous effluent identified by analysis k ($\mu Ci/m^3$)

F_{aj} = effluent stream discharge rate during time increment Δt_j (m^3/sec)

Δt_j = time increment j during which radionuclide i at concentration C_{ik} is being discharged (sec)

A person may be exposed directly to an airborne concentration of radioactive material discharged in effluent and indirectly via pathways involving deposition of radioactive material onto the ground. Dose estimates account for the exposure via applicable ones of the following pathways:

1) direct radiation from noble gases
2) inhalation
3) direct radiation from ground plane deposition
4) fruits and vegetables
5) air-grass-cow-meat
6) air-grass-cow-milk

Of all these pathways, the air-grass-cow-milk pathway is by far the controlling dose contributor. Of the dose by this pathway, the radioiodines contribute essentially all of the dose, with I-131 typically contributing greater than 95%. The dose transfer factors for the radioiodines are much greater than any of the other radionuclides. The critical organ is the infant's thyroid. For this reason, the potential critical organ dose via airborne effluents can be estimated by simply determining an effective dose transfer factor for the radioiodines based on the typical radioactive effluent distribution, the grass-cow-milk-man pathway; and the infant thyroid as the receptor. Then for conservatism the total cumulative release of all radioiodines and particulates can be applied to the effective factor and a conservative estimate of the infant thyroid dose determined.

The requirement, in Specification 3.9.2.~~b.2,~~ c.1 to determine monthly whether cumulative releases have caused a total body or organ dose commitment in excess of limits in Specification 3.9.2.~~b~~ c may be met by using the following equation:

$$DM_k = \frac{3.17 \times 10^{-8}}{0.8} \cdot \frac{D}{Q} \cdot TG_{131} \cdot \sum_i Q_{ik} \qquad (23)$$

where DM_k = the dose commitment (mrem) to an infant's thyroid received from exposure via the air-grass-cow-milk pathway attributable to iodines identified in analysis k of effluent air

D/Q = relative deposition rate onto ground from a mixed mode atmospheric release (m^{-2})

TG_{131} = factor converting ground deposition of radioiodines to the dose commitment to an infant's thyroid exposed via the grass-cow-milk pathway $\frac{\text{mrem/yr}}{\mu\text{Ci}/(m^2 \cdot \text{sec})}$

3.17×10^{-8} = conversion constant (yr/sec)

0.8 = a conservative factor which, in effect, increases the estimated dose to compensate for variability in the radionuclide distribution.

When equation 24 is used to estimate the critical organ (infant's thyroid) dose commitment, the effective dose transfer factor used in the equation is

$$TG_{131} = 6.5 \times 10^{11} \frac{\text{mrem/yr}}{\mu\text{Ci}/(m^2 \cdot \text{sec})}$$

The reference data from which TG_{131} was derived are summarized in Table D-2 of Appendix D.

Alternately, the monthly determination, required by Specification 3.9.2.~~b.2~~ c., may be made by using equations 23, 24, 25 and 26.

Quarterly calculations of dose commitments due to radioiodine and radioactive particulates in effluent air for inclusion in the Annual Radiological Environmental Monitoring Report are performed using equations 23 through 26, following.

The dose commitment via exposure to airborne concentrations resulting from a release, Q_{ik}, of airborne radioactive material other than noble gas, is calculated with the equation

$$D_{ank} = 3.17 \times 10^{-8} \frac{X}{Q}_d \cdot \sum_i Q_{ik} \cdot \sum_p TA_{anip} \qquad (24)$$

The dose commitment via exposure pathways involving radionuclide deposition from the atmosphere onto vegetation or the ground is calculated with the equation

$$D_{ank} = 3.17 \times 10^{-8} \frac{D}{Q} \cdot \sum_i Q_{ik} \cdot \sum_p TG_{anip} \qquad (25)$$

where D_{ank} = the dose commitment (mrem) to organ n of a person in age group a due to radionuclide i identified in analysis k of an air effluent where the analysis is required by Technical Specification Table 3.9-3.

TA_{anip} = a factor converting airborne concentration of radionuclide i to dose commitment to organ n of a person in age group a where exposure is directly to airborne material via pathway p (inhalation, or external exposure to the plume) $\left(\frac{\text{mrem/yr}}{\mu\text{Ci/m}^3}\right)$.

TG_{anip} = factor converting ground deposition of radionuclide i to dose commitment to organ n of a person in age group a where exposure is directly or indirectly to radioactive material that has been deposited on the ground $\left[\frac{\text{mrem/yr}}{\mu\text{Ci/(m}^2 \cdot \text{sec)}}\right]$

X_d/Q = atmospheric dispersion factor for a mixed mode release, adjusted for depletion by deposition (sec/m^3)

D/Q = relative deposition rate onto ground from a mixed mode atmospheric release (m^{-2})

3.17×10^{-8} = conversion constant (yr/sec)

The concentration of tritium in vegetation is a function of the airborne concentration rather than the deposition. Thus the dose commitment from airborne H^3 via vegetation (fruit and vegetables), air-grass-cow-milk, or air-grass-cow-meat pathways is calculated with the equation

$$D_{ank} = 3.17 \times 10^{-8} \cdot \frac{X}{Q} \cdot \sum_i Q_{ik} \cdot \sum_p TA_{anip} \qquad (26)$$

where X/Q = atmospheric dispersion factor for a mixed mode release (sec/m^3)

3.17×10^{-8} = conversion constant (yr/sec)

The dose commitment via a given pathway as a result of measured discharges from a release point is accumulated with

$$D_{an} = \sum_{k} D_{ank} \qquad (27)$$

The counting index k may represent either

p, analysis of a grab sample

w, a weekly sample analysis

m, a monthly composite analysis, or

q, a quarterly composite analysis

The maximum value of each counting index is the number of analyses in a sample category. The total dose commitment during a period of time is obtained by summing the contributions via separate release points and via separate environmental pathways.

When the dose to a person due to iodine and particulates discharged as airborne effluents is calculated as required by Specification 3.9.2.~~b.2.~~ c.1 the air-grass-cow-milk is evaluated by assuming a cow on pasture 4.5 miles west of the plant. (There is no milch or meat animal within 5 miles.) At that location, reference atmospheric dispersion and deposition factors are:

$$\left(\frac{X_d}{Q}\right) = 1 \times 10^{-7} \frac{sec}{m^3} \qquad \left(\frac{D}{Q}\right) = 5 \times 10^{-10} \ m^{-2}$$

The inhalation, fruit and vegetable, and irradiation by airborne radionuclides and by deposition on the ground pathways are evaluated at the nearest garden (with residence assumed) 3.6 miles west northwest of the plant. At that location, reference atmospheric dispersion and deposition factors are:

$$\left(\frac{X}{Q}\right) = 1 \times 10^{-7} \frac{sec}{m^3}$$

$$\left(\frac{X}{Q_d}\right) = 9 \times 10^{-8} \frac{sec}{m^3}$$

$$\left(\frac{D}{Q}\right) = 5 \times 10^{-10} \ m^{-2}$$

3.7 Dose to a Person from Noble Gases

Technical Specification 3.9.2.f requires the calculation of the dose or dose commitment to a person offsite exposed to 12 consecutive months of radioactive liquid and gaseous effluents from the plant. One component of personal dose is total body irradiation by gamma rays from noble gases. Another is irradiation of skin by beta and gamma radiation from noble gases. The methods of calculating these doses are presented in sections 3.7.1 and 3.7.2.

The amount of radioactive noble gas discharges is determined in the manner described in section 3.4.

3.7.1 Gamma Dose to Total Body The gamma radiation dose to the whole body of a member of the public as a consequence of noble gas released from the Station is calculated with the equation:

$$D_\gamma = \sum_i \left(Q_i \cdot \frac{X}{Q} \cdot P_{\gamma i} \right) \qquad (28)$$

where D_γ = noble gas gamma dose to total body (mrem)

$P_{\gamma i}$ = factor converting time integrated, ground level concentration of noble gas nuclide i to air dose from gamma radiation listed in Table 3-4 $\left(\frac{\text{mrem}}{(\mu\text{Ci sec})/\text{m}^3} \right)$

When the total body dose due to gamma radiation from noble gas required by Technical Specification 3.9.2.f.2 or 6.9.4.6 is calculated, the most exposed receptor is located 3.6 miles west northwest of the plant where the reference meteorological dispersion factor, X/Q, is 1×10^{-7} sec/m^3.

3.7.2 Dose to Skin The beta radiation dose to the skin of a member of the public due to beta radiation from noble gas released from the Plant may be calculated with the equation

$$D_\beta = \sum_i S_{\beta i} \left(Q_i \cdot \frac{X}{Q} \right) \quad (29)$$

where D_β = noble gas beta dose to skin (mrem)

$S_{\beta i}$ = factor converting time integrated ground level concentration of noble gas to skin dose from beta radiation listed in Table 3-4 $\left(\frac{\text{mrem}}{\mu\text{Ci} \frac{\text{sec}}{\text{m}^3}} \right)$

When the skin beta dose due to noble gas required by Specification 6.9.4.b is calculated, the most exposed receptor is located 3.6 miles west northwest of the Plant where the reference meteorological dispersion factor, X/Q, is $1.\times 10^{-7}$ sec/m^3.

The total dose to the skin from noble gases is approximately equal to the beta radiation dose to the skin plus the gamma radiation dose to the total body.

3.8 Projected Doses due to Gaseous Effluent

Technical Specification 3.9.2.e.1 requires doses due to radioactive material released in gaseous effluent to be projected over a quarter at least once a month in order to guide plant personnel in operating the radwaste systems. This requirement is satisfied by calculating doses according to equations for

1. dose to air offsite due to noble gas gamma radiation as in § 3.4,
2. dose to air offsite due to noble gas beta radiation as in § 3.5, and
3. maximally exposed organ doses due to gaseous effluents* other than noble gases as in § 3.6.

*Radioactive ~~iodine~~ & I-131, ~~I-133, and I-135~~ tritium, and radioactive material in particulate form having a half-life greater than 8.0 days.

The radioactive releases are measured according to Technical Specification Table 3.9-3 and/or derived as outlined in ODCM sections 3.4 and 3.6.

The dose from gaseous effluents is projected by extrapolating the dose commitment to date during the current quarter to include the entire quarter. If an analysis for a given radionuclide scheduled in Table 3.9-3 has not been made during the current quarter, the concentration measured in the most recently analyzed sample is assumed to be representative of releases during the current quarter.

With these possible modifications in source term data, the beta dose to air is projected with

$$P_\beta = \frac{91}{X} D_\beta \qquad (30)$$

where 91 = number of days in the quarter

X = the number of days to date in the current quarter

D_β = beta dose to air (mrad) per § 3.5

The gamma dose to air is projected with

$$P_\gamma = \frac{91}{X} D_\gamma \qquad (31)$$

where D_γ = gamma dose to air (mrad) per § 3.4.

The projected thyroid dose due to iodine and particulates in gaseous effluents is calculated with the relation

$$P_{an} = \frac{91}{X} \sum_k DM_k$$

where DM_k = the dose commitment to the infant's thyroid due to releases represented by analysis k (mrem).

Alternately, the projected personal dose to age group a and organ n, P_{an}, due to iodine and particulates in gaseous effluents may be calculated with the relation

$$P_{an} = \frac{91}{X} \sum_k D_{ank} \qquad (32)$$

where D_{ank} = the dose commitment to organ n of age group a of the maximally exposed person due to releases represented by analysis k (mrem) per § 3.6.

4.0 Dose Commitment from Releases over Extended Time

4.1 Releases during a Quarter

Technical Specification 6.9.4.a.1 requires an assessment of radiation doses arising from liquid and gaseous effluents from the plant during each calendar quarter. The assessment includes the following calculations of dose as described by equations for

1. total body and maximally exposed organ doses due to liquid effluent via eating fish and shellfish taken from the cooling canals as in § 2.4
2. total body dose due to noble gas γ as in § 3.7.1
3. skin dose due to noble gas β as in § 3.7.2
4. total body and maximally exposed organ doses due to gaseous effluents* other than noble gases as in § 3.6
5. doses to air offsite due to noble gas γ as in § 3.4 and due to noble gas β as in § 3.5.

The dose calculations are based on liquid and gaseous effluents from the Plant during each calendar quarter determined in accord with Technical Specification Tables 3.9-1 and 3.9-3.

Aqueous concentration is estimated according to § 2.2 on the basis of quarterly averaged discharge flow. Quarterly averaged meteorological conditions concurrent with the quarterly gaseous release being evaluated are used to estimate atmospheric dispersion and deposition.

The receptor of the dose is described such that the dose to any resident near the Plant is unlikely to be underestimated. That is, the receptor is selected on the basis of the combination of applicable pathways of exposure to gaseous effluent identified in the annual land use census and maximum ground level X/Q at the residence. Conditions (i.e. location, X/Q, and/or pathways) more conservative (i.e. expected to yield higher calculated doses) than appropriate for the maximally exposed individual may be assumed in the dose assessment. Seasonal appropriateness of exposure pathways may also be accounted for.

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*Radioactive ~~iodine~~ (I-131, ~~I-133, and I-135~~), ~~and I-135~~, tritium, and radioactive material in particulate form having a half-life greater than 8.0 days.

Environmental pathway-to-dose tranfers factors used in the dose calculations are provided in Appendix A of this ODCM.

4.2 Releases during 12 Months

The regulation governing the maximum allowable dose or dose commitment to a member of the public from all uranium fuel cycle sources of radiation and radioactive material in the environment is stated in 40 CFR Part 190. It requires that the dose or dose commitment to a member of the public from all sources not exceed 25 mrem/yr to any organ or 75 mrem/yr to the thyroid.

Fuel cycle sources or nuclear power reactors other than the Turkey Point Plant itself do not measurably or significantly increase the radioactivity concentration in the vicinity of the Plant; therefore, only radiation and radioactivity in the environment attributable to the Plant itself are considered in the assessment of compliance with 40 CFR Part 190.

Evaluations of dose due to liquid and gaseous effluent required by Technical Specification 3.9.2.f to assess compliance are calculated as described by the equations for:

1. total body and maximally exposed organ doses due to liquid effluent via eating fish and shellfish taken from the cooling canals as in § 2.4
2. total body dose due to noble gas γ as in § 3.7.1
3. skin dose due to noble gas β as in § 3.7.2
4. total body and maximally exposed organ doses due to gaseous effluents* other than noble gases as in § 3.6.

The doses are calculated on the basis of liquid and gaseous effluents from the Plant during 12 consecutive months, determined in accord with Technical Specification Tables 3.9-1 and 3.9-3. For the purpose of the Annual Radiological Environmental Report, doses are based upon releases during a calendar year.

*Radioactive ~~iodine~~ (I-131, ~~I-133, and I-135~~), tritium and radioactive material in particulate form having a half-life greater than 8.0 days.

Aqueous radioactive material concentrations are estimated according to § 2.2 on the basis of annual averaged discharge flow. Annual averaged meteorological conditions concurrent with annual gaseous releases being evaluated are used to estimate atmospheric dispersion, deposition, and plume gamma exposure.

The receptor of the dose is described such that the dose to any resident near the Plant is not likely to be underestimated. The receptor is selected on the basis of the combination of applicable pathways of exposure to gaseous effluent identified in the annual land use census and maximum ground level X/Q at the residence. Conditions more conservative than appropriate for the maximally exposed person may be assumed in the dose assessment.

Environmental pathway-to-dose transfer factors used in the dose calculations appear in Appendix A.

Table 3-1

Atmospheric Gaseous Release Points
at the Turkey Point Units 3 and 4

Effluent Source	Release Point
Gas decay tanks	Plant vent
Radwaste Building	Plant vent
Auxiliary Building	Plant vent
Containment Purge	Plant vent
No. 4 spent fuel pit	Plant vent
No. 3 spent fuel pit	Spent fuel pit vent
Air ejectors	Turbine deck
Steam generator blowdown	Blowdown vent

Table 3-2

Distribution of Radioactive Noble Gases
in Gaseous Effluent from Turkey Point Units 3 & 4

Nuclide	Release fraction[a,b]
Ar-41	9.2E-3
Kr-83m	--
Kr-85m	2.5E-4
Kr-85	2.5E-4
Kr-87	1.6E-4
Kr-88	2.1E-4
Xe-131m	4.4E-4
Xe-133m	1.2E-3
Xe-133	0.99
Xe-135m	8.0E-4
Xe-135	3.4E-3
Xe-137	--
Xe-138	3.7E-4

---

[a] Based on measured discharge from Turkey Point Units 3 & 4 during 1978 thru 1980.

[b] To estimate radionuclide concentrations in a sample in which only the total activity concentration has been measured, multiply the total activity concentration by the fraction of respective radionuclides listed here.

Table 3-3

Transfer Factors for Maximum Offsite Air Dose

Radionuclide	Air Dose Transfer Factors	
	$A_{\gamma i}$ $\left(\frac{\text{mrad}}{\mu\text{Ci sec/m}^3}\right)$	$A_{\beta i}$ $\left(\frac{\text{mrad}}{\mu\text{Ci sec/m}^3}\right)$
Kr-83m	6.1E-7	9.1E-6
Kr-85m	3.9E-5	6.2E-5
Kr-85	5.4E-7	6.2E-5
Kr-87	2.0E-4	3.3E-4
Kr-88	4.8E-4	9.3E-5
Kr-89	5.5E-4	3.4E-4
Kr-90	5.2E-4	2.5E-4
Xe-131m	4.9E-6	3.5E-5
Xe-133m	1.0E-5	4.7E-5
Xe-133	1.1E-5	3.3E-5
Xe-135m	1.1E-4	2.3E-5
Xe-135	6.1E-5	7.8E-5
Xe-137	4.8E-5	4.0E-4
Xe-138	2.9E-4	1.5E-4
Ar-41	2.9E-4	1.0E-4

Ref: Regulatory Guide 1.109, Revision 1, Table B-1

Table 3-4

Transfer Factors for Maximum Dose to a Person Offsite due to Radioactive Noble Gases

Radionuclide	Dose Transfer Factors	
	$P_{\gamma i}$ $\left(\frac{\text{mrem}}{\mu\text{Ci sec/m}^3}\right)$	$S_{\beta i}$ $\left(\frac{\text{mrem}}{\mu\text{Ci sec/m}^3}\right)$
Kr-83m	2.4E-9	—
Kr-85m	3.7E-5	4.6E-5
Kr-85	5.1E-7	4.2E-5
Kr-87	1.9E-4	3.1E-4
Kr-88	4.7E-4	7.5E-5
Kr-89	5.3E-4	3.2E-4
Kr-90	4.9E-4	2.3E-4
Xe-131m	2.9E-6	1.5E-5
Xe-133m	8.0E-6	3.1E-5
Xe-133	9.3E-6	9.7E-6
Xe-135m	9.9E-5	2.3E-5
Xe-135	5.7E-5	5.9E-5
Xe-137	4.5E-5	3.9E-4
Xe-138	2.8E-4	1.3E-4
Ar-41	2.8E-4	8.5E-5

Ref: Regulatory Guide 1.109, Revision 1, Table B-1.

Table 3.5

REFERENCE METEOROLOGY
ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

$$\frac{X}{Q} \quad \frac{sec}{m^3}$$

X/Q are annual averaged factors of atmospheric dispersion of a mixed mode gaseous release from the Turkey Point Plant.
Period of record: 01/01/76 to 12/31/77

BASE DISTANCE IN MILES / KILOMETERS

AFTD SECT	DESIGN DIST MI	.25 .40	.75 1.21	1.50 2.41	2.50 4.02	3.50 5.63	4.50 7.24	5.50 8.85	7.00 11.26
NNE	0.	8.9E-07	1.9E-07	8.3E-08	5.0E-08	3.0E-08	2.2E-08	1.9E-08	1.4E-08
NE	0.	6.9E-07	1.5E-07	6.3E-08	3.8E-08	2.5E-08	2.1E-08	1.3E-08	1.0E-08
ENE	0.	8.4E-07	1.4E-07	7.5E-08	3.9E-08	2.8E-08	2.3E-08	1.8E-08	1.3E-08
E	0.	8.6E-07	1.9E-07	9.1E-08	5.1E-08	3.6E-08	2.7E-08	2.2E-08	1.7E-08
ESE	0.	6.6E-07	1.5E-07	7.9E-08	4.5E-08	2.9E-08	2.3E-08	1.9E-08	1.2E-08
SE	0.	1.6E-06	2.8E-07	1.1E-07	6.1E-08	4.2E-08	3.0E-08	2.6E-08	2.1E-08
SSE	0.	4.9E-06	9.2E-07	3.6E-07	1.8E-07	1.1E-07	9.0E-08	7.1E-08	4.9E-08
S	0.	2.9E-06	4.6E-07	1.8E-07	1.0E-07	7.8E-08	5.4E-08	4.6E-08	3.3E-08
SSW	0.	6.5E-07	1.6E-07	6.5E-08	4.6E-08	2.4E-08	2.6E-08	1.8E-08	1.4E-08
SW	0.	1.5E-06	3.2E-07	1.4E-07	7.9E-08	4.9E-08	3.2E-08	2.7E-08	1.9E-08
WSW	0.	2.9E-06	6.3E-07	2.3E-07	1.3E-07	7.6E-08	5.5E-08	4.2E-08	3.1E-08
W	0.	6.3E-06	1.3E-06	5.2E-07	2.6E-07	1.7E-07	1.2E-07	9.2E-08	6.6E-08
WNW	0.	4.1E-06	8.7E-07	3.4E-07	1.7E-07	1.2E-07	8.1E-08	6.3E-08	4.2E-08
NW	0.	2.7E-06	6.0E-07	2.4E-07	1.2E-07	7.6E-08	5.1E-08	4.3E-08	3.2E-08
NNW	0.	1.4E-06	2.9E-07	1.2E-07	6.8E-08	4.5E-08	3.0E-08	2.4E-08	1.5E-08
N	0.	9.5E-07	2.1E-07	8.5E-08	4.5E-08	3.2E-08	2.2E-08	1.7E-08	1.3E-08

BASE DISTANCE IN MILES / KILOMETERS

AFTD SECT	DESIGN DIST MI	9.00 14.48	11.00 17.70	.79 1.27	5.00 8.04	1.00 1.61	2.00 3.22	2.75 4.42	4.30 6.92
NNE	0.	9.8E-09	6.6E-09	1.8E-07	2.0E-08	1.4E-07	6.2E-08	4.4E-08	2.3E-08
NE	0.	7.3E-09	5.4E-09	1.5E-07	1.6E-08	1.1E-07	4.8E-08	3.5E-08	2.1E-08
ENE	0.	1.1E-08	7.4E-09	1.4E-07	2.0E-08	1.0E-07	5.2E-08	3.6E-08	2.4E-08
E	0.	1.3E-08	9.8E-09	1.7E-07	2.4E-08	1.3E-07	6.3E-08	4.6E-08	2.8E-08
ESE	0.	1.1E-08	9.6E-09	1.4E-07	2.0E-08	1.2E-07	5.7E-08	4.0E-08	2.4E-08
SE	0.	1.5E-08	1.3E-08	2.7E-07	2.7E-08	1.9E-07	7.8E-08	5.5E-08	3.1E-08
SSE	0.	3.5E-08	2.7E-08	8.7E-07	7.9E-08	6.3E-07	2.5E-07	1.6E-07	9.4E-08
S	0.	2.3E-08	1.8E-08	4.2E-07	5.0E-08	3.1E-07	1.3E-07	9.5E-08	5.8E-08
SSW	0.	9.4E-09	7.1E-09	1.9E-07	2.1E-08	1.1E-07	5.4E-08	3.8E-08	2.5E-08
SW	0.	1.4E-08	1.0E-08	3.0E-07	2.9E-08	2.3E-07	1.0E-07	6.9E-08	3.5E-08
WSW	0.	2.2E-08	1.8E-08	5.9E-07	4.8E-08	4.3E-07	1.7E-07	1.0E-07	5.2E-08
W	0.	4.5E-08	3.5E-08	1.2E-06	1.0E-07	9.0E-07	3.5E-07	2.3E-07	1.3E-07
WNW	0.	2.9E-08	2.3E-08	8.1E-07	7.1E-08	5.9E-07	2.3E-07	1.6E-07	8.6E-08
NW	0.	2.0E-08	1.5E-08	5.6E-07	4.7E-08	4.1E-07	1.6E-07	1.0E-07	5.6E-08
NNW	0.	1.0E-08	8.3E-09	2.7E-07	2.6E-08	2.0E-07	9.1E-08	6.1E-08	3.2E-08
N	0.	1.0E-08	7.2E-09	1.9E-07	2.0E-08	1.5E-07	5.9E-08	4.0E-08	2.3E-08

NUMBER OF VALID OBSERVATIONS = 16538
NUMBER OF INVALID OBSERVATIONS = 1006
NUMBER OF CALMS LOWER LEVEL = 195
NUMBER OF CALMS UPPER LEVEL = 383

Table 3.6

REFERENCE METEOROLOGY
DEPOSITION DEPLETED ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

$$\frac{X}{Q}_d \quad \frac{sec}{m^3}$$

X_d/Q are annual averaged factors of atmospheric dispersion of a mixed mode release from the Turkey Point Plant which have been corrected for depletion from the plume by fallout and deposition.
Period of Record: 01/01/76 to 12/31/77

BASE DISTANCE IN MILES / KILOMETERS

AFTD SECT	DESIGN DIST MI	.25 .40	.75 1.21	1.50 2.41	2.50 4.02	3.50 5.63	4.50 7.24	5.50 8.85	7.00 11.26
NNE	0.	8.7E-07	1.7E-07	7.3E-08	4.4E-08	2.7E-08	1.9E-08	1.6E-08	1.2E-08
NE	0.	6.9E-07	1.4E-07	5.5E-08	3.3E-08	2.2E-08	1.7E-08	1.2E-08	8.8E-09
ENE	0.	8.0E-07	1.2E-07	6.5E-08	3.4E-08	2.4E-08	2.0E-08	1.6E-08	1.2E-08
E	0.	8.6E-07	1.7E-07	7.6E-08	4.4E-08	3.1E-08	2.4E-08	1.9E-08	1.5E-08
ESE	0.	6.1E-07	1.3E-07	6.9E-08	3.9E-08	2.5E-08	2.0E-08	1.6E-08	1.1E-08
SE	0.	1.5E-06	2.6E-07	9.5E-08	5.2E-08	3.4E-08	2.4E-08	2.1E-08	1.7E-08
SSE	0.	4.7E-06	8.2E-07	3.1E-07	1.5E-07	9.2E-08	7.4E-08	5.8E-08	3.8E-08
S	0.	2.6E-06	4.2E-07	1.5E-07	8.5E-08	6.4E-08	4.4E-08	3.7E-08	2.6E-08
SSW	0.	6.1E-07	1.4E-07	5.6E-08	3.9E-08	2.0E-08	2.2E-08	1.5E-08	1.2E-08
SW	0.	1.3E-06	2.8E-07	1.3E-07	6.7E-08	4.2E-08	2.7E-08	2.3E-08	1.5E-08
WSW	0.	2.7E-06	5.6E-07	2.1E-07	1.0E-07	6.4E-08	4.6E-08	3.5E-08	2.6E-08
W	0.	5.9E-06	1.2E-06	4.4E-07	2.2E-07	1.4E-07	9.9E-08	7.6E-08	5.4E-08
WNW	0.	3.8E-06	7.7E-07	2.9E-07	1.5E-07	9.8E-08	7.0E-08	5.4E-08	3.6E-08
NW	0.	2.5E-06	5.4E-07	2.1E-07	1.1E-07	6.8E-08	4.5E-08	3.8E-08	2.8E-08
NNW	0.	1.4E-06	2.6E-07	1.1E-07	6.0E-08	4.0E-08	2.6E-08	2.0E-08	1.3E-08
N	0.	8.8E-07	1.9E-07	7.8E-08	3.9E-08	2.8E-08	1.9E-08	1.5E-08	1.1E-08

BASE DISTANCE IN MILES / KILOMETERS

AFTD SECT	DESIGN DIST MI	9.00 14.48	11.00 17.70	.79 1.27	5.00 8.04	1.00 1.61	2.00 3.22	2.75 4.42	4.30 6.92
NNE	0.	8.5E-09	6.0E-09	1.6E-07	1.8E-08	1.2E-07	5.5E-08	3.8E-08	2.1E-08
NE	0.	6.3E-09	4.5E-09	1.3E-07	1.4E-08	9.4E-08	4.2E-08	3.0E-08	1.8E-08
ENE	0.	9.0E-09	6.7E-09	1.2E-07	1.8E-08	9.1E-08	4.5E-08	3.1E-08	2.0E-08
E	0.	1.1E-08	7.9E-09	1.5E-07	2.1E-08	1.2E-07	5.5E-08	3.9E-08	2.4E-08
ESE	0.	8.8E-09	8.3E-09	1.3E-07	1.8E-08	1.0E-07	5.0E-08	3.4E-08	2.0E-08
SE	0.	1.3E-08	1.0E-08	2.4E-07	2.3E-08	1.7E-07	6.7E-08	4.7E-08	2.6E-08
SSE	0.	2.7E-08	2.1E-08	7.7E-07	6.4E-08	5.6E-07	2.2E-07	1.3E-07	7.7E-08
S	0.	1.9E-08	1.3E-08	3.8E-07	4.1E-08	2.7E-07	1.1E-07	7.8E-08	4.8E-08
SSW	0.	7.9E-09	5.7E-09	1.4E-07	1.8E-08	9.6E-08	4.7E-08	3.2E-08	2.2E-08
SW	0.	1.1E-08	8.6E-09	2.7E-07	2.4E-08	2.0E-07	9.1E-08	5.9E-08	2.9E-08
WSW	0.	1.8E-08	1.4E-08	5.2E-07	4.9E-08	3.8E-07	1.4E-07	8.7E-08	4.8E-08
W	0.	3.7E-08	2.8E-08	1.1E-06	8.6E-08	7.9E-07	3.1E-07	2.0E-07	1.0E-07
WNW	0.	2.5E-08	2.0E-08	7.3E-07	6.1E-08	5.1E-07	2.0E-07	1.4E-07	7.4E-08
NW	0.	1.8E-08	1.3E-08	5.1E-07	4.1E-08	3.6E-07	1.4E-07	8.9E-08	5.0E-08
NNW	0.	9.1E-09	6.9E-09	2.4E-07	2.3E-08	1.8E-07	7.7E-08	5.4E-08	2.8E-08
N	0.	8.7E-09	6.3E-09	1.8E-07	1.7E-08	1.3E-07	5.2E-08	3.5E-08	2.0E-08

NUMBER OF VALID OBSERVATIONS = 16538
NUMBER OF INVALID OBSERVATIONS = 1006
NUMBER OF CALMS LOWER LEVEL = 195
NUMBER OF CALMS UPPER LEVEL = 383

Table 3.7

REFERENCE METEOROLOGY
ANNUAL AVERAGED RELATIVE DEPOSITION RATE

$$\frac{D}{Q} \quad \frac{1}{m^2}$$

D/Q are annual averaged factors representing the fraction of a mixed mode airborne release from the Turkey Point Plant which is deposited on a square meter area of land at a given distance and direction from the Plant.
Period of Record: 01/01/76 to 12/31/77

BASE DISTANCE IN MILES / KILOMETERS

AFTD SECT	DESIGN DIST MI	.25 .40	.75 1.21	1.50 2.41	2.50 4.02	3.50 5.63	4.50 7.24	5.50 8.85	7.00 11.26
NNE	0.	6.4E-09	1.5E-09	4.7E-10	2.0E-10	9.1E-11	5.5E-11	4.1E-11	2.7E-11
NE	0.	3.5E-09	8.7E-10	2.8E-10	1.2E-10	6.4E-11	4.3E-11	2.5E-11	1.7E-11
ENE	0.	2.8E-09	5.1E-10	2.1E-10	7.6E-11	4.1E-11	2.9E-11	1.9E-11	1.2E-11
E	0.	2.7E-09	6.6E-10	2.4E-10	1.1E-10	5.8E-11	3.7E-11	2.5E-11	1.6E-11
ESE	0.	1.6E-09	4.2E-10	1.9E-10	7.7E-11	4.0E-11	2.7E-11	1.8E-11	1.0E-11
SE	0.	5.3E-09	1.2E-09	3.7E-10	1.6E-10	9.0E-11	5.4E-11	4.2E-11	2.9E-11
SSE	0.	2.6E-08	5.2E-09	1.8E-09	6.8E-10	3.5E-10	2.5E-10	1.8E-10	1.0E-10
S	0.	1.2E-08	2.1E-09	6.7E-10	3.0E-10	2.0E-10	1.2E-10	9.1E-11	5.8E-11
SSW	0.	2.3E-09	7.2E-10	2.4E-10	1.2E-10	5.3E-11	4.0E-11	2.8E-11	2.0E-11
SW	0.	1.1E-08	2.7E-09	1.0E-09	4.3E-10	2.3E-10	1.2E-10	9.6E-11	5.5E-11
WSW	0.	2.3E-08	5.0E-09	1.5E-09	6.1E-10	3.2E-10	2.0E-10	1.4E-10	8.5E-11
W	0.	5.7E-08	1.2E-08	3.9E-09	1.4E-09	7.6E-10	4.9E-10	3.3E-10	2.1E-10
WNW	0.	4.1E-08	9.6E-09	2.7E-09	1.0E-09	5.7E-10	3.4E-10	2.4E-10	1.4E-10
NW	0.	2.4E-08	6.2E-09	1.7E-09	6.1E-10	3.1E-10	1.8E-10	1.3E-10	8.5E-11
NNW	0.	1.2E-08	3.0E-09	9.5E-10	3.6E-10	2.0E-10	1.1E-10	7.5E-11	4.2E-11
N	0.	5.8E-09	1.6E-09	4.8E-10	1.8E-10	9.6E-11	5.8E-11	4.0E-11	2.5E-11

BASE DISTANCE IN MILES / KILOMETERS

AFTD SECT	DESIGN DIST MI	9.00 14.48	11.00 17.70	.79 1.27	5.00 8.04	1.00 1.61	2.00 3.22	2.75 4.42	4.30 6.92
NNE	0.	1.6E-11	9.3E-12	1.4E-09	4.7E-11	9.6E-10	2.8E-10	1.6E-10	6.2E-11
NE	0.	9.9E-12	6.2E-12	8.1E-10	3.2E-11	5.6E-10	1.8E-10	1.1E-10	4.6E-11
ENE	0.	8.1E-12	5.2E-12	5.0E-10	2.3E-11	3.6E-10	1.2E-10	6.4E-11	3.0E-11
E	0.	1.0E-11	6.6E-12	5.9E-10	3.0E-11	4.3E-10	1.5E-10	8.8E-11	3.9E-11
ESE	0.	7.5E-12	5.8E-12	4.1E-10	2.2E-11	3.1E-10	1.2E-10	6.5E-11	2.8E-11
SE	0.	1.8E-11	1.3E-11	1.1E-09	4.7E-11	7.1E-10	2.3E-10	1.3E-10	6.0E-11
SSE	0.	6.6E-11	4.5E-11	4.9E-09	2.1E-10	3.4E-09	1.1E-09	5.8E-10	2.6E-10
S	0.	3.4E-11	2.3E-11	1.9E-09	1.0E-10	1.4E-09	4.4E-10	2.7E-10	1.3E-10
SSW	0.	1.0E-11	6.6E-12	6.7E-10	3.6E-11	4.5E-10	1.7E-10	9.7E-11	4.8E-11
SW	0.	3.5E-11	2.2E-11	2.5E-09	1.1E-10	1.9E-09	6.3E-10	3.6E-10	1.4E-10
WSW	0.	5.5E-11	3.8E-11	4.6E-09	1.6E-10	3.2E-09	9.7E-10	4.9E-10	2.2E-10
W	0.	1.2E-10	8.7E-11	1.1E-08	3.9E-10	7.4E-09	2.2E-09	1.2E-09	5.0E-10
WNW	0.	8.8E-11	6.1E-11	8.7E-09	2.6E-10	5.7E-09	1.6E-09	9.0E-10	3.8E-10
NW	0.	4.5E-11	3.2E-11	5.6E-09	1.5E-10	3.7E-09	9.5E-10	5.0E-10	2.0E-10
NNW	0.	2.5E-11	1.8E-11	2.7E-09	8.8E-11	1.8E-09	5.4E-10	3.0E-10	1.2E-10
N	0.	1.7E-11	1.1E-11	1.4E-09	4.8E-11	1.0E-09	2.7E-10	1.5E-10	6.5E-11

NUMBER OF VALID OBSERVATIONS = 16538
NUMBER OF INVALID OBSERVATIONS = 1006
NUMBER OF CALMS LOWER LEVEL = 195
NUMBER OF CALMS UPPER LEVEL = 383

APPENDIX A
PATHWAY-DOSE TRANSFER FACTORS

Environmental pathway transfer factors, usage factors, and dose commitment factors appropriate for each exposure pathway, age, and organ are combined into integrated environmental concentration-to-dose factors for each radionuclide. This appendix includes tables of values of the transfer factors calculated in accord with equations and values recommended in NUREG-0133[1] for individual environmental pathways. In the event a single, composite transfer factor is desired for a given organ and age group, it can be obtained by summing the factors for appropriate pathways. Appropriate transfer factors from Appendix A are used in performing dose assessment calculations prescribed in the ODCM.

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[1] J. Boegli, et al., eds., 1978, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, NUREG-0133, USNRC, Office Nuclear Reactor Regulation.

Appendix A Tables
are the same as in the
August 14, 1980 version

APPENDIX B

Nuclide Dilution in the Cooling Canals

The effective concentration of each nuclide in the cooling canals is a function of the discharge flow and concentration, the nuclide decay constant, and the various diluting flows in and out of the canal. Reference 1 was a study of the hydraulics of the Turkey Point cooling canals; it revealed that ground water interchange is the most significant means by which net removal of chemical constituents from the canals occurs. One result of the study documented in Reference 1 is a mathematical model of the water and chemical balance in the canals. That model is used here to derive an expression for the nuclide concentrations to which fish and shellfish in the canal are exposed. This expression can then be used to develop the expression in Section 2.4 for calculating accumulated personal dose due to ingestion of fish and shellfish.

Assuming that sampling of nuclide i in the canal by seafood (and, in turn, the ultimate receptors) occurs randomly in space and time, the canal can be treated as a homogeneous mixture. This is certainly true for nuclides with effective lives greater than several days, since the canal average transit time is on the order of two days. For shorter-lived nuclides, this assumption is conservative. (See also Figure 12 of Reference 2.)

A mass balance on the canal yields:

$$V\, dC_i(t)/dt = -V\lambda_i C_i(t) - F_3 C_i(t) + F_1 C_{ik}. \qquad (B1)$$

where V = is the canal volume (ft^3)

λ_i = is the nuclide i decay constant (min^{-1})

$C_i(t)$ = is the average nuclide i concentration in the canal at time t ($\mu c/ml$),

F_3 = is the ground water interchange flow (ft^3/min)

F_1 = flow in radioactive liquid discharge line (ft^3/min)

C_{ik} = radionuclide i concentration in undiluted liquid waste being discharged ($\mu Ci/ml$)

Defining $C_i(t) = C_i^o$ when t = o at the beginning of the discharge, it can be shown that:

$$C_i(t) = C_i^o - (C_i^s - C_i^o) \exp \left[- (\lambda_i + F_3/V)\ t \right] \qquad (B2)$$

where the steady-state concentration

$$C_i^s = F_1\ C_{ik}/(\lambda_i V + F_3) \text{ or} \qquad (B3)$$

$$C_i^s = F_1\ C_{ik}/V\lambda_i^e \qquad (B4)$$

Examination of the above equations reveals that the maximum possible concentration of nuclide i is C_i^s. For continous discharges, C_i^s is approximately equal to the "average" concentration in the cooling canal. For batch discharges this expression is conservative, particularly when the duration of the discharge is small with respect to the period between discharges and the effective decay period, $1/\lambda_i^e$.

Figure 27 of Reference 1 shows the relationship of canal water level (Canal 32 - Line 3) to volume. Figure 9 of Reference 1 shows canal levels during July - December 1978. During that period, the canal volume (V) never decreased below 500 million cubic feet - a conservative lower bound for V.

Tables 3 and 4 of Reference 1 provide an expression for F_3, the ground water interchange flow rate:

$$F_3 = C_1 + C_2 + C_3 + C_4 \ (ft^3/min) \qquad (B5)$$

where C_1 = 30 900 ft^3/min.

$C_2 = \pm$ 1680 neap/spring

C_3 = -.120 June
+.360 July
+ 1320 August
+ 1200 September
- .960 October
- 1440 November

C_4 = 3000

A conservative lower bound for F_3 would be on the order of 30000 cubic feet per minute.

An upper bound on the steady-state concentration of radionuclide i in the canals is thus

$$C_i^s \lesssim \frac{F_1 \cdot C_{ik}}{\lambda_i V + F_3} = \frac{F_1 \cdot C_{ik}}{5 \times 10^8 \text{ ft}^3 \cdot \lambda_i + 30000 \text{ ft}^3/\text{min}} \qquad \text{(B6)}$$

Alternately, stating F_1 and F_3 in units, gal/min, and V in units, gal, produces

$$C_i^s \lesssim \frac{F_1 \cdot C_{ik}}{\lambda_i V + F_3} = \frac{F_1 \cdot C_{ik}}{3.75 \times 10^9 \text{ gal} \cdot \lambda_i + 2.24 \times 10^5 \text{ gal/min}}$$

or

$$C_i^s \lesssim \frac{F_1 \cdot C_{ik}}{V\left(\lambda_i + \frac{F_3}{V}\right)} = \frac{F_1 \cdot C_{ik}}{3.75 \times 10^9 \, (\lambda_i + 6 \times 10^{-5})}$$

or

$$C_i^s \lesssim \frac{F_1 \cdot C_{ik}}{V \cdot \lambda_i^e}$$

where $V = 3.75 \times 10^9$ gal

$\lambda_i^e = \lambda_i + 6 \times 10^{-5}$ min^{-1}

References

1. Evaluation of Cooling System Chemistry, Turkey Point Cooling Canal System, Dames and Moore, June 1979.

2. Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, Regulatory Guide 1.113, Revision 1, USNRC, April 1977.

APPENDIX C

Limited Analysis Dose Assessment for Liquid Radioactive Effluents

The radioactive liquid effluents for the years 1978, 1979, and 1980 were evaluated to determine the dose contribution of the radionuclide distribution. This analysis was performed to evaluate the use of a limited dose analysis for determining environmental doses. Limiting the dose calculation to a few selected radionuclides that contribute the majority of the dose provides a simplified method of determining compliance with the dose limits of Technical Specification 3.9.1.b.2.

Tables C-1 and C-2 present the results of this evaluation. Table C-1 presents the fraction of the adult total body dose contributed by the major radionuclides. Table C-2 presents the same data for the adult GI-LLI dose. The adult total body and adult GI-LLI were determined to be the limiting doses based on an evaluation of all age groups (adult, teenager, child, and infant) and all organs (bone, liver, kidney, lung, and GI-LLI). As the data in the tables show, the radionuclides Co-58, Co-60, Cs-134, and Cs-137 dominate the total body dose; the radionuclides Co-58, Co-60, Nb-95, and Ag-110m dominate the GI-LLI dose. In all but one case (1978—shellfish, GI-LLI-dose) these radionuclides contribute 90% or more of the total dose. If for 1978 the fish and shellfish pathways are combined as is done to determine the total dose, the contribution from these nuclides is 89% of the total GI-LLI dose.

Therefore, the dose commitment due to radioactive material in liquid effluents can be reasonably estimated by limiting the dose calculation to the radionuclides, Co-58, Co-60, Nb-95, Ag-110m, Cs-134, and Cs-137, which cumulatively contribute about 90% or greater of the total dose calculated by using all radionuclides detected. This limited analysis dose assessment method is a simplified calculation that provides a reasonable evaluation of doses due to liquid radioactive effluents.

Tritium is not included in the limited analysis dose assessment for liquid releases because the potential dose resulting from normal reactor releases is negligible and is essentially independent of radwaste system operation. The maximum amount of tritium released annually is about 1000 curies. At Turkey Point, 1000 Ci/yr releases to the cooling canals produces a calculated whole body dose of 0.002 mrem/yr via the fish and shellfish pathways. This amounts to less than 0.1% of the design objective dose of 3 mrem/yr. Furthermore, the release of tritium is a function of operating time and power level and is essentially unrelated to radwaste system operation.

Table C-1

Adult Total Body Dose Contributions
Fraction of Total (Excluding H-3)

Radionuclide	1978		1979		1980	
	Fish	Shellfish	Fish	Shellfish	Fish	Shellfish
Co-58	0.02	0.12	0.02	0.10	0.43	0.9
Co-60	0.11	0.57	0.12	0.56	0.23	0.68
Cs-134	0.41	0.13	0.43	0.13	0.27	0.05
Cs-137	0.44	0.14	0.40	0.12	0.43	0.08
Total	0.98	0.96	0.97	0.91	0.96	0.90

Table C-2

Adult GI-LLI Dose Contribution
Fraction of Total (Excluding H-3)

Radionuclide	1978		1979		1980	
	Fish	Shellfish	Fish	Shellfish	Fish	Shellfish
Co-58	0.07	0.15	0.05	0.13	0.03	0.10
Co-60	0.33	0.70	0.27	0.67	0.14	0.69
Nb-95	0.54	<0.01	0.13	<0.01	0.56	<0.01
Ag-110m	--	--	0.48	0.12	0.24	0.12
Total	0.94	0.85	0.93	0.92	0.97	0.91

APPENDIX D

Technical Bases for A_{eff}

Overview

The evaluation of doses due to releases of radioactive material to the atmosphere can be simplified by the use of effective dose transfer factors instead of using dose factors which are radionuclide specific. These effective factors, which are based on the typical radionuclide distribution in the releases, can be applied to the total radioactivity released to approximate the dose in the environment, ie, instead of having to sum the isotopic distribution multiplied by the isotope specific dose factor only a single multiplication (A_{eff} times the total quantity of radioactive material released) would be needed. This approach provides a reasonable estimate of the actual dose while eliminating the need for a detailed calculational technique.

Determination of A_{eff}

The effective dose transfer factor is based on past operating data. The radioactive effluent distribution for the past years can be used to derive a single effective factor by the following equation.

$$A_{eff} = \sum_i A_i \cdot f_i \qquad \text{(D-1)}$$

where A_{eff} = the effective dose transfer factor

A_i = the dose transfer factor for radionuclide i

f_i = the fractional abundance of radionuclide i in the radioactive effluents

This equation yields a single dose factor, weighted by the typical radionuclide distribution.

To determine the appropriate effective factor to be used and to evaluate the degree of variability, the atmospheric radioactive effluents for the past 3 years have been evaluated. An effective dose transfer factor has been determined for the gaseous effluents for all pathways of interest. Tables D-1 and D-2 present the results of this evaluation.

For the radioiodines and particulates with half-lives greater than 8 days, the effective dose transfer factor is based solely on the radioiodines (I-131, 133, and 135). This approach was selected because the radioiodines contribute essentially all of the dose to the infant's thyroid via the cow-milk pathway. The infant's thyroid and the cow-milk pathway are the critical organ and controlling pathway, respectively, for the releases of radioiodine and particulates. All other particulates contribute less than 1% of the dose. The effective dose transfer factor is determined by applying equation D-1 to the radioiodines. However, in determining the dose, this effective dose transfer factor should be applied to the total release of all radioiodines and to particulates with half lives greater than 8 days. This uniform application is conservative in providing reasonable assurance that the actual dose will not be underestimated by the use of this simplified method.

The determination of A_{eff} was limited to the past three years (1978, 1979, and 1980) because of the changes that have occurred in the waste processing system. A demineralizer system replaced the previously used evaporator in the liquid waste processing system.

As can be seen from Tables D-1 and D-2, the effective dose transfer factor varies little from year to year. The maximum observed variability from the average value is 13% for the noble gases and 25% for the radioiodines. This variability is minor considering other areas of uncertainty and conservatism inherent in the environmental dose calculational models.

To provide an additional degree of conservatism, a factor of 0.8 is introduced into the dose calculational process when the effective dose transfer factor is used. This added conservatism provides additional assurance that the evaluation of doses by the use of a single effective factor will not significantly underestimate any actual doses in the environment.

Reevaluation

The doses due to the gaseous effluents are evaluated by the more detailed calculational methods (ie, use of nuclide specific dose factors) on a yearly bases. At this time a comparison can be made between the simplified method and the detailed method to assure the overall reasonableness of this limited analysis approach. If this comparison indicates that the radionuclide distribution has changed significantly causing the simplified method to underestimate the doses by more than 20%, the value of A_{eff} will need to be reexamined to assure the overall acceptability of this approach. However, this reexamination will only be needed if the doses as calculated by the detailed analysis exceed 50% of the design bases doses (ie, greater than 5 mrads gamma air dose, 10 mrads beta air dose, or 7.5 mrem infant thyroid dose).

In any case, the appropriateness of the A_{eff} value will be periodically evaluated to assume the applicability of a single effective dose factor for evaluating environmental doses.

Table D-1

Effective Dose Transfer Factors

Noble Gases—Air Dose

Year	$A_{\gamma_{eff}}$ mrad / $\mu Ci \cdot sec/m^3$	$A_{\beta_{eff}}$ mrad / $\mu Ci \cdot sec/m^3$
1978	1.3×10^{-5}	3.4×10^{-5}
1979	1.3×10^{-5}	3.4×10^{-5}
1980	1.6×10^{-5}	3.4×10^{-5}
Average	1.4×10^{-5}	3.4×10^{-5}

Table D-2

Effective Dose Transfer Factor for Air-Grass-Cow-Milk-Infant-Thyroid Pathway

Radionuclide	Annual Airborne Release (Ci)	Fraction	Dose Factor[a] $\left(\frac{\text{mrem/yr}}{\mu\text{Ci}/(\text{m}^2 \cdot \text{sec})}\right)$	Weighted Dose Factor $\left(\frac{\text{mrem/yr}}{\mu\text{Ci}/(\text{m}^2 \cdot \text{sec})}\right)$
		Year 1978		
I-131	0.381	0.688	9.9E11	
I-133	0.129	0.233	1.3E10	6.9E11
I-135	0.044	0.079	5.2E6	
		Year 1979		
I-131	0.0188	0.520	9.9E11	
I-133	0.0156	0.432	1.3E10	5.2E11
I-135	0.0018	0.048	5.2E6	
		Year 1980		
I-131	0.0518	0.756	9.9E11	
I-133	0.0124	0.181	1.3E10	7.5E11
I-135	0.0043	0.063	5.2E6	
			avg[b]	6.5E11

[a] air-grass-cow-milk-infant-thyroid dose transfer factor

[b] Effective dose commitment transfer factor is the average of weighted dose transfer factor over three years.

APPENDIX F

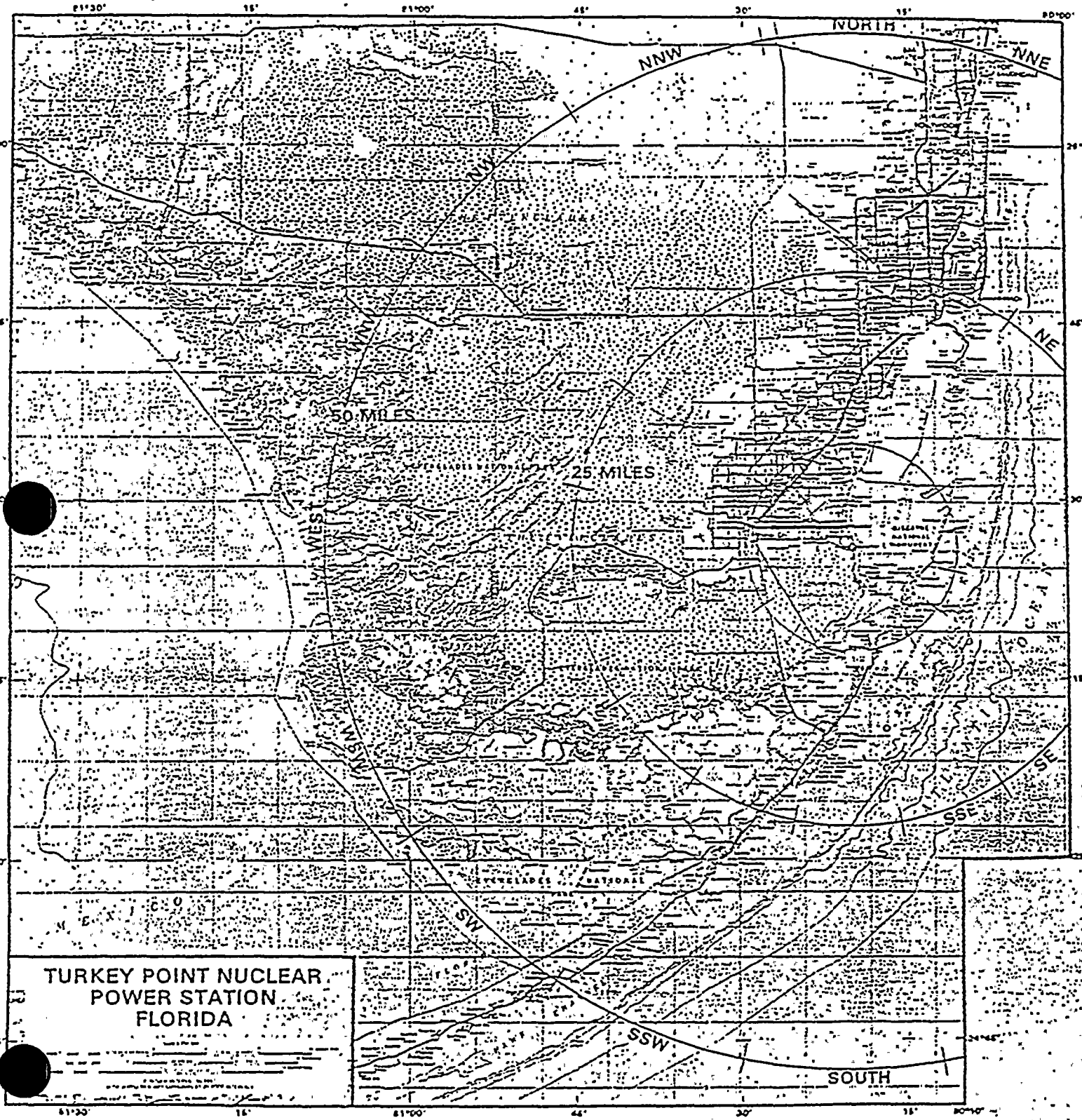
Bases of Technical Specification 3.9.1.d.1 Limits

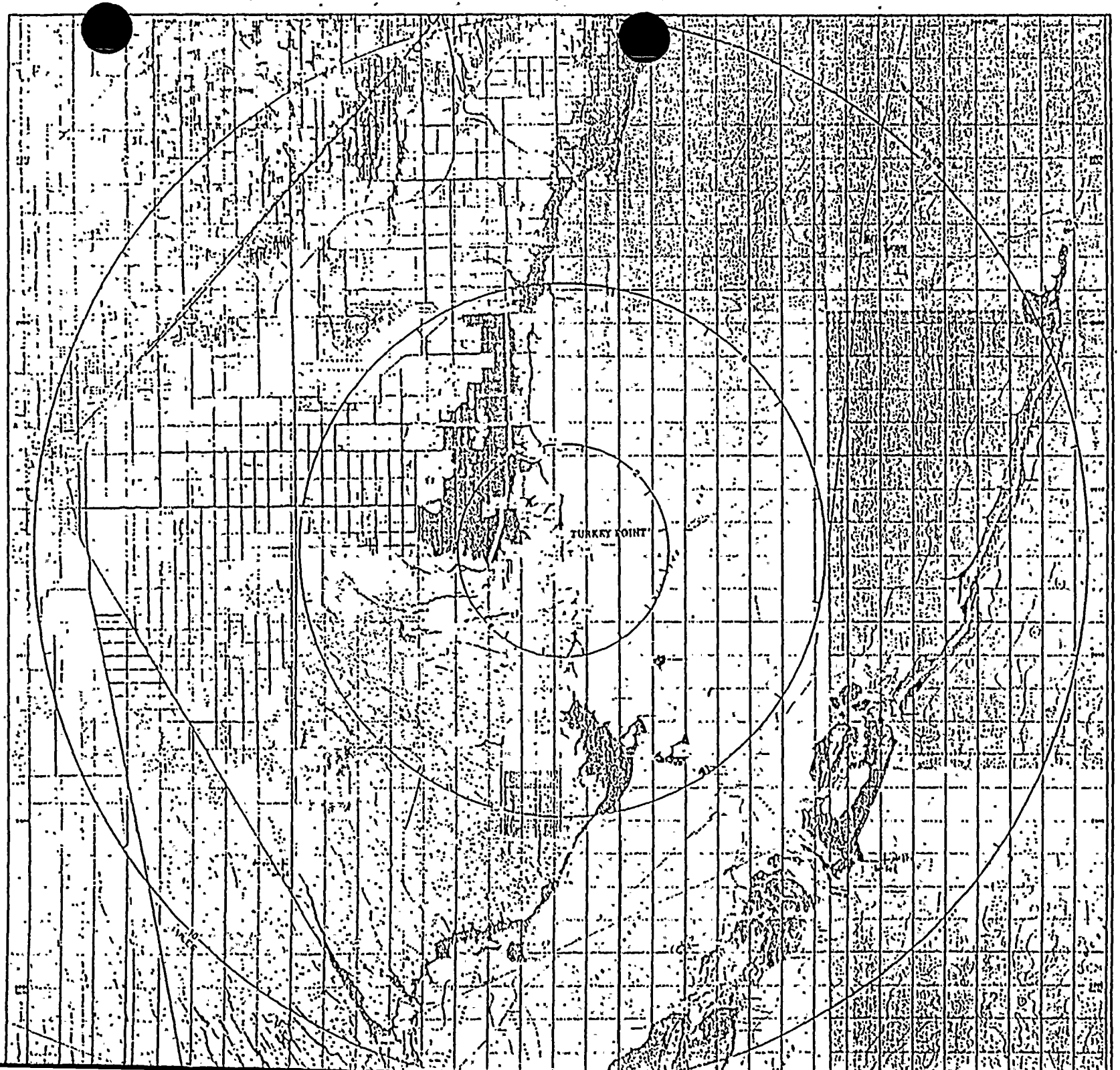
Technical Specification 3.9.1.d.1 conditions for treating radioactive liquid waste are stated on a quarterly basis as one-fourth of the design objective (annual) limits on individual dose commitment. At the Turkey Point Plant, the individual dose limits require lower releases of radioactive material in liquid effluents than would cost-benefit criteria for treatment. This occurs because the closed cooling canal system yields a low population dose. Consequently, a liquid radwaste system operability requirement can be based on Appendix I design basis limits without any further reduction for treatment cost-beneficiality.

ODCM sections 2.4 and 2.5 present the dose projection method used to decide operability. Since 90 percent or more of the dose is normally caused by a limited number of radionuclides, dose projections can be adequately based upon those radionuclides. Appendix C explains the basis for this position.

50-MILE RADIUS EMERGENCY PLANNING MAP

U.S. NUCLEAR REGULATORY COMMISSION-1981





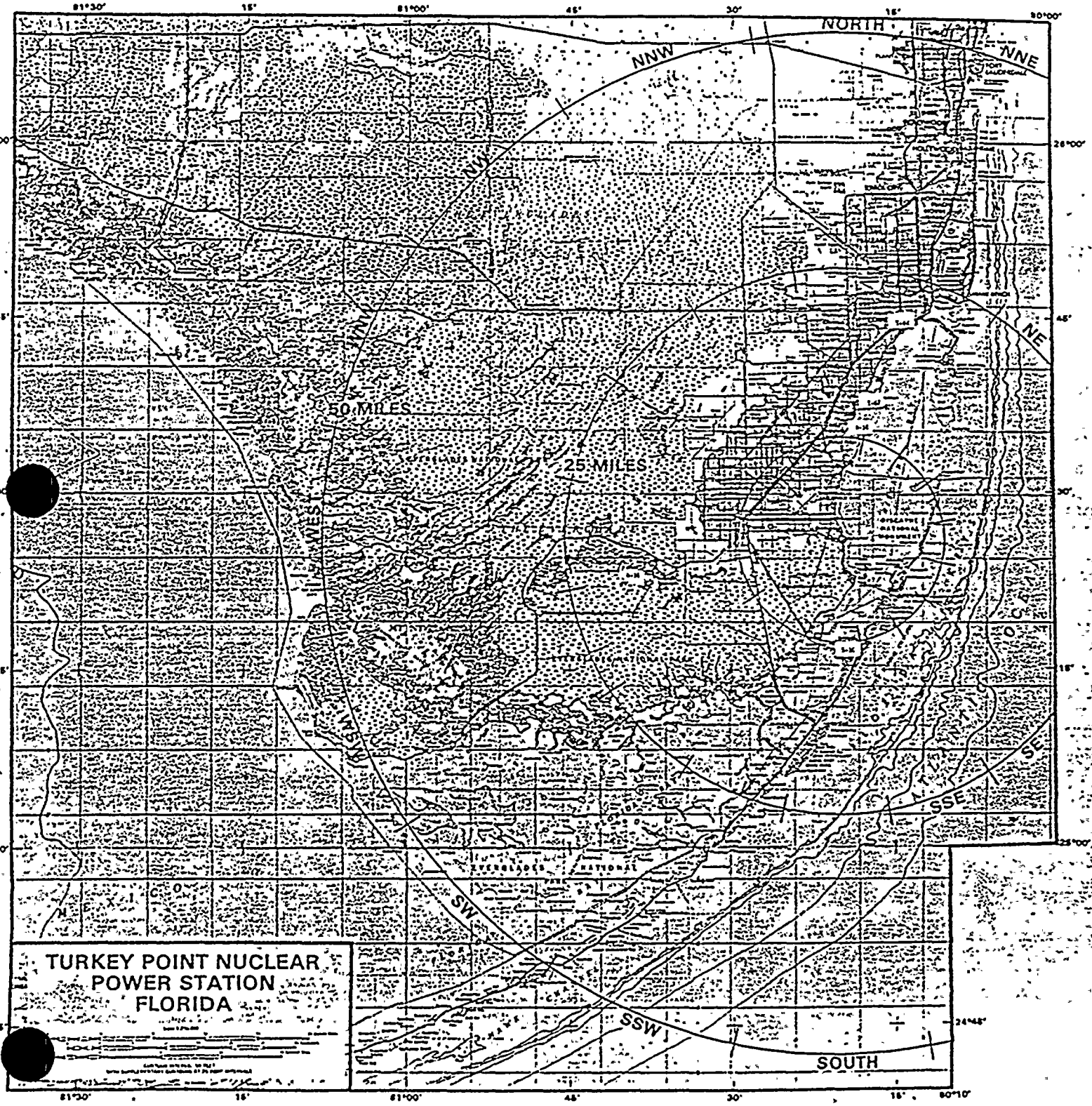
TURKEY POINT NUCLEAR POWER STATION

Biscayne Keys, Fla.

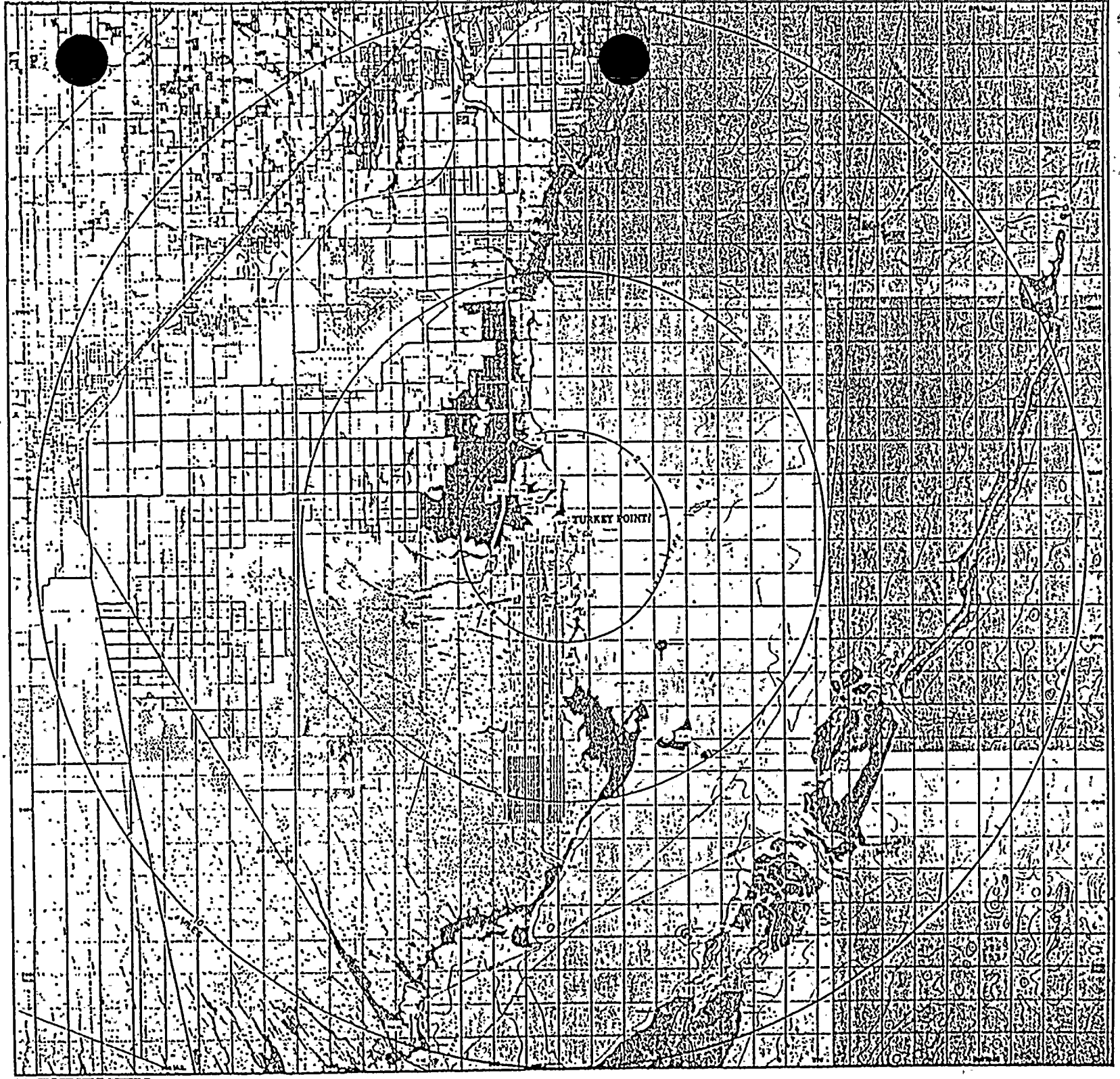
10-Mile Radius
Emergency Planning Map

50-MILE RADIUS EMERGENCY PLANNING MAP

U.S. NUCLEAR REGULATORY COMMISSION-1981



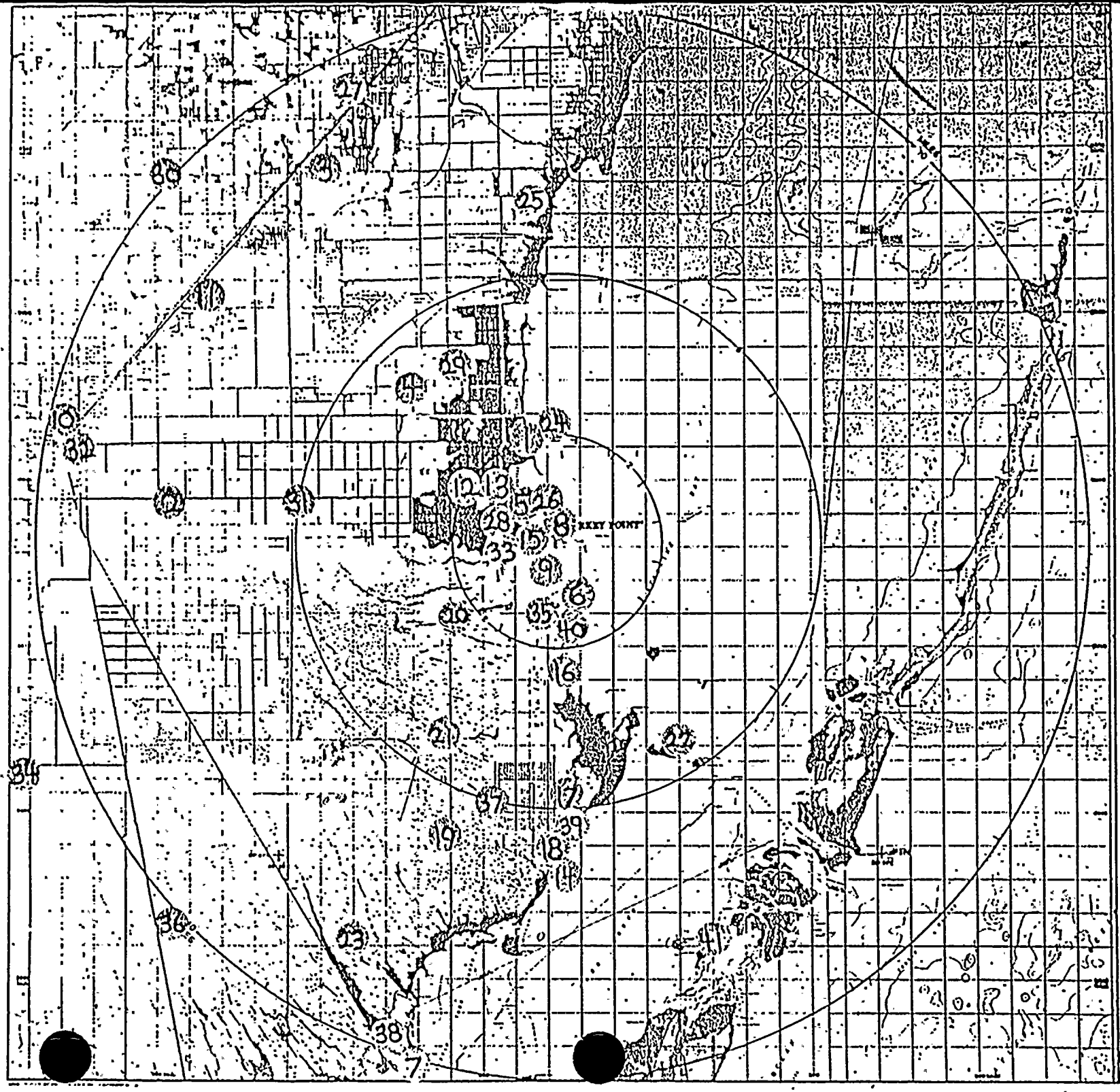
TURKEY POINT NUCLEAR
POWER STATION
FLORIDA



TURKEY POINT
NUCLEAR POWER
STATION

Arsenicker Keys, Fla.

10-Mile Radius
Emergency Planning Map


TURKEY POINT
NUCLEAR POWER
STATION
Arsenicker Keys, Fla.
10-Mile Radius
Emergency Planning Map

TURKEY POINT PLANT
REFERENCE TO MAP LOCATIONS

LOCATION	DESIGNATED NUMBER	LOCATION	DESIGNATED NUMBER
T51	1	N-1	24
T52	2	N-5	25
T56	3		
T57	4	NNW-1	26
T58	5	NNW-10	27
T63	6	NW/WNW-1	28
T66	7	NW-5	29
T71	8	NW-10	30
T72	9	W/WNW-5	31
T73	10	WNW-10	32
T74	11	W-1	33
T75A	12	WSW-10	34
T75B	13	SW/SSW-1	35
T81	14	SW-10	36
T84	15	SSW/SW-5	37
T87	16	SSW-10	38
T88	17	S-5	39
T89	18	SSE/S-1	40
T90	19	SSE-10	41
T91	20		
T92	21	-	
T95	22	-	
T98	23		