

DAVIS-BESSE

OFFSITE DOSE CALCULATION MANUAL

Revision 4, 1991

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THE TOLEDO EDISON COMPANY  
DAVIS-BESSE NUCLEAR POWER STATION  
OFFSITE DOSE CALCULATIONS MANUAL

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## DAVIS-BESSE OFFSITE DOSE CALCULATION MANUAL

### 1.0 INTRODUCTION

The Davis-Besse Offsite Dose Calculation Manual (ODCM) describes the methodology and parameters used in: 1) determining the radioactive material release rates and cumulative releases; 2) calculating the radioactive liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints; and 3) calculating the corresponding dose rates and cumulative quarterly and yearly doses. The Radiological Environmental Monitoring Program is also described. Sampling locations, media and collection frequencies are presented; analytical requirements are specified. The methodology provided in this manual is acceptable for use in demonstrating compliance with concentration limits of 10 CFR 20.106 and the cumulative dose criteria of 10 CFR 50, Appendix I and 40 CFR 190, and the Davis-Besse Radiological Effluent Technical Specifications.

The exposure pathway and dose modeling as presented in this ODCM will, in general, provide estimates (e.g., calculational results) that are conservative (i.e., higher than actual exposures in the environment). This conservatism does not invalidate the modeling since the main purpose of these calculations is for demonstrating "As Low As is Reasonably Achievable" (ALARA) for radioactive effluents. In using these models for evaluation and controlling actual effluents, further simplification and conservatism may be applied. For purposes of demonstrating compliance with the EPA environmental dose standard for the Uranium Fuel Cycle (40 CFR 190), more realistic dose assessment modeling may be used to provide more accurate assessment of actual radiation exposures resulting from the operation of the Davis-Besse Nuclear Power Station.

The ODCM will be maintained at the station for use as a reference guide and training document of accepted methodologies and calculations. Changes to the ODCM calculational methodologies and parameters will be made as necessary to ensure reasonable conservatism in keeping with the principles of 10 CFR 50, Appendix I, Section III and IV. Questions about the ODCM should be directed to the Manager - Radiological Control.

NOTE: Throughout this document words appearing all capitalized denote definitions specified in the Davis-Besse Technical Specifications (TS), in Section 7.5 of this manual, or common acronyms.

Section 2.0 of the ODCM describes equipment for monitoring and controlling liquid effluents, sampling requirements, and dose evaluation methods. Section 3.0 provides similar information on gaseous effluent controls, sampling, and dose evaluation. Section 4.0 describes special dose analyses required for Regulatory Guide 1.21, Semiannual Effluent Reporting and EPA Environmental Dose Standard of 40 CFR 190. Section 5.0 describes the role of the annual land use census in identifying the controlling pathways and locations of exposure for assessing the potential offsite doses. Section 6.0 describes the Radiological Environmental Monitoring Program. Section 7.0 describes the environmental, effluent and special reporting requirements, procedural requirements for major changes to liquid and gaseous radwaste systems, and definitions.



## 2.0 LIQUID EFFLUENTS

This section summarizes information on the liquid effluent radiation monitoring instrumentation and controls. More detailed information is provided in the Davis-Besse USAR, Section 11.2, Liquid Waste Systems and associated design drawings from which this summary was derived. This section also describes the sampling and analysis required by Technical Specifications. Methods for calculating alarm setpoints for the liquid effluent monitors are presented; methods for evaluating doses from liquid effluents are derived.

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with methods in Section 2.3 to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

The radioactive liquid effluent monitoring instrumentation channels listed in Table 2-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of ODCM Section 2.2 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters of Section 2.3.

### 2.1 Radiation Monitoring Instrumentation and Controls

This Section summarizes the instrumentation and controls that monitor the liquid effluents. This discussion focuses on the role of this equipment in assuring compliance with the Davis-Besse Technical Specifications and ODCM. Location and control function of the monitors are displayed in ODCM Figure 2-1.

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 2-2. Each of these operations shall be performed within the specified time interval with a maximum allowable extension not to exceed 25 percent of the specified interval.

NOTE: The monitors indicated in a, b, and c below are inoperable if surveillances are not performed or setpoints are less conservative than required.

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

With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the actions described in Table 2-1. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Effluent and Waste Disposal Report (Section 7.2) why the inoperability was not corrected in a timely manner.

### 2.1.1 Technical Specification Requirement

This section prescribes the monitoring required during liquid releases in order to comply with TS and the backup sampling required when monitors are inoperable. The liquid effluent monitoring instrumentation for controlling and monitoring radioactive liquid effluents in accordance with Davis-Besse TS is summarized below.

#### a) Alarm (and Automatic Termination)

##### i. Clean Radwaste Effluent Monitors (RE-1770 A & B).

Discharges from the Clean Radwaste Monitor Tanks (2) are monitored by redundant radiation monitoring systems (RE-1770 A & B). These monitors detect gross gamma activity in the effluent prior to mixing in the Collection Box. Measurements from each detector read out on the Victoreen panel in the Control Room. Each monitoring system is capable of initiating an alarm and an automatic isolation of the release by closing valve WC-1771. The method for determining setpoints for the High Alarm, which initiates isolation, is discussed in ODCM Section 2.3.

##### ii. Miscellaneous Radwaste Effluent Monitors (RE-1878 A & B).

Discharges from the Miscellaneous Liquid Waste Monitor Tank and the Detergent Waste Drain Tank are monitored by redundant radiation monitoring systems (RE-1878 A & B). These monitors detect gross gamma activity in the effluent line prior to mixing in the Collection Box. Measurements from each detector read out on the Victoreen panel in the Control Room. Each monitor is separately capable of initiating an alarm and automatic isolation of the release by closing valve WM-1876. Setpoint determination for the High Alarm, which initiates isolation, is discussed in ODCM Section 2.3.1.

#### b) Alarm (only)

##### i. Turbine Building Sump Effluent Line (RE-4686).

The purpose of the monitor on the Turbine Building sump effluent line is to detect abnormal radionuclide concentrations in the sump effluent prior to discharge to the onsite basin Training Center Pond. This monitor is located near the end of the storm sewer drain pipe, upstream of the final discharge point into the Training Center Pond. This stream is commonly referred to as the Turbine Building Sump/Storm Sewer Drain (TBS/SSD). The source of any radioactive material in the sump would be from the secondary steam system. Therefore, activity is expected in the turbine building sump effluent system if a primary-to-secondary leak has occurred. If a primary-to-secondary leak is present, the activity in the sump effluent system would be comprised of those radionuclides found in the secondary system. Evaluation of alarm setpoint for RE-4686 is discussed in ODCM Section 2.3.2.

c) Flow Rate Measuring Devices

In order to comply with TS, the release rate of liquid radwaste discharges shall be monitored. The following flow indicators and totalizers meet this requirement:

- i. Clean Radwaste Effluent Line  
Flow Indicator (FI) 1700 A & B  
Flow Totalizer (FQI) 1700 A & B
- ii. Miscellaneous Radwaste Effluent Line  
Flow Indicator (FI) 1887 A & B  
Flow Totalizer (FQI) 1887 A & B
- iii. Dilution Flow to the Collection Box  
Computer Point F201

2.1.2 Non-Technical Specification Monitors

Additional monitors, although not required to satisfy the requirements of the Davis-Besse TS, have been installed to control liquid radioactive material and reduce the likelihood of unmonitored releases. The monitors are:

- Collection Box Outlet to the Lake (RE-8433) - monitors the final station effluent to the lake.
- Component Cooling Water System (CCWS) (RE-1412 & 1413) - monitors the CCWS return line. High alarm closes the atmospheric vent valves on the CCWS surge tank.
- Service Water System (SWS) (RE-8432) - single off-line detector monitoring the SWS outlet prior to discharge to the Collection Box.
- Intake Forebay (RE-8434) - single detector continuously monitors the station intake water from Lake Erie.

2.2 Sampling and Analysis of Liquid Effluents

The program for sampling and analysis of liquid waste is prescribed in this Section and incorporates the basic requirements outlined in TS. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 2-3. Table 2-3 identifies three potential sources of liquid radioactive effluents for which sampling and analysis are required to ensure releases are controlled in accordance with the TS limits.

The results of the radioactivity analyses shall be used in accordance with the methodology and parameters of this Section of the ODCM to assure that the concentrations at the point of release are maintained within the following limits.

The concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS shall be limited to the concentrations specified in 10 CFR Part 20.106 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be



limited to  $2.0 \text{ E-04 } \mu\text{Ci/ml}$ . If the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding these limits, then without delay restore the concentrations to within these limits. The sources of radioactive effluents and the associated sampling and analysis requirements are discussed below.

#### 2.2.1 Batch Waste Release Tanks

BATCH RELEASES are defined as the discharge of liquid wastes of a discrete volume. The releases from the Clean Waste Monitor Tanks 1-1 and 1-2, the Miscellaneous Liquid Waste Monitor Tank, and the Detergent Waste Drain Tank are classified as BATCH RELEASES. The following sampling and analysis requirements must be met for all releases from these tanks.

- Prior to each BATCH RELEASE, analysis of a representative grab sample for principal gamma emitters (including I-131 and other peaks identified by gamma spectroscopy).
- Once per month, analysis of one sample from a BATCH RELEASE for dissolved and entrained gases (gamma emitters). (See note below.)
- Once per month, analysis of a COMPOSITE SAMPLE of all releases that month for tritium and gross alpha activity. The COMPOSITE SAMPLE is required to be representative of the liquids released. Samples contributed to the composite are to be proportional to the quantity of liquid discharged.
- Once per quarter, analysis of a COMPOSITE SAMPLE of all releases that quarter for Strontium (Sr)-89, Sr-90, and Iron (Fe)-55.

NOTE: Identification of noble gases that are principal gamma emitting radionuclides are included as a part of the gamma spectral analysis performed on all liquid radwaste effluents. Therefore, the Table 2-3 requirement for sampling and analysis of one batch per month for noble gases need not be performed as a separate program. The gamma spectral analysis on each BATCH RELEASE meets the intent of this requirement.

#### 2.2.2 Turbine Building Sump/Storm Sewer Drain (TBS/SSD)

Releases\* from the TBS/SSD are classified as continuous releases, since these discharges are not controlled on a batch basis. Table 2-3 requires that a sample shall be collected from the TBS/SSD, if the on-line monitor is out-of-service and the activity level of the condensate (i.e., hot well water) exceeds  $1.0 \text{ E-05 } \mu\text{Ci/ml}$  gross beta/gamma. During this period, a sample is to be collected once every 12 hours and analyzed for principal gamma emitters.

As a back-up to the on-line monitors, grab samples are periodically collected (nominally once per week) from the TBS/SSD and analyzed by gamma spectroscopy. If activity is identified, additional controls are enacted to ensure that the release concentrations are maintained below MPC as required by TS and that the cumulative releases are a small fraction of the dose limit of TS. The following actions will be considered for controlling any radioactive material releases via the TBS/SSD:

- As needed for controlling and quantifying releases, the sampling frequency of the TBS/SSD will be increased to every 8 hours until the source of the contamination is found and controlled.
- Gamma spectral analysis will be performed on each sample for principal gamma emitters.
- The measured radionuclide concentrations from the gamma spectral analysis will be compared with MPC (equation 2-2) to ensure releases are within the limits.
- Based on the measured concentrations, a re-evaluation of the alarm setpoint for the TBS/SSD monitor (RE-4686) will be performed as specified in ODCM Section 2.3.2.
- Each sample will be considered representative of the releases that have occurred since the previous sample. The volume of liquid released will be determined based on the Turbine Building Sump pump runtimes and flows.
- From the sample analysis and the calculated volume released, the total radioactive material released will be determined and considered representative of the release period. Cumulative doses will be determined in accordance with ODCM Section 2.5.

\* Discharges from the TBS/SSD are routed to the Training Center Pond with the pond overflow discharging to the Toussaint River. For conservatism, it is assumed that any radioactive material releases from the TBS/SSD to the Training Center Pond are ultimately discharged to the lake environment (unless actions are taken to prevent occurrence).

#### 2.2.3 Condensate Demineralizer Backwash

Discharges from the Condensate Demineralizer Backwash Receiving Tank (BRT) to the South Settling Basin are controlled as BATCH RELEASES in accordance with Table 2-3. Samples are collected prior to each release of the resin/water slurry and separated into the liquid phase (transfer water) and solid phase (resin). These samples are separately analyzed for principal gamma emitters. Toledo Edison has imposed guidelines on concentrations of radionuclides that may be discharged to the onsite settling basin. These guidelines are presented in Table 2-4.

The radioactive material contamination in the condensate demineralizer backwash will be contained on the powdered resin; soluble or suspended radioactive material associated with the water phase is not expected. However, the resin and the water are analyzed separately thus allowing for a determination of the amounts retained onsite in the settling basin (i.e., the resin) and the amounts released to Lake Erie as an effluent (i.e., the water that is ultimately released to the lake as the decant from the basin).

The BRT receives the spent resin from the Condensate Clean-up System. Low-level radioactive material contamination of the spent resin is periodically expected due primarily to minor weeps in the steam generators and the leaching of residual activity in the secondary system that was deposited from the steam generator tube leaks.

During primary-to-secondary leakage, activity levels will be elevated and typically above the limits imposed for acceptable discharge to the basin. Under these conditions, the powdered resins are retained within the plant and processed as solid radwaste for offsite transport and disposal at a licensed radioactive waste disposal site. If within the criterion of Table 2-4, the BRT may be discharged to the onsite settling basin.

### 2.3 Liquid Effluent Monitor Setpoints

Technical Specifications require that the concentration of radioactive materials released in liquid radioactive effluents from the site to UNRESTRICTED AREAS shall not exceed the UNRESTRICTED AREA MPC at the discharge point to Lake Erie. This limitation provides additional assurance that the levels of radioactive material in bodies of water outside the site should not result in exposures exceeding:

- The Section II.A design objective of Appendix I, 10 CFR Part 50, to an individual, and
- the limits of 10 CFR Part 20.106(e) to the population.

Dissolved or entrained noble gases in liquid effluents are limited to a concentration of  $2.0 \text{ E-04 } \mu\text{Ci/ml}$ , total activity. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

Radiation monitor setpoints shall be established to alarm and trip prior to exceeding the limits specified above. To meet this requirement, the alarm/trip setpoint for liquid effluent monitors are determined in accordance with the following equation:

$$SP \leq \frac{CL (DF+RR)}{RR} \quad (2-1)$$



where:

- CL = the effluent concentration limit implementing 10 CFR Part 20.106 (i.e., MPC at discharge point) in  $\mu\text{Ci/ml}$ , defined in equation (2-4).
- SP = the setpoint, in  $\mu\text{Ci/ml}$ , of the monitor measuring the radioactivity concentration in the effluent line prior to dilution. The setpoint represents a value which, if exceeded, would result in concentrations exceeding the MPC in the UNRESTRICTED AREA.
- RR = the liquid effluent release rate as measured at the radiation monitor location, in volume per unit time, but in the same units as DF, below.
- DF = the dilution water flow as measured prior to the release point in volume per unit time.

At Davis-Besse a minimum required dilution water flow is established for a given release, and the waste tank release rate (RR) and monitor setpoint (SP) are set to meet the condition of equation 2-1 for a given effluent concentration limit, CL.

NOTE: If no dilution is provided,  $SP \leq CL$ . Also, when DF is large compared to RR, then  $(DF + RR) \approx DF$ .

#### 2.3.1 Liquid Radwaste Effluent Line Monitor (RE-1770 A & B, RE-1878 A & B)

The Liquid Radwaste Effluent Line Monitors provide alarm and automatic termination of releases prior to exceeding MPC. As required by Table 2-3 and as discussed in ODCM Section 2.2.1, a sample of the liquid radwaste to be discharged is collected and analyzed by gamma spectroscopy to identify principal gamma emitting radionuclides. From the measured individual radionuclide concentrations, the required dilution flow and the allowable release rate are determined.

The dilution flow and allowable release rate are inversely proportional to the ratio of the radionuclide concentrations to their MPC values. This ratio of measured concentration to MPC values is referred to as the "MPC fraction" and is calculated by the equation:

$$\text{MPCF} = \sum_i \frac{C_i}{\text{MPC}_i} \quad (2-2)$$

where:

MPCF = fraction of the unrestricted area MPC for a mixture of radionuclides

$C_i$  = concentration of each radionuclide (i) measured in tank prior to release ( $\mu\text{Ci/ml}$ )

MPC<sub>i</sub> = unrestricted area MPC for each radionuclide (i) from 10 CFR Part 20, Appendix B, Table II, Column 2. For dissolved and entrained noble gases an MPC value of  $2.0\text{E-}04 \mu\text{Ci/ml}$  shall be used.

As expressed in equation (2-1), the concentration limit (CL) of a liquid radwaste discharge is the same as the effective MPC for the radionuclide mixture of the discharge. Simply, the CL (or effective MPC) represents the equivalent MPC value for a mixture of radionuclides evaluated collectively. The equation for determining CL is:

$$\text{CL} = \frac{\sum C_i}{\text{MPCF}} \quad (2-3)$$

Based on the MPCF, the minimum dilution factor (DF) for the conduct of the release is established at 3.33 times larger than actually required. This safety factor (SF) provides conservatism, accounting for variations in monitor response and flow rates and also for the presence of radionuclides that may not be detected by the monitors (i.e., non-gamma emitters). The following equation is used for calculating the required minimum dilution factor:

$$\text{DF} = \text{MPCF}/\text{SF} \quad (2-4)$$

where:

DF = minimum required dilution factor

SF = 0.3 administrative safety factor

The allowable release rate is then calculated by dividing the available dilution flow (ADF) at the Collection Box by DF as calculated by equation (2-4).

$$\text{MAX RR} = \text{ADF}/\text{DF} \quad (2-5)$$

where:

MAX RR = maximum allowable release rate (gal/min)

ADF = available dilution flow at the Collection Box as measured by Computer Point F 201 (gal/min)

NOTE: Equations (2-3) and (2-4) are valid only for MPCF >1; for MPCF <1, the waste tank concentration meets the limits of 10 CFR Part 20 without dilution, and MAX RR may take on any desired value.

If MAX RR as calculated above is greater than the maximum discharge pump capacity, the pump capacity should be used in establishing the actual release rate RR for the radwaste discharge. For releases from the Miscellaneous Waste Monitor Tank and Detergent Waste Drain Tank, the discharge pump capacity is 100 gpm (i.e., the design limit for MAX RR is 100 gpm); for the Clean Radwaste Tank, this value is 140 gpm.

Based on the calculated release rate (RR), the dilution factor (DF) and the concentration limit (CL), the alarm setpoint is calculated as prescribed in equation (2-1) by the equation:

$$SP = \frac{I (C_i * SEN_i) * DF}{MPCF * RR} + Bkg \quad (2-6)$$

where:

- SP = setpoint of the radiation monitor (counts per second - cps)
- $C_i$  = concentration of radionuclide (i) as measured by gamma spectroscopy ( $\mu\text{Ci/ml}$ )
- $SEN_i$  = monitor sensitivity for radionuclide (i) based on calibration curve (cps/( $\mu\text{Ci/ml}$ ))
- DF = the required minimum dilution factor as defined in equation (2-4) (gal/min)
- RR = actual release rate of the liquid radwaste discharge (gal/min)
- MPCF = MPC fraction as determined by equation (2-2)
- Bkg = background reading of monitor (cps)

The Cs-137 sensitivity may be used in lieu of the sensitivity values for individual radionuclides. The Cs-137 sensitivity provides a reasonably conservative monitor response correlation for radionuclides of interest in reactor effluents. Coupled with the safety factor SF in equation (2-3), this assumption simplifies the evaluation without invalidating the overall conservatism of the setpoint determination.

Prior to conducting any batch liquid radwaste release, equations (2-4) and (2-5) are used to determine the minimum required dilution flow and the allowable release rate. Equation (2-6) is then applied to determine the RE-1770 A & B or RE-1878 A & B alarm setpoints.

### 2.3.2 Turbine Building Sump/Storm Sewer Drain Monitor (RE-4686)

The setpoint for the TBS/SSD radiation monitor RE-4686 shall be established to ensure the radioactive material concentration in the effluent prior to discharge offsite does not exceed MPC, UNRESTRICTED AREA (10 CFR 20, Appendix B, Table II, Column 2). The TBS/SSD is not normally radioactively contaminated. Therefore, the setpoint for this monitor has been established at its lowest practical level (i.e., two times the normal background) in order to provide an early indication of any abnormal conditions. If radioactivity is found in this system, then a setpoint may be determined by using the measured radioactive material concentration from the grab sample coupled with the algorithm of equation (2-8). This approach for determining the alarm setpoint is the same as presented in Section 2.3.1 for the Liquid Radwaste Effluent Line Monitors. Equation (2-1) remains valid, except that, for the TBS/SSD line monitor, the dilution flow previously assumed for diluting the batch liquid radwaste effluents is now the release rate as determined from the Turbine Building sump flow. There is no additional dilution prior to discharge to the Training Center Pond. Thus, Equation (2-1) simplifies to:

$$SP \leq CL \quad (2-7)$$

Also, since discharge is to the Training Center Pond, exceeding a setpoint does not necessarily mean exceeded TS limit on release concentration to the lake. The verification of compliance with the TS limits on concentration should be based on actual samples of the effluent from the pond to the lake environment. (Refer to ODCM Section 2.3.4).

Substituting equation (2-3) for CL, the alarm setpoint can be calculated by the equation:

$$SP \leq \frac{L (C_i * SEN_i)}{MPCF} \quad (2-8)$$

where:

$C_i$  = concentration of each radionuclide (i) in the TBS/SSD effluent ( $\mu\text{Ci/ml}$ )

$MPCF$  = MPC fraction as determined by equation (2-2)

$SEN_i$  = monitor sensitivity for radionuclide (i) based on calibration curve ( $\text{cpm}/\mu\text{Ci/ml}$ )

Again, the Cs-137 sensitivity may be used in lieu of the individual radionuclide evaluation as discussed for equation (2-6).



### 2.3.3 Alarm Setpoints for the Non-Technical Specification Radiation Monitors

- i. Collection Box Outlet to the Lake (RE-8433). As discussed in ODCM Section 2.1.2, the Radiation Monitor on Collection Box outlet utilizes a single off-line detector to continuously monitor all station liquid effluent discharges to the lake. Although this is the final effluent monitor, it does not serve any control function. Control functions have been placed on the upstream undiluted effluent line that will terminate the release prior to exceeding the MPC for UNRESTRICTED AREAS. This monitor therefore provides the Control Room operator with a final check of the total diluted effluent stream. Since this monitor views the diluted radwaste discharges, its response during routine operations will be minimal (i.e., typical of background levels). Therefore, the alarm setpoint for this monitor should be established as close to background as possible without incurring a spurious alarm due to background fluctuations. The alarm is established in accordance with the Radiation Monitor Setpoint Manual.
- ii. Component Cooling Water System (CCWS) (RE-1412 & 1413). The monitors RE-1412 and 1413 provide indication of a breach in the CCWS integrity, allowing primary system water to enter and contaminate the system. Therefore, the alarm setpoint is established as close to background as possible without incurring a spurious alarm due to background fluctuations. The alarm is established in accordance with the Radiation Monitor Setpoint Manual.
- iii. Service Water System (SWS) (RE-8432). No radioactive material is expected to be contained within the SWS during normal operations. Therefore, the alarm setpoint is established as close to background as possible without incurring a spurious alarm due to background fluctuations. The alarm is established in accordance with the Radiation Monitor Setpoint Manual.
- iv. Intake Forebay Monitor (RE-8434). The alarm setpoint for this monitor should be established as close to background as possible without incurring a spurious alarm due to background fluctuations. Although a very remote potential, a verified alarm from this system would indicate a possible contamination of the station intake water. The alarm is established in accordance with the Radiation Monitor Setpoint Manual.

### 2.3.4 Alarm Response - Evaluating Actual Release Conditions

Liquid release rates are controlled and alarm setpoints are established to ensure that releases do not exceed the concentration limits of Section 2.3 (i.e., 10 CFR 20 MPC's at the discharge to Lake Erie). However, if any of the monitors (RE-1770 A & B, RE-1878 A & B, or RE-4686) alarm during a liquid release, it becomes necessary to re-evaluate the release conditions to determine compliance with the limits. Following an alarm, the actual release conditions should be determined. Radioactive material

concentrations should be evaluated by sampling the effluent stream (or resampling the waste tank). Discharge flow and dilution water flow should be redetermined. The following actions should be considered:

- verify radiation monitor alarm setpoint, ensure consistency with the setpoint evaluation for the release.
- re-sample and re-analyze the source of the release (e.g., release tank, TB sump, decant from Training Center Pond to the Toussaint River).
- re-define the release conditions on the release rate and the dilution water flow.

Based on these data, the following equation may be used for evaluating the actual release conditions:

$$I \frac{C_i}{MPC_i} + \frac{RR}{DF + RR} \leq 1 \quad (2-9)$$

where:

- $C_i$  = measured concentration of radionuclide (i) in the effluent stream ( $\mu\text{Ci/ml}$ )
- $MPC_i$  = the MPC value for radionuclide (i) from 10 CFR 20, Appendix B, Table II, Column 2 ( $\mu\text{Ci/ml}$ )
  - =  $2.0\text{E}-04 \mu\text{Ci/ml}$  for dissolved or entrained noble gases
- $RR$  = actual release rate of the liquid effluent at the time of the alarm
- $DF$  = actual dilution water flow at the time of the release alarm

#### 2.4 Liquid Effluent Dose Calculation - 10 CFR 50

The parameters of the liquid release (or estimated parameters, for a pre-release calculation) may be used to calculate the hypothetical dose to a MEMBER OF THE PUBLIC from the release (or planned release). The dose calculation provides a conservative method for estimating the impact of radioactive effluents released by Davis-Besse and for comparing the impact against limits set by the NRC in the Davis-Besse TS.

The limits are specified below as quarterly and calendar year limits.

##### 2.4.1 MEMBER OF THE PUBLIC Dose - Liquid Effluents

This requirement is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50.



Technical Specification limits the dose or dose commitment to MEMBERS OF THE PUBLIC from radioactive materials in liquid effluents from Davis-Besse. The limits are:

- during any calendar quarter:
  - < 1.5 mrem to total body
  - < 5.0 mrem to any organ
- during any calendar year:
  - < 3.0 mrem to total body
  - < 10.0 mrem to any organ

These limits implement the guides set forth in Section II.A of Appendix I, 10 CFR, Part 50.

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Section 7.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This action provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I, 10 CFR Part 50 to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable."

NOTE: For fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements of Section III.A of Appendix I of 10 CFR Part 50 that conformance with the guides of Appendix I is to be shown by calculational procedures based on modes and data such that the actual exposure of an individual thorough appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

TS requires that cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

The calculation of the potential doses to MEMBERS OF THE PUBLIC is a function of the radioactive material releases to the lake, the subsequent transport and dilution in the exposure pathways, and the resultant individual uptake. At Davis-Besse, the combined fish consumption and drinking water pathway has been modeled to provide a conservative dose assessment for exposures to MEMBERS OF THE PUBLIC. For the fish pathway, it has been conservatively assumed that the maximum exposed individual consumes 21 kg per year of fish taken in the immediate vicinity of the Davis-Besse discharge to the lake. For the drinking water pathway, the conservative modeling is based on an individual drinking 730 liters per year of water from the beach wells located 966 m to the NW of the site discharge. (It is important to note that because of the high sulfur content, the water from these beach wells is not suitable for consumption; however, for conservatism this pathway has been included in the dose modeling for the maximum exposed individual.)

The equation for assessing the maximum potential dose to MEMBERS OF THE PUBLIC from liquid radwaste releases from Davis-Besse is:

$$D_o = \frac{1.67E-02 * VOL}{DF * Z} * I (C_i * A_{i,o}) \quad (2-10)$$

where:

- $D_o$  = dose or dose commitment to organ (o) including total body (mrem)
- $A_{i,o}$  = site-specific ingestion dose commitment factor to the total body or any organ (o) for radionuclide (i) (mrem/hr per  $\mu\text{Ci/ml}$ )
- $C_i$  = average concentration of radionuclide (i) in undiluted liquid effluent representative of the the volume VOL ( $\mu\text{Ci/ml}$ )
- VOL = total volume of liquid effluent released (gal)
- DF = average dilution water flow during release period (gal/min)  
(minimum value is typically 20,000 gpm)
- Z = 10, near field dilution factor\*
- 1.67E-02 = 1 hr/60 min

The site-specific ingestion dose/dose commitment factors ( $A_{i,o}$ ) represent a composite dose factor for the fish and drinking water pathway. The site-specific dose factor is based on the NRC's generic maximum individual consumption rates. Values of  $A_{i,o}$  are presented in Table 2-5. These values were derived in accordance with the guidance of NUREG-0133 using the following equation:

$$A_{i,o} = 1.14E+05 (U_w / D_w + U_f * BF_i) DF_i \quad (2-11)$$

where:

- $U_f$  = 21 kg/yr adult fish consumption
- $U_w$  = 730 liters/yr adult water consumption
- $D_w$  = 5.7, additional dilution from the near field to the beach wells (net dilution of 57\*)
- $BF_i$  = bioaccumulation factor for radionuclide (i) in fish from Table 2-6 ( $\mu\text{Ci/kg}$  per  $\mu\text{Ci/l}$ )
- $DF_i$  = dose conversion factor for nuclide (i) for adults in organ (o) from Table E-11 of Regulatory Guide 1.109 (mrem/ $\mu\text{Ci}$ )
- $1.14\text{E}+05 = 10^6 (\mu\text{Ci}/\mu\text{Ci}) * 10^3 (\text{ml/kg}) / 8760 (\text{hr/yr})$

The radionuclides included in the periodic dose assessment required by TS are those identified by gamma spectral analysis of the liquid waste samples collected and analyzed per the requirements of Table 2-3. In keeping with the NUREG-0133 guidance, the adult age group represents the maximum exposed individual age group. Evaluation of doses for other age groups is not required for demonstrating compliance with the dose criteria of TS. The dose analysis for radionuclides requiring radiochemical analysis will be performed after receipt of results of the analysis of the composite samples. In keeping with the required analytical frequencies of Table 2-3, tritium dose analyses will be performed at least monthly; Sr-89, Sr-90 and Fe-55 dose analyses will be performed at least quarterly.

- \* Near field dilution factor and dilution to beach wells are based on a study performed by Stone & Webster for Toledo Edison entitled "Aquatic Dilution Factors within 50 Miles of the Davis-Besse Unit 1 Nuclear Power Plant", June 1980.

#### 2.4.2 Simplified Liquid Effluent Dose Calculation

In lieu of the individual radionuclide dose assessment presented in Section 2.4.1, the following simplified dose calculation may be used for demonstrating compliance with the dose limits required by TS. Radionuclides included in this dose calculation should be those measured in the grab sample of the release (principal gamma emitters measured by gamma spectroscopy). H-3 should not be included in this analysis. Refer to Appendix A for the derivation of this simplified method.

##### Total Body

$$D_{tb} = \frac{9.70\text{E}+02 * \text{VOL}}{DF} * I C_i \quad (2-12)$$

##### Maximum Organ

$$D_{max} = \frac{1.19\text{E}+03 * \text{VOL}}{DF} * I C_i \quad (2-13)$$

where:

$C_i$  = average concentration of radionuclide (i) excluding H-3 in undiluted liquid effluent representative of the volume VOL ( $\mu\text{Ci}/\text{ml}$ )

VOL = volume of liquid effluent released (gal)

DF = average dilution water flow during release period (gal/min)

$D_{tb}$  = conservatively evaluated total body dose (mrem)

$D_{max}$  = conservatively evaluated maximum organ dose (mrem)

$9.70\text{E}+02 = 0.0167 \text{ (hr/min)} * 5.81\text{E}+05 \text{ (mrem/hr per } \mu\text{Ci/ml, Cs-134 total body dose factor from Table 2-5)} / 10 \text{ (near field dilution)}$

$1.19\text{E}+03 = 0.0167 \text{ (hr/min)} * 7.11\text{E}+05 \text{ (mrem/hr per } \mu\text{Ci/ml, Cs-134 liver dose factor from Table 2-5)} / 10 \text{ (near field dilution)}$

#### 2.4.3 Contaminated TBS/SSD System - Dose Calculation

If the TBS/SSD system becomes contaminated, any radioactive material releases must be included in the evaluation of the cumulative dose to a MEMBER OF THE PUBLIC as required by ODCM Section 2.2.2. Section 2.2.2 describes the methods for quantifying and controlling releases from the TBS/SSD system.

Although the discharges are via the Training Center Pond to the Toussaint River (instead of directly to Lake Erie), the modeling of equation (2-10) remains reasonably conservative for determining a hypothetical maximum individual dose. The following assumption should be applied for the dose assessment of any radioactive material releases from the TBS/SSD into the Training Center Pond and subsequently to the Toussaint River:

- If no additional controls are taken, it should be assumed that any radioactive material released to the Training Center Pond will ultimately be discharged to the lake environment.
- If actions are taken to limit any release, the assessment of dose should be made based on an evaluation of actual releases.
- The dilution flow (DF) should consider any additional dilution of the TBS/SSD discharge from other sources into the Training Center Pond prior to release to the river.

#### 2.5 Liquid Effluent Dose Projections

10 CFR 50.36a requires licensees to maintain and operate the radwaste system to ensure releases are maintained ALARA. This Section implements the requirements of 10 CFR Part 50.36c, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50. Based on a cost analysis of treating liquid radwaste, the specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as the dose design objectives as set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents. This requirement is implemented through this ODCM.



The liquid radioactive waste processing system shall be used to reduce the radioactive material levels in the liquid waste prior to release when the projected doses in any 31 day period would exceed:

- 0.06 mrem to the total body, or
- 0.20 mrem to any organ.

When the projected doses exceed either of the above limits, the waste must be processed by the liquid radwaste system prior to release. This dose criteria for processing is established at one quarter (1/4) of the design objective rate (i.e., 1/4 of 3 mrem/yr total body and 10 mrem/yr any organ over a 31 day projection).

With radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Section 7.3, a Special Report that includes the following information:

- Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,
- Action(s) taken to restore the inoperable equipment to OPERABLE status, and
- Summary description of action(s) taken to prevent a recurrence.

The applicable liquid waste processing system for maintaining radioactive material releases ALARA is the ion exchange system as delineated in Figure 2-1.

TS requires that in any month in which radioactive liquid effluent is being discharged without treatment, doses due to liquid releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

The projection of doses is made to evaluate the need for additional radwaste processing to ensure future releases are maintained ALARA. These projections are required only if the radwaste system has not been used (e.g., demineralizer system bypassed or resin bed exhausted). The following equations may be used for the dose projection calculation:

$$D_{tbp} = D_{tb} (31 / d) \quad (2-14)$$

$$D_{maxp} = D_{max} (31 / d) \quad (2-15)$$

where:

- $D_{tbp}$  = the total body dose projection for current 31 day period (mrem)
- $D_{tb}$  = the cumulative total body dose to date for current calendar quarter including release under consideration as determined by equation (2-10) or (2-12) (mrem)

- $D_{max}$  = the maximum organ dose projection for current 31 day period (mrem)
- $D_{max}$  = the maximum organ dose to date for current calendar quarter including release under consideration as determined by equation (2-10) or (2-13) (mrem)
- $d$  = the number of days to date in current calendar quarter
- 31 = the number of days in projection



Table 2-1  
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABILITY</u>	<u>ACTION</u>
1. Gross Radioactivity Monitors Providing Alarms and Automatic Termination of Release			
a. Liquid Radwaste Effluent Line (either Miscellaneous or Clean, but not both simultaneously)	1	(1)	A
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	1	(1)	B
b. Dilution Flow to Collection Box	1	(1)	B
3. Gross Beta or Gamma Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release			
a. Turbine Building/Storm Sewer Drain	1	(1)	B,C

TABLE 2-1 (continued)

TABLE NOTATION

(1) During radioactive releases via this pathway

- ACTION A      With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases may be resumed, provided that prior to initiating a release:
1. At least two independent samples are analyzed in accordance with Table 2-3 for analyses performed with each batch;
  2. At least two independent verification of the release rate calculations are performed;
  3. At least two independent verifications of the discharge valving are performed;
- Otherwise, suspend release of radioactive effluents via this pathway.
- ACTION B      With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.
- ACTION C      With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection no greater than  $1.0E-07$   $\mu\text{Ci/ml}$ .

TABLE 2-2

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>SOURCE CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation				
a. Liquid Radwaste Effluents Line	D <sup>(1)</sup>	P	R <sup>(3)</sup>	Q <sup>(2)</sup>
2. Flow Rate Monitors				
a. Liquid Radwaste Effluent Line	D <sup>(4)</sup>	N.A.	R	Q
b. Dilution Flow to Collection Box	D <sup>(4)</sup>	N.A.	R	Q

TABLE 2-2 (continued)

TABLE NOTATION

- (1) During releases via this pathway.
  - (2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint.
  - (3) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per eighteen months. For high range monitoring instrumentation, where calibration with a radioactive source is impractical, an electronic calibration may be substituted for the radiation source calibration.
  - (4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or BATCH RELEASES are made.
- 
- (D) At least once per 24 hours.
  - (P) Prior to each release.
  - (R) At least once per 18 month (550 dcys).
  - (Q) At least once per 92 days.



TABLE 2-3

## RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ( $\mu\text{Ci/ml}$ ) <sup>a</sup>
A. Batch Waste Release Tanks <sup>d</sup>	P	P	Principal	
	Each Batch	Each Batch	Gamma Emitters <sup>f</sup>	5.0E-07 <sup>b</sup>
			I-131 <sup>f</sup>	1.0E-06
	P	M	Dissolved and Entrained Gases	1.0E-05
	P	M	H-3	1.0E-05
	Each Batch	Composite <sup>c</sup>	Gross Alpha	1.0E-07
	P	Q	Sr-89, Sr-90	5.0E-08
B. Turbine Building Sump/Storm Sewer Drain			Fe-55	1.0E-06
	Continuous	S*	Principal Gamma Emitters <sup>f</sup>	5.0E-07 <sup>b</sup>
			I-131 <sup>f</sup>	1.0E-06
C. Condensate Demineralizer Backwash	P	P	Principal	
	Each Batch	Each Batch	Gamma Emitters <sup>f</sup>	5.0E-07 <sup>b</sup>
			I-131 <sup>f</sup>	1.0E-06

Table 2-3 (continued)

TABLE NOTATION

- a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

where

LLD is the lower limit of detection as defined above (as  $\mu\text{Ci}$  per unit mass or volume);

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

$E$  is the counting efficiency (as counts per transformation);

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

2.22 is the number of transformations per minute per picocurie;

$Y$  is the fractional radiochemical yield (when applicable);

$\lambda$  is the radioactive decay constant for the particular radionuclide;

$\Delta t$  for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

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

Table 2-3 (continued)

TABLE NOTATION

- b. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. For Ce-144, the LLD is  $2.0E-06$   $\mu\text{Ci/ml}$ . Other peaks which are measured and identified shall also be reported.

Nuclides which are below the LLD for the analysis should not be reported as being present at the LLD level. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent and Waste Disposal Report.

- c. A COMPOSITE SAMPLE is one in which the method of sampling employed results in a specimen which is representative of the liquids released.
- d. A BATCH RELEASE is the discharge of liquid wastes of a discrete volume.
- e. When the monitor is out of service, a grab sample shall be taken and analyzed once every 12 hours if the condensate pump discharge exceeds  $1.0E-05$   $\mu\text{Ci/ml}$  gross beta or gamma.
- f. If an isotopic analysis is unavailable, gross beta or gamma measurement of BATCH RELEASE may be substituted provided the concentration released to the UNRESTRICTED AREA does not exceed  $1.0E-07$   $\mu\text{Ci/ml}$  and a COMPOSITE SAMPLE is analyzed for principal gamma emitters when instrumentation is available.
- g. Frequency notation:
- P - Prior to each release.
  - M - At least once per 31 days.
  - Q - At least once per 92 days.
  - S - At least once per 12 hours (when the monitor is inoperable).

Table 2-4

Limiting Radionuclide Concentrations\* In Secondary-Side  
Clean-Up Resins for Discharges to Onsite Settling Basin

Radionuclide	Limiting Concentration** ( $\mu\text{Ci}/\text{cm}^3$ )
Cr-51	3.3E-02
Mn-54	6.2E-05
Fe-59	5.1E-04
Co-58	3.0E-04
Co-60	5.4E-06
Y-91	2.1E-03
Zr-95	4.1E-04
Nb-95	1.0E-03
Mo-99	3.5E-02
Ru-103	1.0E-03
Ru-106	1.6E-05
Ag-110m	1.6E-05
Te-125m	5.4E-05
Te-127m	1.5E-05
Te-129m	6.2E-05
Te-131m	1.1E-02
Te-132	7.4E-03
I-131	1.1E-04
I-133	3.8E-04
I-135	1.5E-03
Cs-134	1.1E-05
Cs-136	2.6E-03
Cs-137	1.0E-05
Ba-140	1.1E-02
La-140	7.4E-03
Ce-141	5.8E-03
Ce-144	4.1E-05
Pr-143	1.9E-02

\* Concentration limits based on the study, Disposal of Low-Level Radioactively Contaminated Secondary-Side Clean-up Resins in the On-site Settling Basins at the Davis-Besse Nuclear Power Station, J. Stewart Bland, May 1983. The limits represent a hypothetical maximum individual dose of less than 1 mrem per year due to an inadvertent release to the offsite environment. The allowable releases limits as presented in Table 2 of the above reference report have been reduced by a factor of 10 for added conservatism - representing a hypothetical dose of less than 0.1 mrem.

\*\* With more than one radionuclide identified in a resin batch, the evaluation for acceptable discharge to the onsite settling basin shall be based on the "sum of the fractions" rule as follows: Determine for each identified radionuclide the ratio between the measured concentration and the limiting concentration; the sum of these ratios for all radionuclides should be less than one (1) for discharge to the basin.



Table 2-5  
Davis-Besse Site-Specific Liquid Ingestion Dose Commitment Factors, A<sub>10</sub>  
(mrem/hr per  $\mu$ Ci/ml)

Nuclide	Bone	Liver	T. Body	Thyroid	Kidney	Lung	GI-LLI
H-3	0.00E+0	1.76E+0	1.76E+0	1.76E+0	1.76E+0	1.76E+0	1.76E+0
C-14	3.13E+4	6.26E+3	6.26E+3	6.26E+3	6.26E+3	6.26E+3	6.26E+3
Na-24	4.32E+2	8.64E+2	8.64E+2	4.32E+2	4.32E+2	4.32E+2	4.32E+2
P-32	1.29E+4	6.44E+4	5.37E+4	0.00E+0	0.00E+0	0.00E+0	1.56E+5
Cr-51	0.00E+0	0.00E+0	1.31E+0	7.85E+1	2.89E+1	1.74E+0	3.30E+2
Mn-54	0.00E+0	4.44E+3	8.48E+2	0.00E+0	1.32E+3	0.00E+0	1.34E+4
Mn-56	0.00E+0	1.12E+2	1.98E+1	0.00E+0	1.42E+2	0.00E+0	1.57E+3
Fe-55	6.99E+2	4.83E+2	1.13E+2	0.00E+0	0.00E+0	2.49E+2	2.77E+2
Fe-59	1.10E+3	2.59E+3	9.93E+2	0.00E+0	0.00E+0	7.24E+2	8.64E+3
Co-57	0.00E+0	2.35E+1	3.91E+1	0.00E+0	0.00E+0	0.00E+0	5.96E+2
Co-58	0.00E+0	1.00E+2	2.24E+2	0.00E+0	0.00E+0	0.00E+0	3.03E+3
Co-60	0.00E+0	2.87E+2	6.34E+2	0.00E+0	0.00E+0	0.00E+0	5.40E+3
Ni-62	3.30E+4	2.29E+3	1.11E+3	0.00E+0	0.00E+0	0.00E+0	4.78E+2
Ni-65	1.34E+2	1.74E+1	7.95E+0	0.00E+0	0.00E+0	0.00E+0	4.42E+2
Cu-64	0.00E+0	1.12E+1	5.25E+0	0.00E+0	2.82E+1	0.00E+0	9.54E+2
Zn-65	2.32E+4	7.40E+4	3.34E+4	0.00E+0	4.95E+4	0.00E+0	4.66E+4
Zn-69	4.95E+1	9.66E+1	6.58E+0	0.00E+0	6.15E+1	0.00E+0	1.42E+1
Br-82	0.00E+0	0.00E+0	2.60E+2	0.00E+0	0.00E+0	0.00E+0	2.98E+2
Br-83	0.00E+0	0.00E+0	4.10E+1	0.00E+0	0.00E+0	0.00E+0	5.91E+1
Br-84	0.00E+0	0.00E+0	5.31E+1	0.00E+0	0.00E+0	0.00E+0	4.17E+1
Br-85	0.00E+0	0.00E+0	2.18E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
Rb-86	0.00E+0	1.01E+5	4.72E+4	0.00E+0	0.00E+0	0.00E+0	2.00E+4
Rb-88	0.00E+0	2.91E+2	1.54E+2	0.00E+0	0.00E+0	0.00E+0	4.01E+9
Rb-89	0.00E+0	1.93E+2	1.35E+2	0.00E+0	0.00E+0	0.00E+0	1.12E+11
Sr-89	2.64E+4	0.00E+0	7.44E+2	0.00E+0	0.00E+0	0.00E+0	4.27E+3
Sr-90	6.55E+5	0.00E+0	1.41E+5	0.00E+0	0.00E+0	0.00E+0	1.89E+4
Sr-91	4.90E+2	0.00E+0	1.98E+1	0.00E+0	0.00E+0	0.00E+0	2.32E+3
Sr-92	1.86E+2	0.00E+0	8.04E+0	0.00E+0	0.00E+0	0.00E+0	3.68E+3
Y-90	7.16E+1	0.00E+0	1.92E+2	0.00E+0	0.00E+0	0.00E+0	7.59E+3
Y-91m	6.77E+3	0.00E+0	2.62E+4	0.00E+0	0.00E+0	0.00E+0	1.99E+2
Y-91	1.05E+1	0.00E+0	2.81E+1	0.00E+0	0.00E+0	0.00E+0	5.78E+3
Y-92	6.29E+2	0.00E+0	1.84E+3	0.00E+0	0.00E+0	0.00E+0	1.10E+3
Y-93	2.00E+1	0.00E+0	5.51E+2	0.00E+0	0.00E+0	0.00E+0	6.33E+3
Zr-95	4.84E+1	2.19E+1	1.49E+1	0.00E+0	3.44E+1	0.00E+0	4.95E+2
Zr-97	3.78E+2	7.63E+3	3.49E+3	0.00E+0	1.15E+2	0.00E+0	2.34E+3
Nb-95	4.47E+2	2.49E+2	1.34E+2	0.00E+0	2.46E+2	0.00E+0	1.51E+6
Nb-97	3.75E+0	9.48E+1	3.46E+1	0.00E+0	1.11E+0	0.00E+0	3.50E+3
Mo-99	0.00E+0	1.64E+2	3.16E+1	0.00E+0	3.76E+2	0.00E+0	3.83E+2
Tc-99m	1.25E+2	3.53E+2	4.49E+1	0.00E+0	5.35E+1	1.71E+2	2.09E+3
Tc-101	1.28E+2	1.89E+2	1.61E+1	0.00E+0	3.33E+1	5.45E+3	5.56E+14
Ru-103	7.13E+0	0.00E+0	3.07E+0	0.00E+0	2.72E+1	0.00E+0	8.32E+2
Ru-105	5.94E+1	0.00E+0	2.34E+1	0.00E+0	7.67E+0	0.00E+0	3.43E+2
Ru-106	1.06E+2	0.00E+0	1.34E+1	0.00E+0	2.05E+2	0.00E+0	4.86E+3
Rh-103m	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
Rh-106	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
Ag-110m	3.22E+0	2.00E+0	1.77E+0	0.00E+0	5.89E+0	0.00E+0	1.21E+3
Sb-124	4.76E+1	8.99E+1	1.89E+1	1.15E+1	0.00E+0	3.70E+1	1.15E+3
Sb-125	3.04E+1	3.40E+1	7.24E+0	3.09E+2	0.00E+0	2.35E+1	3.35E+2
Te-123m	2.61E+3	9.44E+2	3.49E+2	7.64E+2	1.06E+4	0.00E+0	1.04E+4
Te-127m	4.58E+3	2.35E+3	8.02E+2	1.48E+3	2.47E+4	0.00E+0	2.21E+4
Te-127	1.07E+2	3.84E+1	2.31E+1	7.93E+1	4.34E+2	0.00E+0	8.44E+3
Te-129m	1.12E+4	4.17E+3	1.77E+3	3.44E+3	4.47E+4	0.00E+0	5.63E+4
Te-129	3.05E+1	1.15E+1	7.44E+0	2.34E+1	1.28E+2	0.00E+0	2.30E+1
Te-131m	1.68E+3	8.22E+2	6.85E+2	1.30E+3	8.33E+3	0.00E+0	8.17E+4
Te-131	1.52E+1	6.00E+0	6.05E+0	1.57E+1	8.39E+1	0.00E+0	2.71E+0
Te-132	2.45E+2	1.58E+3	1.49E+3	1.75E+3	1.53E+4	0.00E+0	7.50E+4
I-130	3.62E+1	1.13E+2	4.44E+1	9.55E+3	1.74E+2	0.00E+0	9.70E+1
I-131	2.10E+2	3.01E+2	1.72E+2	9.89E+4	5.15E+2	0.00E+0	7.93E+1
I-132	1.03E+1	2.74E+1	9.60E+0	9.60E+2	4.37E+1	0.00E+0	5.15E+0
I-133	7.17E+1	1.25E+2	3.80E+1	1.63E+6	2.18E+2	0.00E+0	1.12E+2
I-134	5.35E+0	1.45E+1	5.20E+0	2.32E+2	2.31E+1	0.00E+0	1.27E+2
I-135	2.24E+1	5.86E+1	2.16E+1	3.88E+3	9.39E+1	0.00E+0	6.42E+1
Ca-134	2.99E+5	7.11E+5	5.81E+5	0.00E+0	2.30E+5	7.64E+4	1.24E+4
Ca-136	3.13E+4	1.23E+5	8.88E+4	0.00E+0	6.87E+4	9.41E+3	1.40E+4
Ca-137	3.83E+5	5.23E+5	3.43E+5	0.00E+0	1.78E+5	5.91E+4	1.01E+4
Ca-138	2.65E+2	5.23E+2	2.59E+2	0.00E+0	3.85E+2	3.80E+1	2.33E+3
Ba-139	2.35E+0	1.67E+3	4.87E+2	0.00E+0	1.56E+3	9.48E+4	4.16E+0
Ba-140	4.91E+2	6.16E+1	3.22E+1	0.00E+0	2.10E+1	3.51E+1	1.01E+3
Ba-141	1.14E+0	8.61E+4	3.84E+2	0.00E+0	6.00E+4	4.88E+4	5.37E+10
Ba-142	5.15E+1	5.29E+4	3.24E+2	0.00E+0	4.47E+4	3.00E+4	7.25E+19
La-140	1.86E+1	9.38E+2	2.48E+2	0.00E+0	0.00E+0	0.00E+0	6.89E+3
La-142	9.53E+3	4.33E+3	1.08E+3	0.00E+0	0.00E+0	0.00E+0	3.16E+1
Ce-141	1.59E+1	1.08E+1	1.22E+2	0.00E+0	5.00E+2	0.00E+0	4.11E+2
Ce-143	2.80E+2	2.07E+1	2.29E+3	0.00E+0	9.13E+3	0.00E+0	7.74E+2
Ce-144	8.29E+0	3.47E+0	4.45E+1	0.00E+0	2.04E+0	0.00E+0	2.80E+3
Pr-143	6.85E+1	2.75E+1	3.79E+2	0.00E+0	1.59E+1	0.00E+0	3.00E+3
Pr-144	2.34E+3	9.31E+4	1.14E+4	0.00E+0	5.25E+4	0.00E+0	1.22E+10
Nd-147	4.48E+1	5.41E+1	3.24E+2	0.00E+0	3.18E+1	0.00E+0	2.40E+3
Nd-147	2.97E+2	2.49E+2	8.48E+1	0.00E+0	0.00E+0	0.00E+0	8.14E+4
Nd-147	4.59E+2	4.51E+3	2.09E+3	0.00E+0	1.41E+2	0.00E+0	9.29E+2

Table 2-6

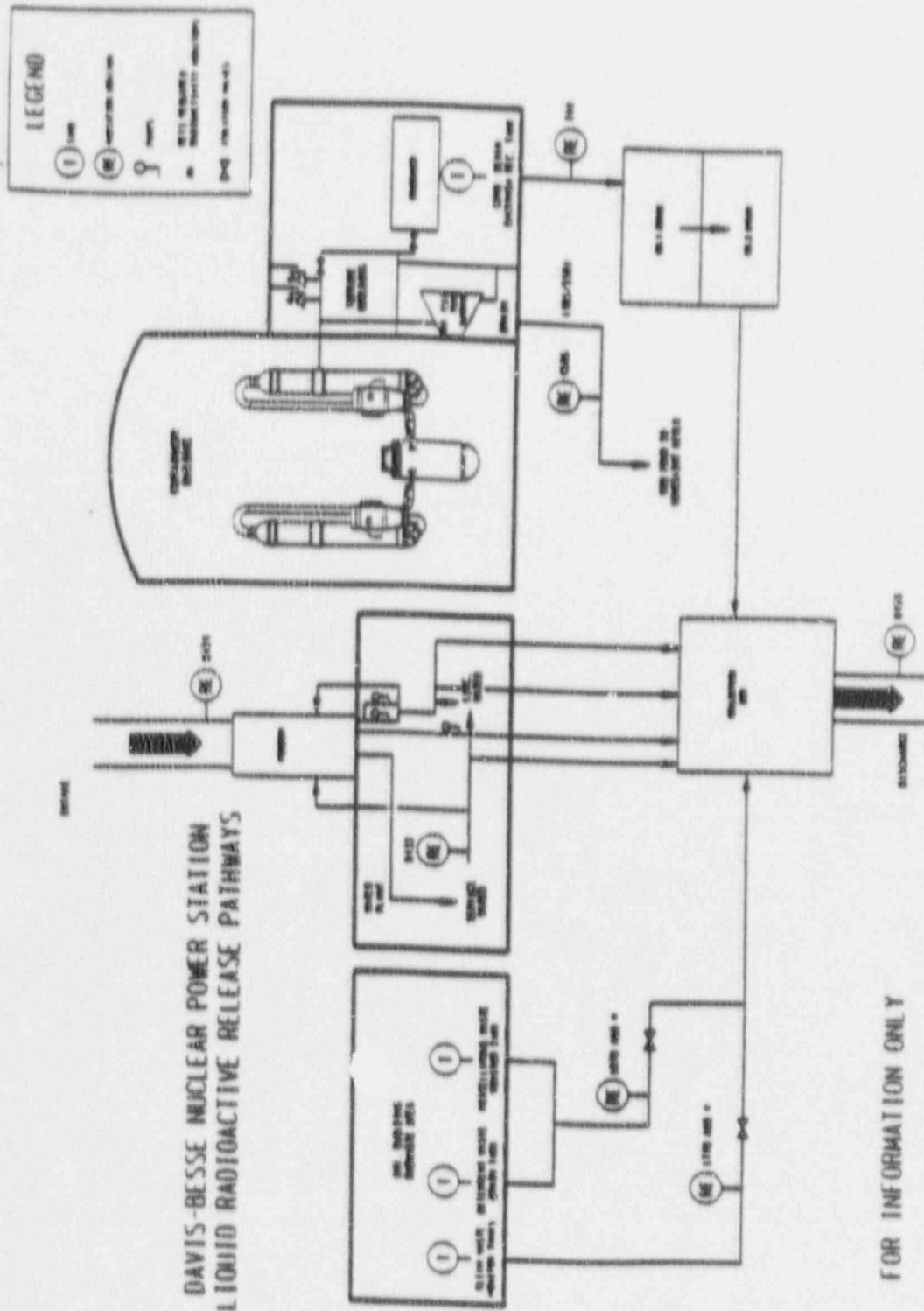
Bioaccumulation Factors (BF<sub>i</sub>)  
(pCi/kg per pCi/liter)\*

Element	Freshwater Fish
H	9.0E-01
C	4.6E+03
Na	1.0E+02
P	3.0E+03
Cr	2.0E+02
Mn	4.0E+02
Fe	1.0E+02
Co	5.0E+01
Ni	1.0E+02
Cu	5.0E+01
Zn	2.0E+03
Br	4.2E+02
Rb	2.0E+03
Sr	3.0E+01
Y	2.5E+01
Zr	3.3E+00
Nb	3.0E+04
Mo	1.0E+01
Tc	1.5E+01
Ru	1.0E+01
Rh	1.0E+01
Ag	2.3E+00
Sb	1.0E+00
Te	4.0E+02
I	1.5E+01
Cs	2.0E+03
Ba	4.0E+00
La	2.5E+01
Ce	1.0E+00
Pr	2.5E+01
Nd	2.5E+01
W	1.2E+03
Np	1.0E+01

\* Values in this Table are taken from Regulatory Guide 1.109 except for phosphorus which is adapted from NUREG/CR-1336 and silver and antimony which are taken from UCRL 50564, Rev. 1, October 1972.

Figure 2-1

Liquid Radioactive Effluent Monitoring and Processing Diagram



DB-10-12-81 DBM FILE: SC-000004.DSM

### 3.0 GASEOUS EFFLUENTS

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with methods in the ODCM Section 3.3 to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of ODCM Section 3.3 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in Section 3.3.

#### 3.1 Radiation Monitoring Instrumentation and Controls

This Section of the ODCM specifies the gaseous effluent monitoring instrumentation required at Davis-Besse for controlling and monitoring radioactive effluents as required by TS. Location and control function of these monitors are displayed in ODCM Figure 3-1.

Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 3-2. Each of these operations shall be performed within the specified time interval with a maximum allowable extension not to exceed 25 percent of the specified interval.

NOTE: The monitors specified in Table 3-2 are inoperable if surveillances are not performed or setpoints are less conservative than required.

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

With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the actions shown in Table 3-1. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Effluent and Waste Disposal Report (Section 7.2) why the inoperability was not corrected in a timely manner.



### 3.1.1 Station Vent Stack (RE-4598 AA, BA)

The Station Vent is the final release point for all gaseous radioactive effluents. The Station Vent stack monitoring system consists of a high and low range isokinetic samplers. Three separate channels (A, B and C) are provided for each monitoring system. Channel A represents a gross gamma detector viewing a fixed particulate filter sampler. Channel B is a gross gamma detector on a cartridge sampler (e.g., charcoal or Ag zeolite); and Channel C is the gross gamma detector viewing a fixed air volume measuring for noble gases. Only the Channel C radiation detector is required in order to comply with the TS requirements. Channels A and B detectors provide information on potential radioiodine and particulate releases. However, these type configurations, monitor viewing a fixed filter, experience wide variations in response due in part to the much more abundant noble gases in the effluent stream relative to the particulate or radioiodines being sampled. Therefore, while the Channels A and B provide useful information for identifying potential particulate and radioiodine releases, they are not used for quantifying the release rate as required by TS. Refer to Section 3.6.

The following sampling/monitoring instrumentation on the Station Vent is required by Table 3-1, Radioactive Gaseous Effluent Monitors - Instrumentation:

- noble gas activity monitor (Channel C)
- iodine sampler cartridge (Channel B)
- particulate sampler filter (Channel A)
- sampler flow rate measuring device
- unit vent flow rate measuring device (computer points, F883 and F885)

### 3.1.2 Waste Gas Decay System (RE-1822 A&B)

The radioactive waste gas discharge line is continuously monitored by two off-line detectors, each measuring gross activity. The monitors control function will isolate the waste discharge lines prior to exceeding the established alarm setpoint. Table 3-1 requires that the Waste Gas Decay System contain as a minimum the following instrumentation:

- noble gas activity monitor (RE-1822 A or B)
- effluent system flow rate measuring device (FT-1821 and 1821 A)

If the noble gas detector is declared inoperable, the contents of the tank may be released provided that prior to the release:

- at least two independent gas samples are collected and analyzed by gamma spectroscopy for principal gamma emitters (noble gases);
- at least two independent verifications of the release rate calculations are performed; and

- at least two independent verifications of the discharge valve line-up are performed.

If the flow rate device is inoperable, effluent releases may continue provided that the flow rate is estimated at least once per 12 hours. Flow rates may be estimated based on fan curves or discharge valve header positioning.

### 3.1.3 Containment Purge Exhaust Filter Monitor (RE-5052 A,B&C)

This detector monitors the containment atmosphere for radioactivity during Containment VENT or PURGE. The noble gas activity monitor (Channel C) is required by Table 3-1, providing an automatic isolation of the release prior to exceeding the limits of Section 3.3 and TS. Although not required in order to comply with TS, the particulate and iodine detectors (Channel A&B, respective) provide indications of increasing levels of particulate and radioiodine releases.

### 3.1.4 Hydrogen Purge Line

The hydrogen purge line serves as a Containment pressure relief route through the Station Vent. A separate radiation monitor on this line is not required. Any release will be monitored by the Station Vent monitor RE-4598.

### 3.1.5 Waste Gas System Oxygen Monitor

The Waste Gas System is provided with an oxygen monitor (with an alarm function) as required by TS 3.3.3.10 to alert operators in the unlikely event of oxygen leakage into the waste gas header. The TS requires that the concentration of oxygen be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume. An oxygen concentration above the specified limit will actuate a local and control room alarm (TS 3.11.2.5).

## 3.2 Sampling and Analysis of Gaseous Effluents

The program for sampling and analysis of gaseous waste is prescribed in this Section and incorporates the basic requirements outlined in TS. Radioactive gaseous wastes shall be sampled and analyzed according to the sampling and analysis program of Table 3-3. This table distinguishes two types of gaseous releases: (1) the waste gas decay tank release and Containment PURGE are treated as BATCH RELEASES; and (2) routine releases from building ventilation system via the Station Vent are treated as continuous releases. Containment pressure releases upon startup are considered batch releases.

### 3.2.1 Station Vent Release

All releases from the Station Vent are required to be continuously sampled for radioactivity. As specified in Table 3-3, the following samples and analysis are required:

- once per week, analysis of an absorption media (e.g., charcoal cartridge) for I-131;

- once per week, analysis of a filter sample for all principal gamma emitters (particulate radioactive material);
- once per month, analysis of a grab gas sample for all principal gamma emitters (noble gas) and tritium;
- once per month, analysis of a composite of the particulate samples of all releases for that month for gross alpha activity;
- once per quarter, analysis of a composite of the particulate samples for all releases for that month for Sr-89 and 90;
- continuous monitoring for noble gases (gross beta and gamma activity) (provided by RE-4598 AA and BA, Channel C as previously discussed in Section 3.1.1).

### 3.2.2 Waste Gas Decay Tank Release and Containment PURGE (BATCH)

Table 3-3 requires that a grab gas sample be collected and analyzed prior to each BATCH RELEASE from the Waste Gas Decay Tanks (WGDT) or a Containment PURGE. The analysis shall include the identification of all principal gamma emitters (noble gas) and tritium.

For a planned Containment PURGE, the results of the sample and analysis are used to establish the acceptable release rate and radiation monitor alarm setpoint in accordance with ODCM Section 3.3. This evaluation is necessary to ensure compliance with the dose rate limits of Section 3.3.1.

## 3.3 Gaseous Effluent Monitor Setpoint Determination

### 3.3.1 Station Vent

All releases of gaseous radioactive effluents are via the Station Vent. TS requires that alarm setpoints shall be established for the gaseous effluent monitoring instrumentation to ensure that the release rate of effluents does not exceed the following limits.

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- for noble gas: less than or equal to 500 mrem/year to the total body and less than or equal to 3000 mrem/year to the skin, and
- iodine-131, tritium and all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to 1500 mrem/year to any organ. (The evaluation of the release rate of the radioiodines and particulates is based on the weekly continuous samplers. Refer to ODCM Section 3.6).

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

This requirement is provided to ensure that the dose at the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that

radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC outside the SITE BOUNDARY to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(a)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release limits restrict the corresponding gamma and beta doses above background to an individual at or beyond the UNRESTRICTED AREA boundary to <500 mrem/year to the total body or to <3000 mrem/year to the skin. These release limits also restrict, at all times, the corresponding thyroid doses above background to <1500 mrem/year via the inhalation pathway to <1500 mrem/year.

From a grab sample analysis of the applicable source (i.e., Station Vent, Waste Gas Decay Tanks, or Containment atmosphere), the radiation monitoring alarm setpoint may be established by the following calculational method.

$$SP_{TB} = CR_{TB} * 500 \quad (3-1)$$

$$SP_s = CR_s * 3000 \quad (3-2)$$

where:

$SP_{TB}$  = limiting concentration in the effluent stream (i.e., setpoint of the monitor) corresponding to the release rate limit for the total body dose rate of 500 mrem per year ( $\mu\text{Ci}/\text{ml}$ )

$SP_s$  = limiting concentration in the effluent stream (i.e., setpoint of the monitor) corresponding to the release rate limit for the skin dose rate of 3000 mrem per year ( $\mu\text{Ci}/\text{ml}$ )

$CR_{TB}$  = monitor response corresponding to a dose rate of one (1) mrem/year, total body ( $\mu\text{Ci}/\text{ml}$  per mrem/yr, total body)

$CR_s$  = monitor response corresponding to a dose rate of one (1) mrem/yr, skin ( $\mu\text{Ci}/\text{ml}$  per mrem/yr, skin)

500 = total body dose rate limit (mrem/yr)

3000 = skin dose rate limit (mrem/yr)

The value for CR (monitor response corresponding to a dose rate of 1 mrem per year) is dependent on the radionuclide distribution. Based on the measured distribution CR is calculated by the equations:

$$CR_{TB} = \frac{I \text{ Ci}}{1.67E+01 * X/Q * VF * I(C_1 * K_1)} \quad (3-3)$$



and

$$CR_i = \frac{I C_i}{1.67E+01 * X/Q * VF * I(C_i * (L_i + 1.1 M_i))} \quad (3-4)$$

where:

$X/Q$  = annual average meteorological dispersion to the controlling site boundary location from Table 3-4 (sec/m)

$VF$  = ventilation system flow rate for the applicable release point and monitor (liters/minute)

$C_i$  = concentration of noble gas radionuclide (i) as determined by gamma spectral analysis of grab sample ( $\mu\text{Ci}/\text{ml}$ )

$K_i$  = total body dose conversion factor for noble gas radionuclide (i) in mrem/yr per  $\mu\text{Ci}/\text{m}^3$  (from Table 3-5)

$L_i$  = beta skin dose conversion factor for noble gas radionuclide (i) in mrem/yr per  $\mu\text{Ci}/\text{m}^3$  (from Table 3-5)

$M_i$  = gamma air dose conversion factor for noble gas radionuclide (i) in mrad/yr per  $\mu\text{Ci}/\text{m}^3$  (from Table 3-5)

1.1 = mrem skin dose per mrad gamma air dose (mrem/mrad)

1.67E+01 =  $1E+03 \text{ (ml/l)} * (1/60) \text{ (min/sec)}$

The more limiting value (i.e., lower of the two values for  $SP_i$  and  $SP_j$ ) as calculated above is used for establishing the monitor setpoint. The Station Vent monitors (RE-4598) sensitivities and read outs are in  $\mu\text{Ci}/\text{ml}$ ; however, the Containment Purge Exhaust Monitors (RE-5052) and the WGD monitors (RE-1822) sensitivities and read outs are in counts per minute. Therefore, for RE-5052 and RE-1822, the setpoints in  $\mu\text{Ci}/\text{ml}$  must be corrected to an equivalent monitor count per minute. The monitor calibration curves are used for determining specific radionuclide sensitivities (cpm/ $\mu\text{Ci}/\text{ml}$ ). Otherwise, the monitor sensitivity for Xe-133 may be used in lieu of the sensitivity values for the individual radionuclides. Because of its lower gamma energy and corresponding monitor response, the Xe-133 sensitivity provides a conservative value for alarm setpoint determination.

### 3.3.2 Conservative, Generic Alarm Setpoints

Conservative alarm setpoints may be established, in lieu of the individual radionuclide evaluation as described above. This approach eliminates the need to adjust the setpoint periodically to reflect minor changes in radionuclide distribution or release flow rate. The alarm setpoint may be conservatively determined based on an assumed Kr-89 release. The Kr-89 total body dose conversion factor is the most limiting.

Therefore, the more restrictive setpoint is based on the total body dose rate limit and may be calculated using equations (3-1) and (3-3). Again, the Xe-133 sensitivity is used for conservatism. The alarm setpoint is established in accordance with the Radiation Monitor Setpoint Manual.

### 3.3.3 Gaseous Effluent Alarm Response - Evaluating Actual Release Conditions

The monitor alarm setpoint is used as the primary method for ensuring and demonstrating compliance with the release rate limits of Section 3.3.1. Not exceeding alarm setpoints constitutes a demonstration that release rates have been maintained within the limits. When an effluent noble gas monitor exceeds the alarm setpoint, an evaluation of compliance with the release rate limits must be performed using actual release conditions. This evaluation requires collecting a sample of the effluent to establish actual radionuclide concentrations and monitor response. The following equations may be used for evaluating compliance with the release rate limit of Section 3.3.1 for noble gases.

$$\dot{D}_{tb} = 1.67E+01 * X/Q * VF * I(K_i * C_i) \quad (3-5)$$

$$\dot{D}_s = 1.67E+01 * X/Q * VF * I((L_i + 1.1 M_i) * C_i) \quad (3-6)$$

where:

$\dot{D}_{tb}$  = total body dose rate (mrem/yr)

$\dot{D}_s$  = skin dose rate (mrem/yr)

$X/Q$  = atmospheric dispersion to the controlling SITE BOUNDARY location from Table 3-6 (sec/m<sup>3</sup>)

VF = ventilation system release rate (liters/min)

$C_i$  = concentration of radionuclide (i) as measured in the grab sample (μCi/ml)

$K_i$  = total body dose conversion factor for noble gas radionuclide (i) in mrem/yr per μCi/m<sup>3</sup>, from Table 3-5

$L_i$  = beta skin dose conversion factor for noble gas radionuclide (i) in mrem/yr per μCi/m<sup>3</sup>, from Table 3-5

$M_i$  = gamma air dose conversion factor for noble gas radionuclide (i) in mrad/yr per μCi/m<sup>3</sup>, from Table 3-5

1.1 = mrem skin dose per mrad gamma air dose (mrem/mrad)

1.67E+01 = 1E+03 (ml/l) \* (1/60) (min/sec)

### 3.4 Release Rate Evaluation - Waste Gas Decay Tank Releases and Containment PURGE

For a Waste Gas Decay Tank release or a Containment PURGE, an evaluation of acceptable release rate shall be performed prior to the release. Based on the measured noble gas concentration in the grab sample collected per the requirements of Table 3-3, the allowable release rate can be calculated by the following equation:

$$RR_{tb} = \frac{500}{1.67E+01 * X/Q * I(K_1 * C_1)} \quad (3-7)$$

or

$$RR_s = \frac{3000}{1.67E+01 * X/Q * I((L_1 + 1.1 M_1) * C_1)} \quad (3-8)$$

where:

$RR_{tb}$  = allowable release rate so as not to exceed a dose rate of 500 mrem/yr, total body (liters/minute)

$RR_s$  = allowable release rate so as not to exceed a dose rate of 3000 mrem/yr, skin (liters/minute)

500 = total body dose rate limit (mrem/yr); Section 3.3.1 noble gas

3000 = skin dose rate limit (mrem/yr); Section 3.3.1 noble gas

The lesser value ( $RR_{tb}$  or  $RR_s$ ) as calculated above should be used for establishing the allowable release rate for the PURGE or WGDY release.

### 3.5 Quantifying Releases - Noble Gases

The determination of doses in the environment from releases is dependent on the mixture of the radioactive material. Also, NRC Regulatory Guide 1.21 requires reporting of individual radionuclides released in gaseous effluents. Therefore, the quantities of the individual radionuclides released in the gaseous effluents must be determined.

#### 3.5.1 Quantifying Releases Using Station Vent Noble Gas Monitor (RE-4598C)

The quantification of the continuous gaseous effluents (noble gases) is based on the sampling and analysis of the Station Vent effluent. The monitor provides a measurement of gross radioactive material concentration in the effluent. As required by Table 3-3, a gas sample is collected at least monthly from the Station Vent. And, as discussed in ODCM Section 3.2.2, this gas sample is analyzed by gamma spectroscopy to identify principal gamma emitting radionuclides (noble gases). The results of the sample analysis may be used to determine the radionuclide releases. This simplified approach reasonably quantifies the continuous release provided that no atypical levels have been observed (e.g., alert setpoint being exceeded).

Based on the average noble gas monitor reading over the release period, the individual noble gas radionuclide releases are quantified by the equation:

$$Q_i = 1.0E+03 * \frac{A_i}{I A_i} * C * VF * T \quad (3-9)$$

where:

- $Q_i$  = total activity released of radionuclide (i) ( $\mu\text{Ci}$ )
- $A_i$  = activity of radionuclide (i) from the gamma spectral analysis of the grab sample from the release point ( $\mu\text{Ci}$ )
- $C$  = average gross activity concentration over the release period as measured by the noble gas monitor, excluding any BATCH RELEASES ( $\mu\text{Ci}/\text{ml}$ )
- $VF$  = ventilation system flow rate (liters/min)
- $T$  = total time of the release period (min)
- $1.0E+03$  = milliliters per liter

### 3.5.2 Quantifying Releases with Inoperable Monitors

With an inoperable radiation monitor on the Station Vent (i.e., the RE-4598, Channel C), the once-per-8 hours grab samples provide the mechanism for the continued control and quantification of releases in accordance with TS requirements. Analysis of grab samples provides the radioactive material concentrations in the effluent. The flow measurement device, or flow estimate, and the release duration provide the total volume released. With these, the release rate and resultant total amount of radioactive material released can be determined.

- a. Release Rate Evaluation. With an inoperable monitor, the demonstration of compliance with the release rate limit of Section 3.3.1 for noble gases must be based on the periodic grab samples. These grab samples provide a measurement of the noble gas concentration in the effluent stream. Equations (3-5) and (3-6) can be used for demonstrating that the measured release rate and corresponding calculated dose rate are within the limits.
- b. Total Release Evaluation. The grab samples are also used to quantify the total releases. The measured noble gas radionuclide concentrations in the grab samples are considered representative of the average effluent concentrations over the release period (i.e., elapsed time since last sample). The following equation may be used for determining the release quantities from any release point based on the grab sample analysis:

$$Q_i = 1.0E+03 * VF * T * C_i \quad (3-10)$$



where:

$Q_i$  = total activity released of radionuclide (i) ( $\mu\text{Ci}$ )

VF = ventilation system release rate (l/min)

T = total time of release period (min)

1.0E+03 = milliliters per liter

$C_i$  = concentration of radionuclide (i) as measured in the grab sample ( $\mu\text{Ci/ml}$ )

### 3.6 SITE BOUNDARY Dose Rate - Radioiodine and Particulates

Section 3.3.1 limits the dose rate to  $\leq 1500$  mrem/yr to any organ for gaseous releases of I-131, tritium and all particulates with half-lives greater than 8 days. To demonstrate compliance with this limit, an evaluation is performed at a frequency no greater than that corresponding to the sampling and analysis time period (nominally once per 7 days). The following equation may be used for the dose rate evaluation:

$$\dot{D}_o = X/Q * \sum (R_i * \dot{Q}_i) \quad (3-11)$$

where:

$\dot{D}_o$  = dose rate to organ (o) over the sampling time period (mrem/yr)

$X/Q$  = atmospheric dispersion to the controlling SITE BOUNDARY location for the inhalation pathway from Table 3-6 ( $\text{sec/m}^3$ )

$R_i$  = dose parameter for radionuclide (i) for the controlling age group via the inhalation pathway from Table 3-7 (mrem/yr per  $\mu\text{Ci/m}^3$ )

$\dot{Q}_i$  = average release rate over the appropriate sampling period and analysis frequency for radionuclide (i), that is I-131, tritium or other radionuclide in particulate form with half-lives greater than 8 days ( $\mu\text{Ci/sec}$ )

#### 3.6.1 Simplified Dose Rate Evaluation for Radioiodines and Particulates

It is conservative to perform a simplified evaluation of allowable releases by applying the I-131 dose factor to the collective releases for all measured radionuclides. By substituting 1500 mrem/yr for  $\dot{D}_o$  and solving for  $\dot{Q}$ , an allowable release rate can be determined. Based on the annual average meteorological dispersion (see Table 3-6) and the dose factor for the most limiting potential pathway, age group and organ (inhalation, child, thyroid --  $R = 1.62\text{E}+07$  mrem/yr per  $\mu\text{Ci/m}^3$ ), the allowable release rate (based on I-131) is 44.1  $\mu\text{Ci/sec}$ . An added conservatism multiplier of 0.8 has been included in this calculation to account for any potential dose contribution from other radioactive particulate material. For a 7-day period, which is the nominal sampling and analysis frequency, the cumulative release would be 26.7 Ci. Therefore, as long as the releases in any 7-day period do not exceed 26.7 Ci, no additional analyses are needed to verify compliance with the Section 3.3.1 limits on allowable release rate.

### 3.7 Noble Gas Effluent Dose Calculations - 10 CFR 50

#### 3.7.1 UNRESTRICTED AREA Dose - Noble Gases

Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in this Section at least once per 31 days. This periodic assessment of releases of noble gases is to evaluate compliance with the quarterly dose limits and calendar year limits.

The air dose due to noble gases released in gaseous effluents to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- during any calendar quarter: less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and
- during any calendar year: less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Section 7.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

This specification is provided to implement the requirements of Section II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The limits specified above provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." This Section implements the requirements of Section III.A of Appendix I that conformance with the guides of Appendix I to be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977.

The following equations may be used to calculate the gamma-air and beta-air doses:

$$D_Y = 3.17E-08 * X/Q * I (M_i * Q_i) \quad (3-12)$$

$$D_B = 3.17E-08 * X/Q * I (N_i * Q_i) \quad (3-13)$$

where:

$D_\gamma$  = air dose due to gamma emissions for noble gas radionuclides (mrad)

$D_\beta$  = air dose due to beta emissions for noble gas radionuclides (mrad)

$X/Q$  = atmospheric dispersion to the controlling SITE BOUNDARY location (sec/m<sup>3</sup>, from Table 3-6)

$Q_i$  = cumulative release of noble gas radionuclide (i) over the period of interest ( $\mu\text{Ci}$ )

$M_i$  = air dose factor due to gamma emissions from noble gas radionuclide (i) (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ , from Table 3-5)

$N_i$  = air dose factor due to beta emissions from noble gas radionuclide (i) (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ , from Table 3-5)

$$3.17\text{E-}08 = 1/3.15\text{E}+07 \text{ (yr/sec)}$$

### 3.7.2 Simplified Dose Calculation for Noble Gases

In lieu of the individual noble gas radionuclide dose assessment presented above, the following simplified dose calculational equations may be used for verifying compliance with the dose limits of Section 3.7.1. (Refer to Appendix B for the derivation and justification of this simplified method.)

$$D_\gamma = 2.0 * 3.17\text{E-}08 * X/Q * M_{eff} * \sum Q_i \quad (3-14)$$

and

$$D_\beta = 2.0 * 3.17\text{E-}08 * X/Q * N_{eff} * \sum Q_i \quad (3-15)$$

where:

$M_{eff}$  =  $5.7\text{E}+02$ , effective gamma-air dose factor from Appendix B (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ )

$N_{eff}$  =  $1.1+03$ , effective beta-air dose factor from Appendix B (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ )

2.0 = conservatism factor to account for potential variability in the radionuclide distribution

### 3.8 Radioiodine and Particulate Dose Calculations - 10 CFR 50

#### 3.8.1 UNRESTRICTED AREA Dose - Radioiodine and Particulates

A periodic assessment is required to evaluate compliance with the quarterly dose limit and the calendar year limit to any organ. Cumulative dose contributions for the current calendar quarter and current calendar year for I-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in this Section at least once per 31 days.

The dose to a MEMBER OF THE PUBLIC from I-131, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- During any calendar quarter: less than or equal to 7.5 mrem to any organ, and
- During any calendar year: less than or equal to 15 mrem to any organ.

With the calculated dose from the release of iodine-131, tritium and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Section 7.3, a Special report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

This requirement is provided to implement the requirements of Section II.C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The limits are the guides set forth in Section II.C of Appendix I. The actions specified provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in this Section implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedure based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977.



The release rate specifications for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the UNRESTRICTED AREA. The pathways which are examined in the development of these calculations are:

- individual inhalation of airborne radionuclides,
- deposition of radionuclides into green leafy vegetation with subsequent consumption by man,
- deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and
- deposition on the ground with subsequent exposure of man.

The following equation may be used to evaluate the maximum organ dose due to releases of iodine-131, tritium and particulates with half-lives greater than 8 days:

$$D_{\text{aop}} = 3.17\text{E-}08 * W * SF_p * \sum (R_{i_o} * Q_i) \quad (3-16)$$

Where:

$D_{\text{aop}}$  = dose or dose commitment via controlling pathway (p) and age group (a) (as identified in table 3-6) to organ (o), including the total body (mrem)

W = atmospheric dispersion parameter to the controlling location(s) as identified in table 3-6

W =  $X/Q$  atmospheric dispersion for inhalation pathway and H-3 dose contribution via other pathways ( $\text{sec}/\text{m}^3$ )

W =  $D/Q$ , atmospheric deposition for vegetation, milk and ground plane exposure pathways ( $\text{m}^{-2}$ )

$R_{i_o}$  = dose factor for organ (o) for radionuclide (i), (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) or ( $\text{m}^2$  - mrem/yr per  $\mu\text{Ci}/\text{sec}$ ) from Table 3-7 for each age group (a) and the applicable pathway (p) as identified in Table 3-6. Values for  $R_{i_o}$  were derived in accordance with the methods described in NUREG-0133.

$Q_i$  = cumulative release over the period of interest for radionuclide (i) — I-131 or radioactive material in particulate form with half-lives greater than 8 days ( $\mu\text{Ci}$ ).

$SF_p$  = annual seasonal correction factor to account for the fraction of the year that the applicable exposure pathway does not exist.

1) For milk and vegetation exposure pathways: A six month fresh vegetation and grazing season (May through October) limits exposure through this pathway to half the year = 0.5

2) For inhalation and ground plane exposure pathways: = 1.0

$$3.17\text{E-}08 = 1/3.15\text{E+}07 \text{ (yr/sec)}$$



The age group with the highest potential dose via the controlling pathway should be used for evaluating the maximum exposed individual. This determination is based on a comparison of the age group pathway dose conversion factors (Table 3-7). Only the controlling age group and pathway identified in Table 3-6 need be evaluated for compliance with the limits of Section 6.8.1.

### 3.8.2 Simplified Dose Calculation for Radioiodines and Particulates

In lieu of the individual radionuclide (I-131 and particulates) dose assessment presented above, the following simplified dose calculation may be used for verifying compliance with the dose limits of Section 3.8.1.

$$D_{max} = 3.17E-08 * W * SF_p * R_{I-131} * IQ_i \quad (3-17)$$

where:

$D_{max}$  = maximum organ dose (mrem)

$R_{I-131}$  = I-131 dose parameter for the thyroid for the identified controlling pathway

= 4.76E+10, child thyroid dose parameter for the vegetable pathway (m - mrem/yr per  $\mu$ Ci/sec)

The ground plane exposure and inhalation pathways need not be considered when the above simplified calculational method is used because of the overall negligible contribution of these pathways to the total thyroid dose. It is recognized that for some particulate radionuclides (e.g., Co-60 and Cs-137), the ground exposure pathway may represent a higher dose contribution than either the vegetation or milk pathway. However, use of the I-131 thyroid dose factor for all radionuclides will maximize the organ dose calculation, especially considering that no other radionuclide has a higher dose factor for any organ via any pathway than I-131 for the thyroid via the vegetable or milk pathway.

The location of exposure pathways (critical receptors) and the corresponding maximum organ dose calculation should be based on the pathways identified by the annual land use census (Section 5.0) and as identified in Table 3-6.

### 3.9 Gaseous Effluent Dose Projection

As with liquid effluents, gaseous effluents require "processing" if the projected dose exceeds specified limits. This requirement implements the requirements of 10 CFR 50.36a on maintaining and using the appropriate radwaste processing equipment to keep releases ALARA.

The GASEOUS RADWASTE TREATMENT SYSTEM (i.e., Waste Gas Decay Tank) shall be used to reduce noble gas levels prior to discharge when the projected air dose due to gaseous effluent releases to areas at and beyond the SITE BOUNDARY would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation in a 31 day period (i.e., one quarter of the design objective rate).

The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioiodine and particulate effluents, prior to their discharge, when the projected dose due to gaseous effluents releases to areas at or beyond the SITE BOUNDARY would exceed 0.3 mrem to any organ in a 31 day period. Figure 3-1 presents the gaseous effluent release points and the GASEOUS RADWASTE and VENTILATION EXHAUST TREATMENT SYSTEMS applicable for reducing effluents prior to release.

With the gaseous waste being discharged without treatment and in excess of the limits, in lieu of a Licensee Event Report prepare and submit to the commission within 30 days, pursuant to Section 7.3 a Special Report that includes the following information:

- Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reasons for the inoperability,
- Actions taken to restore the inoperable equipment to OPERABLE status, and
- Summary description of actions(s) taken to prevent a recurrence.

The requirements that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This requirement implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

If the GASEOUS RADWASTE and VENTILATION EXHAUST TREATMENT SYSTEMS are not being used, dose projections shall be performed at least once per 31 days using the following equations:

$$D_{\gamma_p} = D_{\gamma} * (31/d) \quad (3-18)$$

$$D_{\beta_p} = D_{\beta} * (31/d) \quad (3-19)$$

$$D_{maxp} = D_{max} * (31/d) \quad (3-20)$$

where:

$D_{\gamma_p}$  = gamma-air dose projection for current 31 day projection (mrad)

$D_{\gamma}$  = gamma-air dose to date for current calendar quarter (mrad)

$D_{\beta_p}$  = beta-air dose projection for current 31 day projection (mrad)

$D_{\beta}$  = beta-air dose to date for current calendar quarter (mrad)

$D_{max}$  = maximum organ dose projection for current 31 day  
projection (mrem)

$D_{max}$  = maximum organ dose to date for current calendar quarter as  
determined by equation (3-16) or (3-17) (mrem)

$d$  = number of days to date in current calendar quarter

31 = number of days in projection

- $D_{exp}$  = maximum organ dose projection for current 31 day projection (mrem)
- $D_{max}$  = maximum organ dose to date for current calendar quarter as determined by equation (3-16) or (3-17) (mrem)
- $d$  = number of days to date in current calendar quarter
- 31 = number of days in projection



Table 3-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABILITY</u>	<u>PARAMETER</u>	<u>ACTION</u>
1. Waste Gas Decay System (provides automatic isolation)				
a. Noble Gas Activity Monitor	1	(1)	Radioactivity Measurement	A
b. Effluent System Flow Rate Measuring Device	1	(1)	System Flow Rate Measurement	B
2. Waste Gas System (provides alarm function)				
a. Oxygen Monitor	1	(2)	% Oxygen	D
3. Containment Purge Monitoring System (provides automatic isolation)				
a. Noble Gas Activity Monitor	1	(1)	Radioactivity measurement	C



TABLE 3-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABILITY</u>	<u>PARAMETER</u>	<u>ACTION</u>
4. Station Vent Stack (provides alarm function)				
a. Noble Gas Activity Monitor	1	(1)	Radioactivity Measurement	C
b. Iodine Sampler Cartridge	1	(1)	Verify Presence of Cartridge	E
c. Particulate Sampler Filter	1	(1)	Verify Presence of Filter	E
d. Effluent System Flow Rate Measuring Device	1	(1)	System Flow Rate Measurement	B
e. Sampler Flow Rate Measuring Device	1	(1)	Sampler Flow Rate Measurement	B

TABLE 3-1 (Continued)

TABLE NOTATION

(1) During radioactive waste gas releases via this pathway.

(2) During additions to the waste gas surge tank

- ACTION A With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, the contents of the tank may be released to the environment provided that prior to initiating the release:
1. At least two independent samples are analyzed in accordance with Table 3-3 for analyses performed with each batch;
  2. At least two independent verifications of the release rate calculations are performed;
  3. At least two independent verifications of the discharge valving are performed.
- ACTION B With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 12 hours.
- ACTION C With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.
- ACTION D With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, additions to the waste gas surge tank may continue provided another method for ascertaining oxygen concentrations, such as grab sample analysis, is implemented to provide measurements at least once per four(4) hours during degassing and daily during other operations.
- ACTION E With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment, as required in Table 3-3.

TABLE 3-2

## RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>SOURCE CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>
1. Waste Gas Decay System				
a. Noble Gas Activity Monitor	P <sup>(1)</sup>	P	R <sup>(5)</sup>	Q <sup>(3)</sup>
b. Effluent System Flow Rate	P <sup>(1)</sup>	N/A	R	Q
2. Containment Purge Vent System				
a. Noble Gas Activity Monitor	D <sup>(1)</sup>	P <sup>(7)</sup> ; M <sup>(8)</sup>	R <sup>(5)</sup>	Q <sup>(3)</sup>
3. Station Vent Stack				
a. Noble Gas Activity Monitor	D <sup>(1)</sup>	M	R <sup>(5)</sup>	Q <sup>(4)</sup>
b. Iodine Sampler	W <sup>(1)</sup>	N/A	N/A	N/A
c. Particulate Sampler	W <sup>(1)</sup>	N/A	N/A	N/A
d. System Effluent Flow Rate Measurement Device	D <sup>(1)</sup>	N/A	R	N/A
e. Sampler Flow Rate Measurement Device	W <sup>(1)</sup>	N/A	R	N/A

TABLE 3-2 (Continued)

TABLE NOTATION

- (1) During radioactive waste gas releases via this pathway.
- (2) During additions to the waste gas surge tank.
- (3) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint.
- (4) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint.
- (5) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per eighteen months. For high range monitoring instrumentation, where calibration with a radioactive source is impractical, an electronic calibration may be substituted for the radiation source calibration.
- (6) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:
  - 1. One volume percent oxygen, balance nitrogen; and
  - 2. Four volume percent oxygen, balance nitrogen.
- (7) During containment purges.
- (8) When used in a continuous mode.
- (P) Prior to each release.
- (R) At least once per 18 months (550 days).
- (Q) At least once per 92 days.
- (D) At least once per 24 hours.
- (M) At least once per 31 days.
- (W) At least once per 7 days.

TABLE 3-3

## RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ( $\mu\text{Ci/ml}$ ) <sup>1</sup>
Waste Gas Decay	P Each Release Grab Sample	P Each Release	Principal Gamma Emitters <sup>C</sup>	1.0E-04
			H-3	1.0E-06
Containment Purge	P Each Purge Grab Sample	P Each Purge	Principal Gamma Emitters <sup>C</sup>	1.0E-04
			H-3	1.0E-06
Station Vent Stack	M Grab Sample	M	Principal Gamma Emitters <sup>C</sup>	1.0E-04
			H-3	1.0E-06
	Continuous <sup>b</sup>	W Charcoal Sample	I-131	1.0E-12
	Continuous <sup>b</sup>	W Particulate Sample	Principal Gamma Emitters <sup>C</sup>	1.0E-11
	Continuous <sup>b</sup>	M Composite Particulate Sample	Gross Alpha	1.0E-11
	Continuous <sup>b</sup>	Q Composite Particulate Sample	Sr-89, Sr-90	1.0E-11
	Continuous <sup>b</sup>	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1.0E-06



TABLE 3-3 (Continued)

TABLE NOTATION

- A. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radio-chemical separation):

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

where

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

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

E is the counting efficiency (as counts per transformation);

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

2.22 is the number of transformations per minute per picocurie;

Y is the fractional radiochemical yield (when applicable);

$\lambda$  is the radioactive decay constant for the particular radionuclide;

$\Delta t$  for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

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

- b. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Sections 3.3.1 and 3.8.

TABLE 3-3 (Continued)

TABLE 1 STATION

- c. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measured and identified, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should be reported as "less than" the nuclide's LLD and should not be reported as being present at the LLD level for the nuclide. The "less than" values shall not be used in the required dose calculations. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent and Waste Disposal Report.
- d. Frequency notation
- P - Prior to each release.
  - M - At least once per 31 days.
  - W - At least once per 7 days.
  - Q - At least once per 92 days.

Table 3-4

## Land-Use Census Summary

## Pathway Locations and Atmospheric Dispersion Parameters

<u>Sector</u>	<u>Distance (meters)</u>	<u>Pathways</u>	<u>Age Group</u>	<u>X/Q (sec/m<sup>3</sup>)</u>	<u>D/Q (m<sup>-2</sup>)</u>
N	880**	inhalation	child	9.15E-07	8.4E-09
NNE	870	inhalation	child	1.27E-06	1.47E-08
NE	900	inhalation	child	1.26E-06	1.58E-08
ENE*	---	---	---	---	---
E*	---	---	---	---	---
ESE*	---	---	---	---	---
SE*	---	---	---	---	---
SSE	2,900**	vegetation	child	6.80E-08	7.90E-10
S	1,450**	vegetation	child	1.21E-07	2.46E-09
SSW	2,180**	vegetation	child	6.45E-08	1.19E-09
SW	1,340**	vegetation	child	2.10E-07	3.94E-09
WSW	4,270**	cow/milk	infant	5.71E-08	5.31E-10
W	1,050**	vegetation	child	5.72E-07	8.87E-09
WNW	3,290**	vegetation	child	6.28E-08	5.18E-10
NW	2,040**	vegetation	child	8.25E-08	7.28E-10
NNW	1,210**	vegetation	child	2.70E-07	1.92E-09

\* Since these sectors are located over marsh areas and Lake Erie, no ingestion or inhalation pathways are present.

\*\* These values are a change to this table as a result of the 1990 Land Use Census.

Table 3-5  
Dose Factors for Noble Gases\*

Nuclide	Total Body Gamma Dose Factor K (mrem/yr per $\mu\text{Ci}/\text{m}^3$ )	Skin Beta Dose Factor L (mrem/yr per $\mu\text{Ci}/\text{m}^3$ )	Gamma Air Dose Factor M (mrad/yr per $\mu\text{Ci}/\text{m}^3$ )	Beta Air Dose Factor N (mrad/yr per $\mu\text{Ci}/\text{m}^3$ )
Kr-83m	7.56E-02	-----	1.93E+01	2.88E+02
Kr-85m	1.17E+03	1.46E+03	1.23E+03	1.97E+03
Kr-85	1.61E+01	1.34E+03	1.72E+01	1.95E+03
Kr-87	5.92E+03	9.73E+03	6.17E+03	1.03E+04
Kr-88	1.47E+04	2.37E+03	1.52E+04	2.93E+03
Kr-89	1.66E+04	1.01E+04	1.73E+04	1.06E+04
Kr-90	1.56E+04	7.29E+03	1.63E+04	7.83E+03
Xe-131m	9.15E+01	4.76E+02	1.56E+02	1.11E+03
Xe-133m	2.51E+02	9.94E+02	3.27E+02	1.48E+03
Xe-133	2.94E+02	3.06E+02	3.53E+02	1.05E+03
Xe-135m	3.12E+03	7.11E+02	3.36E+03	7.39E+02
Xe-135	1.81E+03	1.86E+03	1.92E+03	2.46E+03
Xe-137	1.42E+03	1.22E+04	1.51E+03	1.27E+04
Xe-138	8.83E+03	4.13E+03	9.21E+03	4.75E+03
Ar-41	8.84E+03	2.69E+03	9.30E+03	3.28E+03

\* Dose factors taken from NRC Regulatory Guide 1.109



Table 3-6  
Controlling Locations, Pathways and Atmospheric Dispersion  
for Dose Calculations

ODCM Section	Location	Pathway(s)	Controlling Age Group	Atmospheric Dispersion	
				$X/Q$ ( $\text{sec}/\text{m}^3$ )	$D/Q$ ( $\text{m}^{-2}$ )
3.3.1 Noble Gas	SITE BOUNDARY NNE	Noble Gases direct exposure	N/A	1.83E-06	N/A
3.3.1 Radioiodine & Particulates	SITE BOUNDARY NNE	inhalation	child	1.68E-06	N/A
3.7.1	SITE BOUNDARY NNE	gamma-air beta-air	N/A	1.83E-06	N/A
3.8.1	COW 4270m, WSW	COW milk	infant	5.71E-08	5.31E-10

NOTES:

1. All meteorological dispersion values have been taken from Stone and Webster report, Handbook for ODCM X/Q and D/Q Calculations, October 1983.
2. The noble gas, direct exposure X/Qs are based on the decayed, undepleted values.
3. The inhalation pathway X/Qs are based on the decayed, depleted values.



Table 3-7  
Gaseous Effluent Pathway Dose Commitment Factors  
R<sub>10</sub>: Inhalation Pathway Dose Factors - ADULT  
(mrem/yr per  $\mu\text{Ci}/\text{m}^3$ )

Radionuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T-Body
H-3	-	1.26E+3	1.26E+3	1.26E+3	1.26E+3	1.26E+3	1.26E+3
C-14	1.82E+4	3.41E+3	3.41E+3	3.41E+3	3.41E+3	3.41E+3	3.41E+3
Na-24	1.02E+4	1.02E+4	1.02E+4	1.02E+4	1.02E+4	1.02E+4	1.02E+4
P-32	1.32E+6	7.71E+4	-	-	-	8.64E+4	5.01E+4
Cr-51	-	-	5.95E+1	2.28E+1	1.44E+4	3.32E+3	1.00E+2
Mn-54	-	3.96E+4	-	9.84E+3	1.40E+6	7.74E+4	6.30E+3
Mn-56	-	1.24E+0	-	1.30E+0	9.44E+3	2.02E+4	1.83E+1
Fe-55	2.46E+4	1.70E+4	-	-	7.21E+4	6.03E+3	3.94E+3
Fe-59	1.18E+4	2.78E+4	-	-	1.02E+6	1.88E+5	1.06E+4
Co-57	-	6.92E+2	-	-	3.70E+5	3.14E+4	6.71E+2
Co-58	-	1.58E+3	-	-	9.28E+5	1.06E+5	2.07E+3
Co-60	-	1.15E+4	-	-	5.97E+6	2.85E+5	1.48E+4
Ni-63	4.32E+5	3.14E+4	-	-	1.78E+5	1.34E+4	1.45E+4
Ni-65	1.54E+0	2.10E+1	-	-	5.60E+3	1.23E+4	9.12E+2
Cu-64	-	1.46E+0	-	4.62E+0	6.78E+3	4.90E+4	6.15E+1
Zn-65	3.24E+4	1.03E+5	-	6.90E+4	8.64E+5	5.34E+4	4.66E+4
Zn-69	3.38E+2	6.51E+2	-	4.22E+3	9.30E+2	1.63E+1	4.52E+3
Br-82	-	-	-	-	-	1.04E+4	1.35E+4
Br-83	-	-	-	-	-	2.32E+2	2.41E+2
Br-84	-	-	-	-	-	1.64E+3	3.13E+2
Br-85	-	-	-	-	-	-	1.28E+1
Rb-86	-	1.35E+5	-	-	-	1.66E+4	5.90E+4
Rb-88	-	3.47E+2	-	-	-	3.34E+9	1.93E+2
Rb-89	-	2.56E+2	-	-	-	-	1.70E+2
Sr-89	3.04E+5	-	-	-	1.40E+6	3.50E+5	8.72E+3
Sr-90	9.92E+7	-	-	-	9.60E+6	7.22E+5	6.10E+6
Sr-91	6.19E+1	-	-	-	3.65E+4	1.91E+5	2.50E+0
Sr-92	6.74E+0	-	-	-	1.65E+4	4.30E+4	2.91E+1
Y-90	2.09E+3	-	-	-	1.70E+5	5.06E+5	5.61E+1
Y-91m	2.61E+1	-	-	-	1.92E+3	1.33E+0	1.02E+2
Y-91	4.62E+5	-	-	-	1.70E+6	3.85E+5	1.34E+4
Y-92	1.03E+1	-	-	-	1.57E+4	7.35E+4	3.02E+1
Y-93	9.44E+1	-	-	-	4.85E+4	4.22E+5	2.61E+0
Zr-95	1.07E+5	3.44E+4	-	5.42E+4	1.77E+6	1.50E+5	2.33E+4
Zr-97	9.68E+1	1.94E+1	-	2.97E+1	7.87E+4	5.23E+5	9.04E+0
Nb-95	1.41E+4	7.82E+3	-	7.74E+3	5.05E+5	1.04E+5	4.21E+3
Nb-97	2.22E+1	5.62E+2	-	6.54E+2	2.40E+3	2.42E+2	2.05E+2
Ko-99	-	1.21E+2	-	2.91E+2	9.12E+4	2.48E+5	2.30E+1
Tc-99m	1.03E+3	2.91E+3	-	4.42E+2	7.64E+2	4.16E+3	3.70E+2
Tc-101	4.18E+5	6.02E+5	-	1.08E+3	3.99E+2	-	5.90E+4
Ru-103	1.53E+3	-	-	5.83E+3	5.05E+5	1.10E+5	6.58E+2
Ru-105	7.90E+1	-	-	1.02E+0	1.10E+4	4.82E+4	3.11E+1
Ru-106	6.91E+4	-	-	1.34E+5	9.36E+6	9.12E+5	8.72E+3
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	1.08E+4	1.00E+4	-	1.97E+4	4.63E+6	3.02E+5	5.94E+3
Sb-124	3.12E+4	5.89E+2	7.55E+1	-	2.48E+6	4.06E+5	1.24E+4
Sb-125	5.34E+4	5.95E+2	5.40E+1	-	1.74E+6	1.01E+5	1.26E+4
Te-125m	3.42E+3	1.58E+3	1.05E+3	1.24E+4	3.14E+5	7.06E+4	4.67E+2
Te-127m	1.26E+4	5.77E+3	3.29E+3	4.58E+4	9.60E+5	1.50E+5	1.57E+3
Te-127	1.40E+0	6.42E+1	1.06E+0	5.10E+0	6.51E+3	5.74E+4	3.10E+1
Te-129m	9.76E+3	4.67E+3	3.44E+3	3.66E+4	1.16E+6	3.83E+5	1.58E+3
Te-129	4.98E+2	2.39E+2	3.90E+2	1.87E+1	1.94E+3	1.57E+2	1.24E+2
Te-131m	6.99E+1	4.36E+1	5.50E+1	3.09E+2	1.46E+5	5.56E+5	2.90E+1
Te-131	1.11E+2	5.95E+2	9.38E+3	4.37E+2	1.39E+3	1.84E+1	3.59E+3
Tc-132	2.60E+2	2.15E+2	1.90E+2	1.46E+3	2.88E+5	5.10E+5	1.62E+2
I-130	4.58E+3	1.34E+4	1.14E+6	2.09E+4	-	7.69E+3	5.28E+3
I-131	2.52E+4	3.58E+4	1.19E+7	6.13E+4	-	6.28E+3	2.05E+4
I-132	1.16E+3	3.26E+3	1.14E+5	5.18E+3	-	4.04E+3	1.16E+3
I-133	8.64E+3	1.48E+4	2.15E+6	2.58E+4	-	8.88E+3	4.52E+3
I-134	6.44E+2	1.73E+3	2.98E+4	2.75E+3	-	1.01E+0	6.15E+2
I-135	2.60E+3	6.98E+3	4.48E+5	1.11E+4	-	5.25E+3	2.57E+3
Cs-134	3.73E+5	8.48E+5	-	2.87E+5	9.76E+4	1.04E+4	7.28E+5
Cs-136	3.90E+4	1.46E+5	-	8.56E+4	1.20E+4	1.17E+4	1.10E+5
Cs-137	4.78E+5	6.21E+5	-	2.22E+5	7.52E+4	8.60E+3	4.28E+5
Cs-138	3.31E+2	6.21E+2	-	4.80E+2	4.86E+1	1.86E+3	3.24E+2
Ba-139	9.36E+1	6.64E+4	-	6.22E+6	2.76E+3	8.96E+2	2.74E+2
Ba-140	3.90E+4	4.90E+1	-	1.67E+1	1.27E+6	2.18E+5	2.57E+3
Ba-141	1.00E+1	7.51E+5	-	7.00E+5	1.94E+3	1.16E+7	3.36E+3
Ba-142	2.63E+2	2.70E+5	-	2.29E+5	1.19E+3	-	1.64E+3
La-140	3.44E+2	1.74E+2	-	-	1.36E+5	4.58E+5	4.58E+1
La-142	6.83E+1	3.10E+1	-	-	6.33E+3	2.11E+3	7.72E+2
Ce-141	1.99E+4	1.35E+4	-	6.26E+3	3.62E+5	1.20E+5	1.53E+3
Ce-143	1.86E+2	1.38E+2	-	6.08E+1	7.98E+4	2.28E+5	1.53E+1
Ce-144	3.43E+6	1.43E+6	-	8.48E+5	7.78E+6	8.16E+5	1.84E+5
Pr-143	9.36E+3	3.75E+3	-	2.16E+3	2.81E+5	2.00E+5	4.64E+2
Pr-144	3.01E+2	1.35E+2	-	7.05E+3	1.02E+3	2.15E+0	1.53E+3
Nd-147	5.27E+3	6.10E+3	-	3.56E+3	2.11E+5	1.73E+5	3.65E+2
W-187	8.48E+0	7.08E+0	-	-	2.90E+4	1.55E+5	2.48E+0
Wp-239	2.10E+2	2.26E+1	-	7.00E+1	3.76E+4	1.19E+5	1.24E+1

Table 3-7 (Continued)  
 R. Inhalation Pathway Dose Factors - TEENAGER  
 (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ )

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-1	-	1.27E+3	1.27E+3	1.27E+3	1.27E+3	1.27E+3	1.27E+3
C-14	2.60E+4	4.87E+3	4.87E+3	4.87E+3	4.87E+3	4.87E+3	4.87E+3
Na-24	1.38E+4	1.38E+4	1.38E+4	1.38E+4	1.38E+4	1.38E+4	1.38E+4
P-32	1.89E+6	1.10E+5	-	-	-	9.28E+4	7.16E+4
U-231	-	-	7.50E+1	3.07E+1	2.10E+4	3.00E+3	1.35E+2
Mn-54	-	5.11E+4	-	1.27E+4	1.98E+6	6.68E+4	8.40E+3
Mn-56	-	1.70E+0	-	1.79E+0	1.52E+4	5.74E+4	2.52E+1
Fe-55	3.34E+4	2.38E+4	-	-	1.24E+5	6.39E+3	5.54E+3
Fe-59	1.59E+4	1.70E+4	-	-	1.53E+6	1.78E+5	1.43E+4
Co-57	-	6.92E+2	-	-	5.86E+5	3.14E+4	9.20E+2
Co-58	-	2.07E+3	-	-	1.34E+6	9.52E+4	2.78E+3
Co-60	-	1.51E+4	-	-	8.72E+6	2.59E+5	1.98E+4
Ni-63	5.80E+5	4.34E+4	-	-	3.07E+5	1.42E+4	1.98E+4
Ni-65	2.18E+0	2.93E-1	-	-	9.36E+3	3.67E+4	1.27E-1
Cu-64	-	2.03E+0	-	6.41E+0	1.11E+4	6.14E+4	8.48E-1
Zn-65	3.86E+4	1.34E+5	-	54E+4	1.24E+6	4.66E+4	6.24E+4
Zn-69	4.83E+2	9.20E-2	-	8.7E+2	1.58E+3	2.85E+2	6.46E-3
Br-82	-	-	-	-	-	-	1.82E+4
Br-83	-	-	-	-	-	-	3.44E+2
Br-84	-	-	-	-	-	-	4.33E+2
Br-85	-	-	-	-	-	-	1.83E+1
Rb-86	-	1.90E+5	-	-	-	1.77E+4	8.40E+4
Rb-88	-	5.46E+2	-	-	-	2.92E+5	2.72E+2
Rb-89	-	3.52E+2	-	-	-	3.38E+7	2.33E+2
Sr-89	4.34E+5	-	-	-	2.42E+3	3.71E+5	1.25E+4
Sr-90	1.08E+6	-	-	-	1.65E+7	7.65E+5	6.68E+6
Sr-91	8.80E+1	-	-	-	6.07E+4	2.59E+5	3.51E+0
Sr-92	9.52E+0	-	-	-	2.74E+4	1.19E+5	4.06E-1
Y-90	2.98E+3	-	-	-	2.93E+5	5.59E+5	8.00E+1
Y-91m	3.70E-1	-	-	-	3.20E+3	3.02E+1	1.42E-2
Y-91	6.61E+5	-	-	-	2.94E+6	4.09E+5	1.77E+4
Y-92	1.47E+1	-	-	-	2.88E+4	1.65E+5	4.29E-1
Y-93	1.35E+2	-	-	-	8.32E+4	5.79E+5	3.72E+0
Zr-95	1.46E+5	4.58E+4	-	6.74E+4	2.69E+6	1.49E+5	3.15E+4
Zr-97	1.38E+2	2.72E+1	-	4.12E+1	1.30E+5	6.30E+5	1.26E+1
Nb-95	1.86E+4	1.03E+4	-	1.00E+4	7.51E+5	9.68E+4	5.66E+3
Nb-97	3.14E-1	7.78E-2	-	9.12E-2	3.93E+3	2.17E+3	2.84E-2
Mo-99	-	1.69E+2	-	4.11E+2	1.54E+5	2.89E+5	3.22E+1
Tc-99m	1.38E-3	3.86E-3	-	5.76E-2	1.15E+3	6.13E+3	4.99E-2
Tc-101	5.92E-5	8.40E-5	-	1.52E-3	6.67E+2	8.72E+7	8.24E-4
Ru-103	2.10E+3	-	-	7.43E+3	7.83E+5	1.09E+5	8.96E+2
Ru-105	1.12E+0	-	-	1.41E+0	1.82E+4	9.04E+4	4.34E-1
Ru-106	9.84E+4	-	-	1.90E+5	1.61E+7	9.60E+5	1.24E+4
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	1.38E+4	1.31E+4	-	2.50E+4	6.75E+6	2.73E+5	7.99E+3
Sb-124	4.10E+4	7.94E+2	9.76E+1	-	3.85E+6	3.98E+5	1.68E+4
Sb-125	7.38E+4	8.08E+2	7.04E+1	-	2.74E+6	9.92E+4	1.72E+4
Te-125m	4.88E+3	2.24E+3	1.40E+3	-	5.38E+5	7.50E+4	6.67E+2
Te-127m	1.80E+4	8.16E+3	4.38E+3	6.54E+4	1.86E+6	1.59E+5	2.18E+3
Te-127	2.01E+0	9.12E-1	1.42E+0	7.28E+0	1.12E+4	8.08E+4	4.42E-1
Te-129m	1.39E+4	6.58E+3	4.58E+3	5.19E+4	1.98E+6	4.05E+5	2.25E+3
Te-129	7.10E-2	3.38E-2	5.18E-2	2.66E-1	3.30E+3	1.62E+3	1.76E-2
Te-131m	9.84E+1	6.01E+1	7.25E+1	4.39E+2	2.38E+5	6.21E+5	4.02E-1
Te-131	1.58E-2	8.32E-3	1.24E-2	6.18E-2	2.34E+3	1.51E+1	5.04E-3
Te-132	3.60E+2	2.80E+2	2.46E+2	1.95E+3	4.49E+5	4.63E+5	2.19E+2
I-130	6.24E+3	1.79E+4	1.49E+6	2.75E+4	-	9.12E+3	7.17E+3
I-131	3.54E+4	4.91E+4	1.46E+7	8.40E+4	-	6.49E+3	2.64E+4
I-132	1.59E+3	4.38E+3	1.51E+5	6.92E+3	-	1.27E+3	1.58E+3
I-133	1.22E+4	2.05E+4	2.92E+6	3.59E+4	-	1.03E+4	6.22E+3
I-134	8.88E+2	2.32E+3	3.95E+4	3.66E+3	-	2.04E+1	8.40E+2
I-135	3.70E+3	9.44E+3	6.21E+5	1.49E+4	-	6.95E+3	3.49E+3
Ca-134	5.02E+5	1.13E+6	-	3.75E+5	1.46E+5	9.76E+3	5.49E+5
Ca-136	5.15E+4	1.94E+5	-	1.10E+5	1.78E+4	1.09E+4	1.37E+5
Ca-137	6.70E+5	8.48E+5	-	3.04E+5	1.21E+5	8.48E+3	3.11E+5
Ca-138	4.66E+2	8.56E+2	-	6.62E+2	7.87E+1	2.70E-1	4.44E+2
Ba-139	1.34E+0	9.44E-4	-	8.88E-4	6.46E+3	6.45E+3	3.90E-2
Ba-140	5.47E+4	6.70E+1	-	2.28E+1	2.03E+6	2.29E+5	3.52E+3
Ba-141	1.42E-1	1.06E-4	-	9.84E-5	3.29E+3	7.46E-4	4.74E-3
Ba-142	3.70E-2	3.70E-5	-	3.14E-5	1.91E+3	-	2.27E-3
La-140	4.79E+2	2.36E+2	-	-	2.14E+5	4.87E+5	6.26E+1
La-142	9.60E+1	4.25E-1	-	-	1.02E+6	1.20E+4	1.06E-1
Ca-141	2.94E+4	1.90E+4	-	8.88E+3	6.14E+5	1.28E+5	2.17E+3
Ca-143	2.66E+2	1.94E+2	-	8.64E+1	1.30E+5	2.55E+5	2.16E+1
Ca-144	4.89E+6	2.02E+6	-	1.21E+6	1.34E+7	8.64E+5	2.62E+5
Pr-143	1.34E+4	5.31E+3	-	3.09E+3	4.83E+5	2.14E+5	6.62E+2
Pr-144	4.30E+2	1.76E+2	-	1.01E+2	1.75E+3	2.35E+4	2.18E-3
Nd-147	7.86E+3	8.56E+3	-	5.02E+3	3.72E+5	1.82E+5	5.13E+2
N-187	1.20E+1	9.76E+0	-	-	4.74E+4	1.77E+5	3.63E+0
Nd-229	3.38E+2	3.19E+1	-	1.00E+2	6.49E+4	1.32E+5	1.77E+1

Table 3-7 (continued)  
 $R_{10}$  Inhalation Pathway Dose Factors - CHILD  
 (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ )

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-3	-	1.12E+3	1.12E+3	1.12E+3	1.12E+3	1.12E+3	1.12E+3
C-14	3.59E+4	6.73E+3	6.73E+3	6.73E+3	6.73E+3	6.73E+3	6.73E+3
Na-24	1.61E+4	1.61E+4	1.61E+4	1.61E+4	1.61E+4	1.61E+4	1.61E+4
P-32	2.60E+6	1.14E+5	-	-	-	4.22E+4	9.88E+4
Tr-51	-	-	8.55E+1	2.43E+1	1.70E+4	1.08E+3	1.54E+2
Mn-54	-	4.29E+4	-	1.00E+4	1.58E+6	2.29E+4	9.51E+3
Mn-56	-	1.66E+0	-	1.67E+0	1.31E+4	1.23E+5	3.12E+1
Fe-57	4.74E+4	2.52E+4	-	-	1.11E+5	2.87E+3	7.77E+3
Fe-59	2.07E+4	3.34E+4	-	-	1.27E+6	7.07E+4	1.67E+4
Co-57	-	9.03E+2	-	-	5.07E+5	1.32E+4	1.07E+3
Co-58	-	1.77E+3	-	-	1.11E+6	3.44E+4	3.16E+3
Co-60	-	1.31E+4	-	-	7.07E+6	9.62E+4	2.26E+4
Ni-63	8.21E+5	4.63E+4	-	-	2.75E+5	6.33E+3	2.80E+4
Ni-65	2.99E+0	2.96E+1	-	-	8.18E+3	8.40E+4	1.64E+1
Cu-64	-	1.99E+0	-	6.03E+0	9.58E+3	3.67E+4	1.07E+0
Zn-65	4.26E+4	1.13E+5	-	7.14E+4	9.95E+5	1.63E+4	7.03E+4
Zn-69	6.70E+2	9.66E+2	-	5.85E+2	1.42E+3	1.01E+4	8.92E+3
Br-82	-	-	-	-	-	-	2.09E+4
Br-83	-	-	-	-	-	-	4.74E+2
Br-84	-	-	-	-	-	-	5.48E+2
Br-85	-	-	-	-	-	-	2.53E+1
Rb-86	-	1.98E+5	-	-	-	7.99E+3	1.14E+5
Rb-88	-	5.62E+2	-	-	-	1.72E+1	3.66E+2
Rb-89	-	3.45E+2	-	-	-	1.89E+0	2.90E+2
Sr-89	5.59E+5	-	-	-	2.16E+6	1.67E+5	1.72E+4
Sr-90	1.01E+6	-	-	-	1.48E+7	3.43E+5	6.44E+6
Sr-91	1.21E+2	-	-	-	5.33E+4	1.74E+5	4.59E+0
Sr-92	1.31E+1	-	-	-	2.40E+4	2.42E+5	5.25E+1
Y-90	4.11E+3	-	-	-	2.62E+5	2.68E+5	1.11E+2
Y-91m	5.07E+1	-	-	-	2.81E+3	1.72E+3	1.84E+2
Y-91	9.14E+5	-	-	-	2.63E+6	1.84E+5	2.44E+4
Y-92	2.04E+1	-	-	-	2.39E+4	2.39E+5	5.81E+1
Y-93	1.86E+2	-	-	-	7.44E+4	3.89E+5	5.11E+0
Zr-95	1.90E+5	4.18E+4	-	5.96E+4	2.23E+6	6.11E+4	3.70E+4
Zr-97	1.88E+2	2.72E+1	-	3.89E+1	1.13E+5	3.51E+5	1.60E+1
Nb-95	2.35E+4	9.18E+3	-	8.62E+3	6.14E+5	3.70E+4	6.55E+3
Nb-97	4.29E+1	7.70E+2	-	8.55E+2	3.42E+3	2.78E+4	3.60E+2
Mo-99	-	1.72E+2	-	3.92E+2	1.35E+5	1.27E+5	4.26E+1
Tc-99m	1.78E+3	3.48E+3	-	5.07E+2	9.51E+2	4.81E+3	5.77E+2
Tc-101	8.10E+5	8.51E+5	-	1.45E+3	5.85E+2	1.63E+1	1.08E+3
Ru-103	2.79E+3	-	-	7.03E+3	6.62E+5	4.48E+4	1.07E+3
Ru-105	1.53E+0	-	-	1.34E+0	1.59E+4	9.95E+4	5.55E+1
Rh-106	1.36E+5	-	-	1.84E+5	1.43E+7	4.29E+5	1.69E+4
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	1.69E+4	1.14E+4	-	2.12E+4	5.48E+6	1.00E+5	9.14E+3
Sb-124	5.74E+4	7.40E+2	1.26E+2	-	3.24E+6	1.64E+5	2.00E+4
Sb-125	9.84E+4	7.59E+2	9.10E+1	-	2.32E+6	4.03E+4	2.07E+4
Te-125m	6.73E+3	2.33E+3	1.92E+3	-	4.77E+5	3.38E+4	9.14E+2
Te-127m	2.49E+4	8.55E+3	6.07E+3	6.36E+4	1.48E+6	7.14E+4	3.02E+3
Te-127	2.77E+0	9.51E+1	1.96E+2	7.07E+0	1.00E+4	5.62E+4	6.11E+1
Te-129m	1.92E+4	6.85E+3	6.33E+3	5.03E+4	1.76E+6	1.82E+5	3.04E+3
Te-129	9.77E+2	3.50E+2	7.14E+2	2.57E+1	2.93E+3	2.55E+4	2.38E+2
Te-131m	1.34E+2	5.92E+1	9.77E+1	4.00E+2	2.06E+5	3.08E+5	5.07E+1
Te-131	2.17E+2	8.44E+3	1.70E+2	5.84E+2	2.05E+3	1.33E+3	6.59E+3
Te-132	4.81E+2	2.72E+2	3.17E+2	1.77E+3	3.77E+5	1.38E+5	2.63E+2
I-130	8.18E+3	1.64E+4	1.85E+6	2.45E+4	-	5.11E+3	8.44E+3
I-131	4.81E+4	4.81E+4	1.62E+7	7.88E+4	-	2.84E+3	2.73E+4
I-132	2.12E+3	4.07E+3	1.94E+5	6.25E+3	-	3.20E+3	1.88E+3
I-133	1.66E+4	2.03E+4	3.85E+6	3.38E+4	-	5.48E+3	7.70E+3
I-134	1.17E+3	2.16E+3	5.07E+4	3.30E+3	-	9.55E+2	9.95E+2
I-135	4.92E+3	8.73E+3	7.92E+5	1.34E+4	-	4.44E+3	4.14E+3
Ca-134	6.51E+5	1.01E+6	-	3.30E+5	1.21E+5	3.85E+3	2.25E+5
Ca-136	6.51E+4	1.71E+5	-	9.55E+4	1.45E+4	4.18E+3	1.16E+5
Ca-137	9.07E+5	8.25E+5	-	2.82E+5	1.04E+5	3.62E+3	1.28E+5
Ca-138	6.33E+2	8.40E+2	-	6.22E+2	6.81E+1	2.70E+2	5.55E+2
Ba-139	1.84E+0	9.84E+4	-	8.62E+4	5.77E+3	5.77E+4	5.37E+2
Ba-140	7.40E+4	6.48E+1	-	2.11E+1	1.74E+6	1.02E+5	4.33E+3
Ba-141	1.96E+1	1.09E+4	-	9.47E+5	2.92E+3	2.75E+2	6.36E+3
Ba-142	5.00E+2	3.60E+5	-	2.91E+5	1.64E+3	2.74E+0	2.79E+3
La-140	6.44E+2	2.25E+2	-	-	1.83E+5	2.26E+5	7.55E+1
La-142	1.30E+0	4.11E+1	-	-	6.70E+3	7.59E+4	1.29E+1
Ce-141	3.92E+4	1.95E+4	-	8.55E+3	5.44E+5	5.66E+4	2.90E+3
Ce-143	3.66E+2	1.99E+2	-	8.36E+1	1.15E+5	1.27E+5	2.87E+1
Ce-144	6.77E+6	2.12E+6	-	1.17E+6	1.10E+7	3.89E+5	3.61E+5
Pr-143	1.85E+4	5.55E+3	-	3.00E+3	4.33E+5	9.73E+4	9.14E+2
Pr-144	5.96E+2	1.83E+2	-	9.77E+3	1.57E+3	1.97E+2	3.00E+3
Nd-147	1.08E+4	8.73E+3	-	4.81E+3	3.28E+5	8.21E+4	6.81E+2
W-187	1.63E+1	9.66E+0	-	-	4.11E+4	9.10E+4	4.33E+0
Ho-209	4.66E+2	3.34E+1	-	9.73E+1	5.81E+4	6.40E+4	2.35E+1



Table 3-7 (continued)  
 $R_{10}$  Inhalation pathway Dose Factors - INFANT  
 (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ )

Radionuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T-Body
H-3	-	6.47E+2	6.47E+2	6.47E+2	6.47E+2	6.47E+2	6.47E+2
C-14	2.65E+4	5.31E+3	5.31E+3	5.31E+3	5.31E+3	5.31E+3	5.31E+3
Na-24	1.06E+4	1.06E+4	1.06E+4	1.06E+4	1.06E+4	1.06E+4	1.06E+4
P-32	2.03E+6	1.12E+5	-	-	-	1.61E+4	7.74E+4
Cr-51	-	-	5.75E+1	1.32E+1	1.28E+4	3.57E+2	8.95E+1
Mn-54	-	2.53E+4	-	4.98E+3	1.00E+6	7.06E+3	4.98E+3
Mn-56	-	1.54E+0	-	1.10E+0	1.25E+4	7.17E+4	2.21E+1
Fe-55	1.97E+4	1.17E+4	-	-	8.69E+4	1.09E+3	3.33E+3
Fe-59	1.36E+4	2.35E+4	-	-	1.02E+6	2.48E+4	9.48E+5
Co-57	-	6.51E+2	-	-	3.79E+5	4.86E+3	6.41E+2
Co-58	-	1.22E+3	-	-	7.77E+5	1.11E+4	1.82E+3
Co-60	-	8.02E+3	-	-	4.51E+6	3.19E+4	1.18E+4
Ni-63	3.39E+5	2.04E+4	-	-	2.09E+5	2.42E+3	1.16E+4
Ni-65	2.39E+0	2.84E+1	-	-	8.12E+3	5.01E+4	1.23E+1
Cu-64	-	1.88E+0	-	3.98E+0	9.30E+3	1.50E+4	7.74E+1
Zn-65	1.93E+4	6.26E+4	-	3.25E+4	6.47E+5	5.14E+4	3.11E+4
Zn-69	5.39E+2	9.67E+2	-	4.02E+2	1.47E+3	1.32E+4	7.18E+3
Br-82	-	-	-	-	-	-	1.33E+4
Br-83	-	-	-	-	-	-	3.81E+2
Br-84	-	-	-	-	-	-	4.00E+2
Br-85	-	-	-	-	-	-	2.04E+1
Rb-86	-	1.90E+5	-	-	-	3.04E+3	8.82E+4
Rb-88	-	5.57E+2	-	-	-	3.39E+2	2.87E+2
Rb-89	-	3.21E+2	-	-	-	6.82E+1	2.06E+2
Sr-89	3.98E+5	-	-	-	2.03E+6	6.40E+4	1.14E+4
Sr-90	4.09E+7	-	-	-	1.12E+7	1.31E+5	2.59E+6
Sr-91	9.56E+1	-	-	-	5.26E+4	7.34E+4	3.46E+0
Sr-92	1.05E+1	-	-	-	2.38E+4	1.40E+5	3.91E+1
Y-90	3.29E+3	-	-	-	2.69E+5	1.04E+5	8.82E+1
Y-91m	4.07E+1	-	-	-	2.79E+3	2.35E+3	1.39E+2
Y-91	5.88E+5	-	-	-	2.45E+6	7.03E+4	1.57E+4
Y-92	1.64E+1	-	-	-	2.45E+4	1.27E+5	4.61E+1
Y-93	1.50E+2	-	-	-	7.64E+4	1.67E+5	4.07E+0
Zr-95	1.15E+5	2.79E+4	-	3.11E+4	1.75E+6	2.17E+4	2.03E+4
Zr-97	1.50E+2	2.56E+1	-	2.59E+1	1.10E+5	1.40E+5	1.17E+1
Nb-95	1.57E+4	6.43E+3	-	4.72E+3	4.79E+5	1.27E+4	3.78E+3
Nb-97	3.42E+1	7.29E+2	-	5.70E+2	3.32E+3	2.69E+4	2.63E+2
Mo-99	-	1.65E+2	-	2.65E+2	1.35E+5	4.87E+4	3.23E+1
Tc-99m	1.40E+3	2.88E+3	-	3.11E+2	8.11E+2	2.03E+3	3.72E+2
Tc-101	6.51E+5	8.23E+5	-	9.79E+4	5.84E+2	8.44E+2	8.12E+4
Ru-103	2.02E+3	-	-	4.24E+3	5.52E+5	1.61E+4	6.79E+2
Ru-105	1.22E+0	-	-	8.99E+1	1.57E+4	4.84E+4	4.10E+1
Ru-106	8.68E+4	-	-	1.07E+5	1.16E+7	1.64E+5	1.09E+4
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	9.98E+3	7.22E+3	-	1.09E+4	3.67E+6	3.30E+4	5.00E+3
Sb-124	3.79E+4	5.56E+2	1.01E+2	-	2.65E+6	5.91E+4	1.20E+4
Sb-125	5.17E+4	4.77E+2	6.23E+1	-	1.64E+6	1.47E+4	1.09E+4
Te-125m	4.78E+3	1.89E+3	1.62E+3	-	4.47E+5	1.29E+4	6.58E+2
Te-127m	1.67E+4	6.90E+3	4.87E+3	3.75E+4	1.31E+6	2.73E+4	2.07E+3
Te-127	2.23E+0	9.53E+1	1.85E+0	4.86E+0	1.03E+4	2.44E+4	4.89E+1
Te-129m	1.41E+4	6.09E+3	5.47E+3	3.18E+4	1.68E+6	6.90E+4	2.23E+3
Te-129	7.88E+2	3.47E+2	6.75E+2	1.75E+1	3.00E+3	2.63E+4	1.88E+2
Te-131m	1.07E+2	5.50E+1	8.93E+1	2.65E+2	1.99E+5	1.19E+5	3.63E+1
Te-131	1.74E+2	8.22E+3	1.58E+2	3.99E+3	2.04E+3	8.22E+3	5.00E+3
Te-132	3.72E+2	2.37E+2	2.79E+2	1.03E+3	3.40E+5	4.41E+4	1.76E+2
I-130	6.36E+3	1.39E+4	1.60E+6	1.53E+4	-	1.99E+3	5.57E+3
I-131	3.79E+4	4.44E+4	1.48E+7	5.18E+4	-	1.04E+3	1.96E+4
I-132	1.69E+3	3.54E+3	1.69E+5	3.95E+3	-	1.90E+3	1.24E+3
I-133	1.32E+4	1.92E+4	3.56E+6	2.24E+6	-	2.16E+3	5.60E+3
I-134	9.21E+2	1.88E+3	4.45E+4	2.09E+3	-	1.29E+3	6.65E+2
I-135	3.86E+3	7.60E+3	6.96E+5	8.47E+3	-	1.83E+3	2.77E+3
Cs-134	3.96E+5	7.03E+5	-	1.90E+5	7.97E+4	1.33E+3	7.45E+4
Cs-136	4.83E+4	1.35E+5	-	5.64E+4	1.18E+4	1.43E+3	5.29E+4
Cs-137	5.49E+5	6.12E+5	-	1.72E+5	7.13E+4	1.33E+3	4.55E+4
Cs-138	5.05E+2	7.81E+2	-	4.10E+2	6.54E+1	8.76E+2	3.98E+2
Ba-139	1.48E+0	9.84E+4	-	5.92E+4	5.95E+3	5.10E+4	4.30E+2
Ba-140	5.60E+4	5.60E+1	-	1.34E+1	1.60E+6	3.84E+4	2.90E+3
Ba-141	1.57E+1	1.08E+4	-	6.50E+5	2.97E+3	4.75E+3	4.97E+3
Ba-142	3.98E+2	3.30E+5	-	1.90E+5	1.55E+3	6.93E+2	1.96E+3
La-140	5.05E+2	2.00E+2	-	-	1.68E+5	8.48E+4	5.15E+1
La-142	1.03E+0	3.77E+1	-	-	8.22E+3	5.95E+4	9.04E+2
Ce-141	2.77E+4	1.67E+4	-	5.25E+3	5.17E+5	2.16E+4	1.99E+3
Ce-142	2.93E+2	1.93E+2	-	5.64E+1	1.16E+5	4.97E+4	2.21E+1
Ce-144	3.19E+6	1.21E+6	-	5.38E+5	9.84E+6	1.48E+5	1.76E+5
Pr-143	1.40E+4	5.24E+3	-	1.97E+3	4.33E+5	3.72E+4	6.99E+2
Pr-144	4.79E+2	1.85E+2	-	6.72E+3	1.61E+3	4.28E+3	2.41E+3
Nd-147	7.94E+3	8.13E+3	-	3.15E+3	3.22E+5	3.12E+4	5.00E+2
W-187	1.30E+1	9.02E+0	-	-	3.96E+4	3.58E+4	3.12E+0
Re-229	3.71E+2	3.32E+1	-	6.62E+1	5.95E+4	2.49E+4	1.88E+1

Table 3-7 (continued)  
 R<sub>10</sub> Grass - Cow - Milk Pathway Dose Factors - ADULT  
 (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) for H-3 and C-14  
 ( $\text{m}^3 \cdot \text{mrem}/\text{yr}$  per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T-Body
H-3	-	7.63E+2	7.63E+2	7.63E+2	7.63E+2	7.63E+2	7.63E+2
C-14	3.63E+5	7.26E+4	7.26E+4	7.26E+4	7.26E+4	7.26E+4	7.26E+4
Na-24	2.54E+6	2.54E+6	2.54E+6	2.54E+6	2.54E+6	2.54E+6	2.54E+6
P-32	1.71E+10	1.06E+9	-	-	-	1.92E+9	6.60E+8
P-31	-	-	1.71E+4	6.30E+3	3.80E+4	7.20E+6	2.86E+4
Mn-54	-	8.40E+6	-	2.50E+6	-	2.57E+7	1.80E+6
Mn-56	-	4.23E+3	-	5.38E+3	-	1.35E+1	7.51E+4
Fe-55	2.51E+7	1.73E+7	-	-	9.67E+6	9.95E+6	4.04E+6
Fe-59	2.98E+7	7.00E+7	-	-	1.95E+7	2.33E+8	2.68E+7
Co-57	-	1.28E+6	-	-	-	1.25E+7	2.13E+6
Co-58	-	4.72E+6	-	-	-	1.57E+7	1.06E+7
Co-60	-	1.64E+7	-	-	-	3.08E+8	3.62E+7
Ni-63	6.73E+9	4.66E+8	-	-	-	9.73E+7	2.26E+8
Ni-65	3.70E+1	4.81E+2	-	-	-	1.22E+0	2.19E+2
Cu-64	-	2.41E+4	-	6.08E+4	-	2.05E+6	1.13E+4
Zn-65	1.37E+9	4.36E+9	-	2.92E+9	-	2.75E+9	1.97E+9
Zn-69	-	-	-	-	-	-	-
Br-82	-	-	-	-	-	3.72E+7	1.25E+7
Br-83	-	-	-	-	-	1.49E+1	1.03E+1
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
Rb-86	-	2.59E+9	-	-	-	5.11E+6	1.21E+9
Rb-88	-	-	-	-	-	-	-
Rb-87	-	-	-	-	-	-	-
Sr-89	1.45E+9	-	-	-	-	2.33E+8	4.16E+7
Sr-90	4.68E+10	-	-	-	-	1.35E+9	1.15E+10
Sr-91	3.13E+4	-	-	-	-	1.49E+5	1.27E+3
Sr-92	4.89E+1	-	-	-	-	9.68E+0	2.11E+2
Y-90	7.07E+1	-	-	-	-	7.50E+5	1.90E+0
Y-91m	-	-	-	-	-	-	-
Y-91	8.60E+3	-	-	-	-	4.73E+6	2.30E+2
Y-92	5.42E+5	-	-	-	-	9.49E+1	1.58E+6
Y-93	2.33E+1	-	-	-	-	7.39E+3	6.43E+3
Zr-95	9.46E+2	3.03E+2	-	4.76E+2	-	9.62E+5	2.05E+2
Zr-97	4.26E+1	8.59E+2	-	1.30E+1	-	2.66E+4	3.93E+2
Nb-95	8.25E+4	4.59E+4	-	4.54E+4	-	2.79E+8	2.47E+4
Nb-97	-	-	-	-	-	5.47E+9	-
Mo-99	-	2.52E+7	-	5.72E+7	-	5.85E+7	4.80E+6
Tc-99m	3.25E+0	9.19E+0	-	1.40E+2	4.50E+0	8.44E+3	1.17E+2
Tc-101	-	-	-	-	-	-	-
Ru-103	1.02E+3	-	-	3.89E+3	-	1.19E+5	4.39E+2
Ru-105	8.57E+4	-	-	1.11E+2	-	5.24E+1	3.38E+4
Ru-106	2.04E+4	-	-	3.94E+4	-	1.32E+6	2.58E+3
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	5.83E+7	5.39E+7	-	1.06E+8	-	2.20E+10	3.20E+7
Sb-124	2.57E+7	4.86E+5	6.24E+4	-	2.00E+7	7.31E+6	1.02E+7
Sb-125	2.04E+7	2.28E+5	2.08E+4	-	1.58E+7	2.25E+8	4.86E+6
Te-125m	1.63E+7	5.90E+6	4.90E+6	6.63E+7	-	6.50E+7	2.18E+6
Te-127m	4.58E+7	1.64E+7	1.17E+7	1.86E+8	-	1.54E+8	5.58E+6
Te-127	6.72E+2	2.41E+2	4.98E+2	2.74E+3	-	5.30E+4	1.45E+2
Te-129m	6.04E+7	2.25E+7	2.08E+7	2.52E+8	-	3.04E+8	9.57E+6
Te-129	-	-	-	-	-	-	-
Te-131m	3.61E+5	1.77E+5	2.80E+5	1.79E+6	-	1.75E+7	1.47E+5
Te-131	-	-	-	-	-	-	-
Te-132	2.39E+6	1.55E+6	1.71E+6	1.49E+7	-	7.32E+7	1.45E+6
I-130	6.26E+5	1.26E+6	1.07E+6	1.96E+6	-	1.06E+6	4.96E+5
I-131	2.96E+8	4.24E+8	1.39E+11	7.27E+8	-	1.12E+8	2.43E+8
I-132	1.64E+1	4.37E+1	1.53E+1	6.97E+1	-	8.22E+2	1.53E+1
I-133	3.97E+6	6.90E+6	1.01E+9	1.20E+7	-	6.20E+6	2.10E+6
I-134	-	-	-	-	-	-	-
I-135	1.39E+4	3.63E+4	2.40E+6	5.83E+4	-	4.10E+4	1.34E+4
Cs-134	5.65E+9	1.34E+10	-	4.35E+9	1.44E+9	2.35E+8	1.10E+10
Cs-136	2.61E+8	1.03E+9	-	5.74E+8	7.87E+7	1.17E+8	7.42E+8
Cs-137	7.38E+9	1.01E+10	-	3.43E+9	1.14E+9	1.95E+8	6.61E+9
Cs-138	-	-	-	-	-	-	-
Ba-139	4.70E+8	-	-	-	-	8.34E+8	1.38E+9
Ba-140	2.69E+7	3.38E+4	-	1.15E+4	1.93E+4	5.54E+7	1.76E+6
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	4.49E+0	3.26E+0	-	-	-	1.64E+5	5.97E+1
La-142	-	-	-	-	-	3.03E+8	-
Ce-141	4.84E+3	3.27E+3	-	1.52E+3	-	1.25E+7	3.71E+2
Ce-143	4.19E+1	3.09E+4	-	1.36E+1	-	1.16E+6	3.42E+0
Ce-144	3.58E+5	1.50E+5	-	8.87E+4	-	1.21E+8	1.92E+5
Pr-143	1.59E+2	6.37E+1	-	3.68E+1	-	6.96E+5	7.88E+0
Pr-144	-	-	-	-	-	-	-
Nd-147	9.42E+1	1.09E+2	-	6.37E+1	-	5.23E+5	6.52E+0
W-187	6.56E+3	5.48E+3	-	-	-	1.80E+6	1.92E+3
Np-239	3.64E+0	3.60E+1	-	1.12E+0	-	7.39E+4	1.98E+1



Table 3-7 (continued)  
R(10), Grass - Cow - Milk Pathway Dose Factors - TEENAGER  
(mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) for H-3 and C-14  
( $\text{m}^2 \cdot \text{mrem}/\text{yr}$  per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-3	-	9.94E+2	9.94E+2	9.94E+2	9.94E+2	9.94E+2	9.94E+2
C-14	6.70E+5	1.34E+5	1.34E+5	1.34E+5	1.34E+5	1.34E+5	1.34E+5
Na-24	4.44E+6	4.44E+6	4.44E+6	4.44E+6	4.44E+6	4.44E+6	4.44E+6
P-32	1.15E+10	1.95E+9	-	-	-	2.65E+9	1.22E+9
Tr-51	-	-	2.78E+4	1.10E+4	7.13E+4	8.40E+6	5.00E+4
Mn-54	-	1.40E+7	-	4.17E+6	-	2.87E+7	2.78E+6
Mn-56	-	7.51E+3	-	9.50E+3	-	4.94E+1	1.33E+3
Fe-55	4.45E+7	3.16E+7	-	-	2.00E+7	1.37E+7	7.36E+6
Fe-59	5.20E+7	1.21E+8	-	-	3.82E+7	2.87E+8	4.68E+7
Co-57	-	2.25E+6	-	-	-	4.19E+7	3.76E+6
Co-58	-	7.95E+6	-	-	-	1.10E+8	1.83E+7
Co-60	-	2.78E+7	-	-	-	3.62E+8	6.26E+7
Ni-63	1.18E+10	8.35E+8	-	-	-	1.33E+8	4.01E+8
Ni-65	6.78E+1	8.66E+2	-	-	-	4.70E+0	3.94E+2
Cu-64	-	4.29E+4	-	1.09E+5	-	3.33E+6	2.02E+4
Zn-65	2.11E+9	7.31E+9	-	4.68E+9	-	3.10E+9	3.41E+9
In-69	-	-	-	-	-	-	-
Sr-82	-	-	-	-	-	-	-
Sr-83	-	-	-	-	-	-	5.64E+7
Sr-84	-	-	-	-	-	-	1.91E+1
Sr-85	-	-	-	-	-	-	-
Rb-86	-	4.73E+9	-	-	-	7.00E+8	2.22E+9
Rb-88	-	-	-	-	-	-	-
Rb-89	-	-	-	-	-	-	-
Sr-89	2.67E+9	-	-	-	-	3.18E+8	7.66E+7
Sr-90	6.61E+10	-	-	-	-	1.86E+9	1.63E+10
Sr-91	5.75E+4	-	-	-	-	2.61E+5	2.29E+3
Sr-92	8.95E+1	-	-	-	-	2.20E+1	3.81E+2
Y-90	1.30E+2	-	-	-	-	1.07E+6	1.50E+0
Y-91m	-	-	-	-	-	-	-
Y-91	1.58E+4	-	-	-	-	6.48E+6	4.24E+2
Y-92	1.00E+4	-	-	-	-	2.75E+0	2.90E+6
Y-95	4.30E+1	-	-	-	-	1.31E+4	1.18E+2
Zr-95	1.65E+3	5.22E+2	-	7.67E+2	-	1.20E+6	3.59E+2
Zr-97	7.75E+1	1.53E+1	-	2.32E+1	-	4.15E+4	7.06E+2
Nb-95	1.41E+5	7.80E+4	-	7.57E+4	-	3.34E+8	4.30E+4
Nb-97	-	-	-	-	-	6.34E+8	-
Mo-99	-	4.56E+7	-	1.04E+8	-	8.16E+7	8.69E+6
Tc-99m	5.64E+0	1.57E+1	-	2.34E+2	8.73E+0	1.03E+4	2.04E+2
Tc-101	-	-	-	-	-	-	-
Ru-103	1.81E+3	-	-	6.40E+3	-	1.52E+5	7.75E+2
Ru-105	1.57E+3	-	-	1.97E+2	-	1.26E+0	6.08E+4
Ru-106	3.75E+4	-	-	7.23E+4	-	1.80E+6	4.73E+3
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	9.63E+7	9.11E+7	-	1.74E+8	-	2.56E+10	5.54E+7
Sb-124	4.59E+7	8.46E+5	1.04E+5	-	4.01E+7	9.25E+8	1.79E+7
Sb-125	3.65E+7	3.99E+5	3.49E+4	-	3.21E+7	2.84E+8	8.54E+6
Te-125m	3.00E+7	1.08E+7	8.39E+6	-	-	8.86E+7	4.02E+6
Te-127m	8.44E+7	2.99E+7	2.01E+7	3.42E+8	-	2.10E+8	1.00E+7
Te-127	1.24E+3	4.41E+2	8.59E+2	5.04E+3	-	9.61E+4	2.68E+2
Te-129m	1.11E+6	4.10E+7	3.57E+7	4.62E+8	-	4.15E+8	1.75E+7
Te-129	-	-	-	1.67E+9	-	2.18E+9	-
Te-131m	6.57E+5	3.15E+5	4.74E+5	3.29E+6	-	2.53E+7	2.63E+5
Te-131	-	-	-	-	-	-	-
Te-132	4.28E+6	2.71E+6	2.86E+6	2.60E+7	-	8.58E+7	2.55E+6
I-130	7.49E+5	2.17E+6	1.77E+6	3.34E+6	-	1.67E+6	8.66E+5
I-131	5.38E+8	7.53E+8	2.20E+11	1.30E+9	-	1.49E+8	4.04E+8
I-132	2.90E+1	7.59E+1	2.56E+1	1.20E+0	-	3.31E+1	2.72E+1
I-133	7.24E+6	1.23E+7	1.72E+9	2.15E+7	-	9.30E+6	3.75E+6
I-134	-	-	-	-	-	-	-
I-135	2.47E+4	6.33E+4	4.08E+6	1.00E+5	-	7.03E+4	2.35E+4
Ca-134	9.81E+9	2.31E+10	-	7.34E+9	2.80E+9	2.87E+8	1.07E+10
Ca-136	4.45E+8	1.75E+9	-	9.53E+8	1.50E+8	1.41E+8	1.18E+9
Ca-137	1.34E+10	1.78E+10	-	6.06E+9	2.15E+9	2.53E+8	6.20E+9
Ca-138	-	-	-	-	-	-	-
Ba-139	8.69E+8	-	-	-	-	7.75E+7	2.53E+9
Ba-140	4.85E+7	5.95E+4	-	2.02E+4	4.00E+4	7.49E+7	3.13E+6
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	8.06E+0	1.96E+0	-	-	-	2.27E+5	1.05E+0
La-142	-	-	-	-	-	2.23E+7	-
Ce-141	8.87E+3	5.92E+3	-	2.79E+3	-	1.69E+7	6.81E+2
Ce-143	7.69E+1	5.60E+4	-	2.51E+1	-	1.68E+6	6.25E+0
Ce-144	6.58E+5	2.77E+5	-	1.63E+5	-	1.64E+8	3.54E+4
Pr-143	2.92E+2	1.17E+2	-	5.77E+1	-	9.61E+5	1.45E+1
Pr-144	-	-	-	-	-	-	-
Nd-147	1.81E+2	1.97E+2	-	1.16E+2	-	7.11E+5	1.18E+1
W-187	1.20E+4	9.78E+3	-	-	-	2.63E+6	3.43E+3
Mo-239	6.99E+0	6.59E+1	-	2.07E+0	-	1.06E+5	3.66E+1

Table 3-7 (continued)  
R(10), Grass - Cow - Milk Pathway Dose Factors - CHILD  
(mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) for H-3 and C-14  
( $\text{m}^2 \cdot \text{mrem}/\text{yr}$  per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-3	-	1.57E+3	1.57E+3	1.57E+3	1.57E+3	1.57E+3	1.57E+3
C-14	1.65E+6	3.29E+5	3.29E+5	3.29E+5	3.29E+5	3.29E+5	3.29E+5
Na-24	9.23E+6	9.23E+6	9.23E+6	9.23E+6	9.23E+6	9.23E+6	9.23E+6
P-32	7.77E+10	3.64E+9	-	-	-	2.15E+9	3.00E+9
Cr-51	-	-	5.66E+4	1.55E+4	1.03E+5	5.41E+6	1.02E+5
Mn-54	-	2.0E+7	-	5.87E+6	-	1.76E+7	5.58E+6
Mn-56	-	1.0E+7	-	1.58E+7	-	1.90E+7	2.95E+7
Fe-55	1.12E+8	5.9E+7	-	-	3.35E+7	1.10E+7	1.84E+7
Fe-59	1.20E+8	1.9E+8	-	-	5.65E+7	2.03E+8	9.71E+7
Co-57	-	3.84E+7	-	-	-	3.14E+7	7.77E+6
Co-58	-	1.21E+7	-	-	-	7.08E+7	3.72E+7
Co-60	-	4.32E+7	-	-	-	2.39E+8	1.27E+8
Ni-63	2.96E+10	1.59E+9	-	-	-	1.07E+8	1.01E+9
Ni-65	1.66E+10	1.56E+1	-	-	-	1.91E+1	9.11E+2
Cu-64	-	7.55E+4	-	1.82E+5	-	3.54E+6	4.56E+4
Zn-65	4.13E+9	1.10E+10	-	6.94E+9	-	1.93E+9	8.85E+9
Zn-69	-	-	-	-	-	2.14E+9	-
Br-82	-	-	-	-	-	-	1.15E+8
Br-83	-	-	-	-	-	-	4.69E+1
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
Rb-86	-	8.77E+9	-	-	-	5.64E+8	5.39E+9
Rb-88	-	-	-	-	-	-	-
Rb-89	-	-	-	-	-	-	-
Sr-89	6.62E+9	-	-	-	-	2.56E+8	1.89E+8
Sr-90	1.12E+11	-	-	-	-	1.51E+9	2.83E+10
Sr-91	1.41E+5	-	-	-	-	3.12E+5	5.33E+3
Sr-92	2.19E+0	-	-	-	-	4.14E+1	8.76E+2
Y-90	3.22E+2	-	-	-	-	9.15E+5	8.61E+0
Y-91m	-	-	-	-	-	-	-
Y-91	1.91E+4	-	-	-	-	5.21E+6	1.04E+3
Y-92	2.46E+4	-	-	-	-	7.10E+0	7.03E+6
Y-93	1.06E+0	-	-	-	-	1.57E+4	2.90E+2
Zr-95	3.84E+3	8.45E+2	-	1.21E+3	-	8.81E+5	7.52E+2
Zr-97	1.89E+0	2.72E+1	-	3.91E+1	-	4.13E+6	1.61E+1
Nb-95	3.18E+5	1.24E+5	-	1.16E+5	-	2.29E+8	8.84E+4
Nb-97	-	-	-	-	-	1.45E+6	-
Mo-99	-	8.29E+7	-	1.77E+8	-	8.86E+7	2.05E+7
Tc-99m	1.29E+1	2.54E+1	-	3.68E+2	1.29E+1	1.44E+4	1.20E+2
Tc-101	-	-	-	-	-	-	-
Ru-103	4.29E+1	-	-	1.08E+4	-	1.11E+5	1.65E+3
Ru-105	3.82E+3	-	-	3.16E+2	-	2.49E+0	1.39E+3
Ru-106	9.24E+4	-	-	1.23E+5	-	1.44E+6	1.15E+4
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	2.09E+6	1.41E+8	-	2.63E+8	-	1.68E+10	1.13E+8
Sb-124	1.09E+8	1.41E+8	2.40E+5	-	6.03E+7	6.79E+6	3.81E+7
Sb-125	8.70E+7	1.41E+6	8.06E+4	-	4.85E+7	2.08E+6	1.82E+7
Te-125m	7.38E+7	2.00E+7	2.07E+7	-	-	7.12E+7	9.84E+6
Te-127m	2.08E+8	5.60E+7	4.97E+7	5.93E+8	-	1.68E+8	2.47E+7
Te-127	1.06E+3	8.25E+2	2.12E+3	8.71E+3	-	1.20E+5	6.56E+2
Te-129m	2.72E+8	7.61E+7	8.78E+7	8.00E+8	-	3.32E+8	4.23E+7
Te-129	-	-	-	2.87E+9	-	6.12E+8	-
Te-131m	1.60E+6	5.53E+5	1.14E+6	5.35E+6	-	2.24E+7	5.89E+5
Te-131	-	-	-	-	-	-	-
Te-132	1.02E+7	4.52E+6	6.58E+6	4.20E+7	-	4.55E+7	5.46E+6
I-130	1.75E+6	3.54E+6	3.70E+6	5.29E+6	-	1.66E+6	1.82E+6
I-131	1.30E+9	1.31E+9	4.34E+11	2.15E+9	-	1.17E+8	7.46E+8
I-132	6.86E+1	1.26E+0	5.85E+1	1.93E+0	-	1.48E+0	5.80E+1
I-133	1.76E+7	2.18E+7	4.04E+9	3.63E+7	-	8.77E+6	8.23E+6
I-134	-	-	-	-	-	-	-
I-135	5.84E+4	1.05E+5	9.30E+6	1.61E+5	-	8.00E+4	4.97E+4
Cs-134	2.26E+10	3.71E+10	-	1.15E+10	4.13E+9	2.00E+8	7.83E+9
Cs-136	1.00E+9	2.76E+9	-	1.47E+9	2.19E+8	9.70E+7	1.79E+9
Cs-137	3.22E+10	3.09E+10	-	1.01E+10	3.62E+9	1.93E+8	4.55E+9
Cs-138	-	-	-	-	-	-	-
Ba-139	2.14E+7	-	-	-	-	1.23E+5	6.19E+9
Ba-140	1.17E+8	1.03E+5	-	3.34E+4	6.12E+4	5.94E+7	6.84E+6
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	1.93E+1	6.74E+0	-	-	-	1.88E+5	2.27E+0
La-142	-	-	-	-	-	2.51E+6	-
Ce-141	2.19E+4	1.09E+4	-	4.78E+3	-	1.36E+7	1.62E+3
Ce-143	1.89E+2	1.02E+5	-	4.29E+1	-	1.50E+6	1.48E+1
Ce-144	1.62E+6	5.09E+5	-	2.82E+5	-	1.33E+8	8.66E+4
Pr-143	7.23E+2	2.17E+2	-	1.17E+2	-	7.80E+5	1.59E+1
Pr-144	-	-	-	-	-	-	-
Nd-147	4.45E+2	3.60E+2	-	1.98E+2	-	5.71E+5	2.79E+1
H-187	2.91E+4	1.72E+4	-	-	-	2.42E+6	7.73E+3
Np-239	1.72E+1	1.23E+0	-	1.57E+0	-	9.14E+4	8.68E+1

Table 3-7 (continued)  
R(10), Grass - Cow - Milk Pathway Dose Factors - INFANT  
(mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) for H-3 and C-14  
( $\text{m}^2 \cdot \text{mrem}/\text{yr}$  per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-3	-	2.38E+3	2.38E+3	2.38E+3	2.38E+3	2.38E+3	2.38E+3
C-14	3.22E+6	6.89E+5	6.89E+5	6.89E+5	6.89E+5	6.89E+5	6.89E+5
Na-24	1.61E+7	1.61E+7	1.61E+7	1.61E+7	1.61E+7	1.61E+7	1.61E+7
P-32	1.60E+11	9.42E+9	-	-	-	2.17E+9	6.21E+9
Zr-91	-	-	1.05E+5	2.30E+4	2.05E+5	4.71E+6	1.61E+5
Mn-54	-	3.89E+7	-	8.63E+6	-	1.43E+7	8.83E+6
Mn-56	-	3.21E+2	-	2.76E+2	-	2.91E+0	5.53E+3
Fe-55	1.35E+8	8.72E+7	-	-	4.27E+7	1.11E+7	2.33E+7
Fe-59	2.25E+8	3.93E+8	-	-	1.16E+8	1.88E+8	1.55E+8
Co-57	-	8.95E+6	-	-	-	3.05E+7	1.46E+7
Co-58	-	2.43E+7	-	-	-	6.05E+7	6.06E+7
Co-60	-	8.81E+7	-	-	-	2.10E+8	2.08E+8
Ni-63	3.49E+10	2.16E+9	-	-	-	1.07E+8	1.21E+9
Ni-65	3.51E+0	3.97E+1	-	-	-	3.02E+1	1.81E+1
Cu-64	-	1.88E+5	-	3.17E+5	-	3.85E+6	8.69E+4
Zn-65	5.55E+9	1.90E+10	-	9.23E+9	-	1.61E+10	8.78E+9
Zn-69	-	-	-	-	-	7.36E+9	-
Br-82	-	-	-	-	-	-	1.94E+8
Br-83	-	-	-	-	-	-	9.95E+1
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
Rb-86	-	2.22E+10	-	-	-	5.69E+8	1.10E+10
Rb-88	-	-	-	-	-	-	-
Rb-89	-	-	-	-	-	-	-
Sr-89	1.26E+10	-	-	-	-	2.39E+8	3.61E+8
Sr-90	1.22E+11	-	-	-	-	1.52E+9	3.10E+10
Sr-91	2.94E+5	-	-	-	-	3.48E+5	1.06E+4
Sr-92	4.65E+0	-	-	-	-	5.01E+1	1.73E+1
Y-90	6.80E+2	-	-	-	-	9.39E+5	1.82E+1
Y-91m	-	-	-	-	-	-	-
Y-91	7.33E+4	-	-	-	-	5.26E+6	1.95E+3
Y-92	5.22E+4	-	-	-	-	9.97E+0	1.47E+5
Y-93	2.25E+0	-	-	-	-	1.78E+4	6.13E+2
Zr-95	6.83E+3	1.66E+3	-	1.79E+3	-	8.28E+5	1.18E+3
Zr-97	3.99E+0	6.85E+1	-	6.91E+1	-	4.37E+4	3.13E+1
Nb-95	5.93E+5	2.44E+5	-	1.75E+5	-	2.06E+8	1.41E+5
Nb-97	-	-	-	-	-	3.70E+6	-
Mo-99	-	2.12E+8	-	2.17E+8	-	6.98E+7	4.13E+7
Tc-99m	2.69E+1	5.55E+1	-	5.97E+2	2.90E+1	1.61E+4	7.15E+2
Tc-101	-	-	-	-	-	-	-
Ru-103	8.69E+3	-	-	1.81E+4	-	1.06E+5	2.91E+3
Ru-105	8.06E+3	-	-	5.92E+2	-	3.21E+0	2.71E+3
Ru-106	1.90E+5	-	-	2.25E+5	-	1.44E+6	2.38E+4
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	3.86E+8	2.82E+8	-	4.03E+8	-	1.46E+10	1.86E+8
Sb-124	2.09E+8	3.08E+6	5.56E+5	-	1.31E+8	6.46E+8	6.49E+7
Sb-125	1.49E+8	1.45E+6	1.87E+5	-	9.38E+7	1.99E+8	3.07E+7
Te-125m	1.51E+8	5.04E+7	5.07E+7	-	-	7.18E+7	2.04E+7
Te-127m	4.21E+8	1.40E+8	1.22E+8	1.04E+9	-	1.70E+8	5.10E+7
Te-127	6.50E+3	2.18E+3	5.29E+3	1.59E+4	-	1.36E+5	1.40E+3
Te-129m	5.59E+8	1.92E+8	2.15E+8	1.40E+9	-	3.34E+8	8.62E+7
Te-129	2.08E+9	-	1.75E+9	5.18E+9	-	1.66E+7	-
Te-131m	3.38E+6	1.36E+6	2.76E+6	9.35E+6	-	2.29E+7	1.12E+6
Te-131	-	-	-	-	-	-	-
Te-132	2.10E+7	1.04E+7	1.54E+7	6.51E+7	-	3.85E+7	9.72E+6
I-130	3.60E+6	7.92E+6	8.88E+8	8.70E+6	-	1.70E+6	3.18E+6
I-131	2.72E+9	3.21E+9	1.05E+12	3.75E+9	-	1.15E+8	1.41E+9
I-132	1.42E+0	2.89E+0	1.35E+2	3.22E+0	-	2.34E+0	1.03E+0
I-133	3.72E+7	5.41E+7	9.84E+9	6.36E+7	-	9.16E+6	1.58E+7
I-136	-	-	1.01E+9	-	-	-	-
I-135	1.21E+5	2.41E+5	2.16E+7	2.69E+5	-	8.74E+4	8.80E+4
Cs-134	3.55E+10	6.80E+10	-	1.75E+10	7.18E+9	1.85E+8	6.87E+9
Cs-136	1.96E+9	5.77E+9	-	2.30E+9	4.70E+8	8.76E+7	2.15E+9
Cs-137	5.15E+10	6.02E+10	-	1.62E+10	6.55E+9	1.88E+8	4.27E+9
Cs-138	-	-	-	-	-	-	-
Ba-139	4.55E+7	-	-	-	-	2.88E+5	1.32E+8
Ba-140	2.41E+8	2.41E+5	-	5.73E+4	1.48E+5	5.92E+7	1.24E+7
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	4.03E+1	1.59E+1	-	-	-	1.87E+5	4.09E+0
La-142	-	-	-	-	-	5.21E+6	-
Ce-141	4.33E+4	2.64E+4	-	8.15E+3	-	1.37E+7	2.11E+3
Ce-143	4.00E+2	2.65E+5	-	7.72E+1	-	1.55E+6	3.02E+1
Ce-144	2.33E+6	9.52E+5	-	3.85E+5	-	1.33E+8	1.30E+5
Pr-143	1.49E+3	5.59E+2	-	2.08E+2	-	7.89E+5	7.41E+1
Pr-144	-	-	-	-	-	-	-
Nd-147	8.82E+2	9.06E+2	-	3.49E+2	-	5.74E+5	5.55E+1
N-147	6.12E+4	4.26E+4	-	-	-	2.50E+6	1.47E+4
Nd-239	3.64E+1	3.25E+0	-	6.49E+0	-	9.40E+4	1.84E+0

Table 3-7 (continued)  
 R(10), Grass - Cow - Meat pathway Dose Factors - ADULT  
 (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) for H-3 and C-14  
 ( $\text{m}^3$  \* mrem/yr per  $\mu\text{Ci}/\text{sec}$ ) for others

Radionuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T-Body
H-3	-	3.25E+2	3.25E+2	3.25E+2	3.25E+2	3.25E+2	3.25E+2
C-14	3.33E+5	6.66E+4	6.66E+4	6.66E+4	6.66E+4	6.66E+4	6.66E+4
Na-24	1.84E+3	1.84E+3	1.84E+3	1.84E+3	1.84E+3	1.84E+3	1.84E+3
P-32	4.65E+9	2.89E+8	-	-	-	5.23E+8	1.80E+8
Cr-51	-	-	4.22E+3	1.56E+3	9.38E+3	1.78E+6	7.07E+3
Mn-54	-	9.15E+6	-	2.72E+6	-	2.80E+7	1.75E+6
Rn-56	-	-	-	-	-	-	-
Fe-55	2.93E+8	2.02E+8	-	-	1.13E+8	1.16E+8	4.72E+7
Fe-59	2.67E+8	6.27E+8	-	-	1.75E+8	2.09E+9	2.40E+8
Co-57	-	5.64E+6	-	-	-	1.43E+8	9.37E+6
Co-58	-	1.83E+7	-	-	-	-	-
Co-60	-	7.52E+7	-	-	-	3.70E+8	4.10E+7
Ni-63	1.89E+10	1.31E+9	-	-	-	1.41E+9	1.66E+8
Ni-65	-	-	-	-	-	2.73E+8	6.33E+8
Cu-64	-	2.95E+7	-	7.45E+7	-	2.52E+5	1.39E+7
Zn-65	3.56E+8	1.13E+9	-	7.57E+8	-	7.13E+8	5.12E+8
Zn-69	-	-	-	-	-	-	-
Br-82	-	-	-	-	-	-	-
Br-83	-	-	-	-	-	1.44E+3	1.26E+3
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
Rb-86	-	4.87E+6	-	-	-	-	-
Rb-88	-	-	-	-	-	9.60E+7	2.27E+8
Rb-89	-	-	-	-	-	-	-
Sr-89	3.01E+8	-	-	-	-	-	-
Sr-90	1.24E+10	-	-	-	-	4.84E+7	8.65E+6
Sr-91	-	-	-	-	-	3.59E+8	3.05E+9
Sr-92	-	-	-	-	-	1.38E+9	-
Y-90	1.07E+2	-	-	-	-	-	-
Y-91m	-	-	-	-	-	1.13E+6	2.86E+0
Y-91	1.13E+6	-	-	-	-	-	-
Y-92	-	-	-	-	-	6.24E+8	3.03E+4
Y-93	-	-	-	-	-	-	-
Zr-95	1.88E+6	6.04E+5	-	9.48E+5	-	2.08E+7	-
Zr-97	1.83E+5	3.69E+6	-	5.58E+6	-	1.91E+9	4.09E+5
Nb-95	2.29E+6	1.28E+6	-	1.26E+6	-	1.14E+0	1.69E+6
Nb-97	-	-	-	-	-	7.75E+9	6.86E+5
Nb-99	-	1.09E+5	-	2.46E+5	-	-	-
Tc-99m	-	-	-	-	-	2.52E+5	2.07E+4
Tc-101	-	-	-	-	-	-	-
Ru-103	1.06E+8	-	-	4.03E+8	-	1.23E+10	4.55E+7
Ru-105	-	-	-	-	-	-	-
Ru-106	2.80E+9	-	-	5.40E+9	-	1.81E+11	3.54E+8
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	6.69E+6	6.19E+6	-	1.22E+7	-	2.52E+9	3.67E+6
Sb-124	1.98E+7	3.74E+5	4.80E+4	-	1.54E+7	5.62E+8	7.85E+6
Sb-125	1.91E+7	2.13E+5	1.94E+4	-	1.47E+7	2.10E+8	4.54E+6
Te-125m	3.59E+8	1.30E+8	1.08E+8	1.46E+9	-	1.43E+9	4.81E+7
Te-127m	1.12E+9	3.99E+8	2.85E+8	4.53E+9	-	3.74E+9	1.36E+8
Te-127	-	-	-	1.09E+9	-	-	-
Te-129m	1.74E+9	4.27E+8	3.93E+8	4.77E+9	-	2.10E+8	-
Te-129	-	-	-	-	-	5.76E+9	1.81E+8
Te-131m	4.51E+2	2.21E+2	3.50E+2	2.24E+3	-	-	-
Te-131	-	-	-	-	-	2.19E+4	1.84E+2
Te-132	1.40E+6	9.07E+5	1.00E+6	8.73E+6	-	-	-
I-130	2.35E+6	6.94E+6	9.88E+4	1.08E+5	-	4.29E+7	8.51E+5
I-131	1.08E+7	1.54E+7	5.05E+9	2.64E+7	-	5.98E+6	2.74E+6
I-132	-	-	-	-	-	4.07E+6	8.83E+6
I-133	4.30E-1	7.47E-1	1.10E+2	1.30E+0	-	6.72E-1	2.28E-1
I-134	-	-	-	-	-	-	-
I-135	-	-	-	-	-	-	-
Cs-134	6.57E+8	1.56E+9	-	5.06E+8	1.68E+8	2.74E+7	1.28E+9
Cs-136	1.18E+7	4.67E+7	-	2.60E+7	3.56E+6	5.30E+6	3.36E+7
Cs-137	8.72E+8	1.19E+9	-	4.05E+8	1.15E+8	2.31E+7	7.81E+8
Cs-138	-	-	-	-	-	-	-
Ba-139	-	-	-	-	-	-	-
Ba-140	2.88E+7	3.61E+4	-	1.23E+4	2.07E+4	5.92E+7	1.89E+6
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	3.60E+2	1.81E+2	-	-	-	1.33E+3	4.79E-3
La-142	-	-	-	-	-	-	-
Ce-141	1.40E+4	9.48E+3	-	4.40E+3	-	3.62E+7	1.08E+3
Ce-143	2.09E+2	1.55E+1	-	6.80E-3	-	5.78E+2	1.71E-3
Ce-144	1.46E+6	6.09E+5	-	3.61E+5	-	4.93E+8	7.83E+4
Pr-143	2.13E+4	8.54E+3	-	4.93E+3	-	9.33E+7	1.06E+3
Pr-144	-	-	-	-	-	-	-
Nd-147	7.08E+3	8.18E+3	-	4.78E+3	-	3.93E+7	4.90E+2
Nd-187	2.16E+2	1.81E+2	-	-	-	5.92E+0	6.32E-3
Np-239	2.56E-1	2.51E-2	-	7.84E-2	-	5.15E-3	1.39E-2



Table 3-7 (continued)  
R(10), Grass - Cow - Meat Pathway Dose Factors - TEENAGER  
(mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) for H-3 and C-14  
( $\text{m}^3 \times \text{mrem}/\text{yr}$  per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-3	-	1.94E+2	1.94E+2	1.94E+2	1.94E+2	1.94E+2	1.94E+2
C-14	2.81E+5	5.62E+4	5.62E+4	5.62E+4	5.62E+4	5.62E+4	5.62E+4
Na-24	1.47E+3	1.47E+3	1.47E+3	1.47E+3	1.47E+3	1.47E+3	1.47E+3
P-32	3.93E+9	2.44E+8	-	-	-	3.30E+8	1.52E+8
Cr-51	-	-	3.14E+3	1.24E+3	8.07E+3	9.50E+5	5.65E+3
Mn-54	-	6.98E+6	-	2.08E+6	-	1.43E+7	1.38E+6
Mn-56	-	-	-	-	-	-	-
Fe-55	2.38E+8	1.69E+8	-	-	1.07E+8	7.30E+7	3.93E+7
Fe-59	2.13E+8	4.98E+8	-	-	1.57E+8	1.18E+9	1.92E+8
Co-57	-	4.53E+6	-	-	-	8.45E+7	7.59E+6
Co-58	-	1.41E+7	-	-	-	1.94E+8	3.25E+7
Co-60	-	5.83E+7	-	-	-	7.60E+8	1.31E+8
Ni-63	1.52E+10	1.07E+9	-	-	-	1.71E+8	5.15E+6
Ni-65	-	-	-	-	-	-	-
Cu-64	-	2.41E+7	-	6.10E+7	-	1.87E+5	1.13E+7
Zn-65	2.50E+8	8.69E+8	-	5.56E+8	-	3.68E+8	4.05E+8
Zn-69	-	-	-	-	-	-	-
Br-82	-	-	-	-	-	-	9.98E+2
Br-83	-	-	-	-	-	-	-
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
Rb-86	-	4.06E+8	-	-	-	6.01E+7	1.91E+8
Rb-88	-	-	-	-	-	-	-
Rb-89	-	-	-	-	-	-	-
Sr-89	2.54E+8	-	-	-	-	3.03E+7	7.29E+6
Sr-90	8.05E+9	-	-	-	-	2.26E+8	1.99E+9
Sr-91	-	-	-	-	-	1.10E+9	-
Sr-92	-	-	-	-	-	-	-
Y-90	8.98E+1	-	-	-	-	7.40E+5	2.42E+0
Y-91m	-	-	-	-	-	-	-
Y-91	9.56E+5	-	-	-	-	3.92E+8	2.56E+4
Y-92	-	-	-	-	-	-	-
Y-93	-	-	-	-	-	-	-
Zr-95	1.51E+6	4.76E+5	-	6.99E+5	-	1.69E+7	-
Zr-97	1.53E+5	3.02E+6	-	4.58E+6	-	1.10E+9	3.27E+5
Nb-95	1.79E+6	9.94E+5	-	9.64E+5	-	8.18E+1	1.39E+6
Nb-97	-	-	-	-	-	4.25E+9	5.47E+5
Mo-99	-	8.98E+4	-	2.06E+5	-	1.61E+5	1.71E+4
Tc-99m	-	-	-	-	-	-	-
Tc-101	-	-	-	-	-	-	-
Ru-103	8.60E+7	-	-	3.03E+8	-	7.18E+9	3.68E+7
Ru-105	-	-	-	-	-	-	-
Ru-106	2.36E+9	-	-	4.55E+9	-	1.13E+11	2.97E+8
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	5.06E+6	4.79E+6	-	9.14E+6	-	1.35E+9	2.91E+6
Sb-124	1.62E+7	2.98E+5	3.67E+4	-	1.41E+7	3.26E+8	6.31E+6
Sb-125	1.56E+7	1.71E+5	1.49E+4	-	1.37E+7	1.22E+8	3.56E+6
Te-125m	3.03E+8	1.09E+8	8.47E+7	-	-	8.94E+8	4.05E+7
Te-127m	9.41E+8	3.34E+8	2.24E+8	3.82E+9	-	2.35E+9	1.12E+8
Te-127	-	-	-	-	-	1.75E+8	-
Te-129m	9.58E+5	3.56E+8	3.09E+8	4.01E+8	-	3.60E+9	1.52E+8
Te-129	-	-	-	-	-	-	-
Te-131m	3.76E+2	1.80E+2	2.71E+2	1.88E+3	-	1.45E+4	1.50E+2
Te-131	-	-	-	-	-	-	-
Te-132	1.15E+6	7.26E+5	7.66E+5	6.97E+6	-	2.30E+7	6.84E+5
I-130	1.89E+6	5.48E+6	4.47E+4	8.44E+6	-	4.21E+6	2.19E+6
I-131	8.95E+6	1.25E+7	3.66E+9	2.16E+7	-	2.48E+6	6.73E+6
I-132	-	-	-	-	-	-	-
I-133	3.59E-1	6.10E-1	8.51E+1	1.07E+0	-	4.61E+1	1.88E-1
I-134	-	-	-	-	-	-	-
I-135	-	-	-	-	-	-	-
Ca-134	5.23E+8	1.23E+9	-	3.91E+8	1.49E+8	1.53E+7	5.71E+8
Ca-136	9.22E+6	3.63E+7	-	1.97E+7	3.11E+6	2.92E+6	2.44E+7
Ca-137	7.24E+8	9.63E+8	-	3.28E+8	1.27E+8	1.37E+7	3.38E+8
Ca-138	-	-	-	-	-	-	-
Ba-139	-	-	-	-	-	-	-
Ba-140	2.38E+7	2.91E+4	-	9.88E+3	1.96E+4	3.67E+7	1.53E+6
Sr-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	2.96E+2	1.45E+2	-	-	-	8.35E+2	3.87E-3
La-142	-	-	-	-	-	-	-
Ce-141	1.18E+4	7.86E+3	-	3.70E+3	-	2.25E+7	9.03E+2
Ce-143	1.76E+2	1.28E+1	-	5.74E-3	-	3.85E+2	1.43E-3
Ce-144	1.23E+6	5.08E+5	-	3.04E+5	-	3.09E+8	6.60E+4
Pr-143	1.79E+4	7.15E+3	-	4.16E+3	-	5.90E+7	8.92E+2
Pr-144	-	-	-	-	-	-	-
Nd-147	6.24E+3	6.79E+3	-	3.98E+3	-	2.45E+7	4.06E+2
W-187	1.81E+2	1.48E-2	-	-	-	3.99E+0	5.17E-3
Re-229	2.23E-1	2.11E-2	-	8.61E-2	-	3.39E+3	1.17E-2



Table 3-7 (continued)  
R(10), Grass - Cow - Meat Pathway Dose Factors - CHILD  
(mrem/yr per  $\mu\text{Ci}/\text{M3}$ ) for H-3 and C-14  
(m2 \* mrem/yr per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-3	-	2.34E+2	2.34E+2	2.34E+2	2.34E+2	2.34E+2	2.34E+2
C-14	5.29E+5	1.06E+5	1.06E+5	1.06E+5	1.06E+5	1.06E+5	1.06E+5
Na-24	2.34E+3	2.34E+3	2.34E+3	2.34E+3	2.34E+3	2.34E+3	2.34E+3
P-32	7.41E+9	3.47E+9	-	-	-	2.05E+8	2.86E+8
Ce-51	-	-	4.89E+3	1.34E+3	8.93E+3	4.67E+5	8.61E+3
Mn-54	-	7.99E+6	-	2.24E+6	-	6.70E+6	2.13E+6
Rn-56	-	-	-	-	-	-	-
Fe-55	4.51E+1	2.42E+6	-	-	1.37E+8	4.49E+7	7.51E+7
Fe-59	3.78E+1	6.12E+6	-	-	1.77E+8	6.37E+6	3.05E+6
Co-57	-	5.92E+6	-	-	-	4.85E+7	1.20E+7
Co-58	-	1.65E+7	-	-	-	9.60E+7	5.04E+7
Co-60	-	6.93E+7	-	-	-	3.84E+8	2.04E+8
Ni-63	2.91E+10	1.56E+9	-	-	-	1.05E+8	9.91E+8
Ni-65	-	-	-	-	-	-	-
Cu-64	-	3.24E+7	-	7.82E+7	-	1.52E+5	1.96E+7
Zn-65	3.75E+8	1.00E+9	-	6.30E+8	-	1.76E+8	6.22E+8
Zn-69	-	-	-	-	-	-	-
Br-82	-	-	-	-	-	-	1.56E+3
Br-83	-	-	-	-	-	-	-
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
U-86	-	5.76E+8	-	-	-	3.71E+7	3.54E+8
Rb-86	-	-	-	-	-	-	-
Rb-89	-	-	-	-	-	-	-
Sr-89	4.82E+8	-	-	-	-	1.86E+7	1.38E+7
Sr-90	1.04E+10	-	-	-	-	1.40E+8	2.84E+9
Sr-91	-	-	-	-	-	1.01E+9	-
Sr-92	-	-	-	-	-	-	-
Y-90	1.70E+2	-	-	-	-	4.84E+5	4.55E+0
Y-91m	-	-	-	-	-	-	-
Y-91	1.81E+6	-	-	-	-	2.41E+8	4.83E+4
Y-92	-	-	-	-	-	-	-
Y-93	-	-	-	-	-	1.55E+7	-
Zr-95	2.68E+6	5.89E+5	-	8.43E+5	-	6.14E+8	5.24E+5
Zr-97	2.84E+5	4.10E+6	-	5.89E+6	-	6.21E+1	2.42E+6
Nb-95	3.09E+6	1.20E+6	-	1.13E+6	-	2.23E+9	8.61E+5
Nb-97	-	-	-	-	-	-	-
Mo-99	-	1.25E+5	-	2.67E+5	-	1.03E+5	3.09E+4
Tc-99m	-	-	-	-	-	-	-
Tc-101	-	-	-	-	-	-	-
Ru-103	1.56E+8	-	-	3.92E+8	-	4.02E+9	5.98E+7
Ru-105	-	-	-	-	-	-	-
Ru-106	4.44E+9	-	-	5.99E+9	-	6.90E+10	5.54E+8
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	8.40E+6	5.67E+6	-	1.06E+7	-	4.75E+6	4.33E+6
Sb-124	2.93E+7	3.80E+5	6.46E+4	-	1.62E+7	1.83E+8	1.03E+7
Sb-125	2.85E+7	2.19E+5	2.64E+4	-	1.59E+7	8.80E+7	5.96E+6
Te-123m	5.69E+6	1.54E+8	1.60E+8	-	-	5.49E+8	7.59E+7
Te-127m	1.77E+9	4.78E+8	4.24E+8	5.06E+9	-	1.44E+9	2.11E+8
Te-127	-	-	-	1.21E+9	-	1.66E+8	-
Te-129m	1.81E+9	5.04E+8	5.62E+8	5.30E+9	-	2.20E+9	2.80E+6
Te-129	-	-	-	-	-	-	-
Te-131m	7.00E+2	2.42E+2	4.98E+2	2.34E+3	-	9.82E+3	2.58E+2
Te-131	-	-	-	-	-	-	-
Te-132	2.09E+6	9.27E+5	1.35E+6	8.60E+6	-	9.33E+6	1.12E+6
I-130	3.39E+6	6.85E+6	7.54E+4	1.02E+5	-	3.20E+6	3.53E+6
I-131	1.66E+7	1.67E+7	5.52E+9	2.74E+7	-	1.49E+6	9.49E+6
I-132	-	-	-	-	-	-	-
I-133	6.68E+1	8.26E+1	1.53E+2	1.38E+0	-	3.33E+1	3.12E+1
I-134	-	-	-	-	-	-	-
I-135	-	-	-	-	-	-	-
Ca-134	9.22E+6	1.51E+9	-	4.69E+6	1.68E+6	8.15E+6	3.19E+6
Ca-136	1.59E+7	4.37E+7	-	2.33E+7	3.47E+6	1.54E+6	2.83E+7
Ca-137	1.33E+9	1.28E+9	-	4.16E+8	1.30E+8	7.99E+6	1.88E+8
Ca-138	-	-	-	-	-	-	-
Ba-139	-	-	-	-	-	-	-
Ba-140	4.39E+7	3.85E+4	-	1.25E+4	2.29E+4	2.22E+7	2.56E+6
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	5.41E+2	1.89E+2	-	-	-	5.27E+2	6.38E+3
La-142	-	-	-	-	-	-	-
Ce-141	2.22E+4	1.11E+4	-	4.84E+3	-	1.38E+7	1.64E+3
Ce-143	3.30E+2	1.79E+1	-	7.51E+3	-	2.62E+2	2.59E+3
Ce-144	2.32E+6	7.26E+5	-	4.02E+5	-	1.89E+8	1.24E+5
Pr-143	3.39E+4	1.02E+4	-	5.51E+3	-	3.66E+7	1.60E+3
Pr-144	-	-	-	-	-	-	-
Nd-147	1.17E+4	9.48E+3	-	5.20E+3	-	1.50E+7	7.34E+2
H-187	3.36E+2	1.99E+2	-	-	-	2.79E+0	8.92E+3
Mn-71m	4.70E+1	3.02E+2	-	8.71E+2	-	2.23E+3	2.12E+2

Table 3-7 (continued)  
 R<sub>10</sub> Vegetation Pathway Dose Factors - ADULT  
 (mrem/yr per  $\mu\text{Ci}/\text{m}^2$ ) for H-3 and C-14  
 ( $\text{m}^2 \cdot \text{mrem}/\text{yr}$  per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T. Body
H-3	-	2.26E+3	2.26E+3	2.26E+3	2.26E+3	2.26E+3	2.26E+3
C-14	8.97E+5	1.79E+5	1.79E+5	1.79E+5	1.79E+5	1.79E+5	1.79E+5
Na-24	2.76E+5	2.76E+5	2.76E+5	2.76E+5	2.76E+5	2.76E+5	2.76E+5
P-32	1.40E+9	6.72E+7	-	-	-	1.58E+6	2.42E+7
Cr-51	-	-	2.79E+4	1.03E+4	6.19E+4	1.17E+7	4.66E+4
Mn-54	-	3.11E+8	-	9.27E+7	-	9.54E+6	5.94E+7
Mn-56	-	1.61E+1	-	2.04E+1	-	5.13E+2	2.85E+0
Fe-55	2.09E+6	1.45E+6	-	-	8.06E+7	8.29E+7	3.37E+7
Fe-59	1.27E+8	2.99E+8	-	-	8.25E+7	9.96E+6	1.14E+8
Co-57	-	1.17E+7	-	-	-	2.97E+6	1.95E+7
Co-58	-	3.09E+7	-	-	-	4.26E+8	6.92E+7
Co-60	-	1.67E+8	-	-	-	3.14E+9	3.69E+8
Ni-63	1.04E+10	7.21E+8	-	-	-	1.50E+8	3.49E+8
Ni-65	6.15E+1	7.99E+0	-	-	-	2.03E+2	3.65E+0
Cu-64	-	9.27E+3	-	2.34E+4	-	7.90E+5	4.35E+3
Zn-65	3.17E+8	1.01E+9	-	6.75E+4	-	6.36E+8	4.56E+8
Zn-69	8.75E+6	1.67E+5	-	1.09E+5	-	2.51E+6	1.16E+6
Br-82	-	-	-	-	-	1.73E+6	1.51E+6
Br-83	-	-	-	-	-	4.63E+0	3.21E+0
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
Rb-86	-	2.19E+8	-	-	-	4.32E+7	1.02E+8
Rb-88	-	-	-	-	-	-	-
Rb-89	-	-	-	-	-	-	-
Sr-89	9.96E+9	-	-	-	-	1.60E+9	2.66E+8
Sr-90	6.05E+11	-	-	-	-	1.75E+10	1.48E+11
Sr-91	3.20E+5	-	-	-	-	1.52E+6	1.29E+4
Sr-92	4.27E+2	-	-	-	-	8.46E+3	1.85E+1
Y-90	1.33E+4	-	-	-	-	1.41E+8	3.56E+2
Y-91m	5.83E+9	-	-	-	-	1.71E+8	-
Y-91	5.13E+6	-	-	-	-	2.82E+9	1.37E+6
Y-92	9.01E+1	-	-	-	-	1.58E+4	2.63E+2
Y-93	1.74E+2	-	-	-	-	5.52E+6	4.80E+0
Zr-95	1.19E+6	3.61E+5	-	5.97E+5	-	1.21E+9	2.58E+5
Zr-97	3.32E+2	6.73E+1	-	1.02E+2	-	2.08E+7	3.08E+1
Nb-95	1.42E+5	7.91E+4	-	7.81E+4	-	4.80E+8	4.25E+4
Nb-97	2.90E+6	7.34E+7	-	8.56E+7	-	2.71E+3	2.68E+7
Mo-99	-	6.25E+6	-	1.41E+7	-	1.45E+7	1.19E+6
Tc-99m	3.06E+0	8.66E+0	-	1.32E+2	4.24E+0	5.12E+3	1.10E+2
Tc-101	-	-	-	-	-	-	-
Ru-103	4.80E+6	-	-	1.83E+7	-	5.61E+8	2.07E+6
Ru-105	5.39E+1	-	-	6.96E+2	-	3.30E+4	2.13E+1
Ru-106	1.93E+8	-	-	3.72E+8	-	1.25E+10	2.44E+7
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Aq-110m	1.06E+7	9.76E+6	-	1.92E+7	-	3.98E+9	5.80E+6
Sb-124	1.04E+8	1.96E+6	2.52E+5	-	8.08E+7	2.95E+9	4.11E+7
Sb-125	1.26E+8	1.52E+6	4.39E+5	-	1.05E+8	1.50E+9	3.25E+7
Te-125m	9.46E+7	3.50E+7	2.90E+7	3.93E+6	-	3.86E+8	1.29E+7
Te-127m	3.49E+8	1.25E+8	8.92E+7	1.42E+9	-	1.17E+9	4.26E+7
Te-127	5.76E+3	2.07E+3	4.27E+3	2.35E+4	-	4.54E+5	1.25E+3
Te-129m	2.55E+8	9.50E+7	8.75E+7	1.06E+9	-	1.28E+9	4.03E+7
Te-129	6.65E+4	2.50E+4	5.10E+4	2.79E+3	-	5.02E+4	1.62E+4
Te-131m	9.12E+5	4.46E+5	7.06E+5	4.52E+6	-	4.43E+7	3.72E+5
Te-131	-	-	-	-	-	-	-
Te-132	4.29E+6	2.77E+6	3.06E+6	2.67E+7	-	1.31E+3	3.60E+6
I-130	3.96E+5	1.17E+6	9.90E+7	1.82E+6	-	1.01E+6	4.61E+5
I-131	6.09E+7	1.16E+8	3.79E+10	1.98E+8	-	3.05E+7	6.63E+7
I-132	5.74E+1	1.54E+2	5.38E+3	2.45E+2	-	2.89E+1	5.38E+1
I-133	2.12E+6	3.69E+6	5.42E+6	6.44E+6	-	3.31E+6	1.12E+6
I-134	1.06E+4	2.88E+4	5.00E+3	4.59E+4	-	2.51E+7	1.03E+4
I-135	4.08E+4	1.07E+5	7.04E+6	1.71E+5	-	1.21E+5	3.94E+4
Ca-134	4.66E+9	1.11E+10	-	3.59E+9	1.19E+9	1.94E+8	9.07E+8
Ca-136	4.20E+7	1.66E+8	-	9.24E+7	1.27E+7	1.85E+7	1.19E+8
Ca-137	6.36E+9	8.70E+9	-	2.95E+9	9.81E+8	1.62E+8	5.70E+9
Ca-138	-	-	-	-	-	-	-
Ba-139	2.95E+2	2.10E+5	-	1.96E+5	1.19E+5	5.23E+2	8.64E+4
Ba-140	1.29E+8	1.62E+5	-	5.49E+4	9.25E+4	2.65E+8	6.43E+6
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	1.97E+3	9.92E+2	-	-	-	7.28E+7	2.62E+2
La-142	1.40E+4	6.35E+5	-	-	-	4.64E+1	1.58E+5
Ce-141	1.96E+5	1.33E+5	-	6.17E+4	-	5.08E+8	1.51E+4
Ce-143	1.00E+3	7.42E+5	-	3.26E+2	-	2.77E+7	6.21E+1
Ce-144	3.29E+7	1.38E+7	-	8.16E+6	-	1.11E+10	1.77E+6
Pr-143	6.34E+4	2.54E+4	-	1.47E+4	-	2.78E+8	3.14E+3
Pr-144	-	-	-	-	-	-	-
Nd-147	3.34E+6	3.86E+4	-	2.25E+4	-	1.85E+8	2.31E+3
W-187	3.82E+4	3.19E+4	-	-	-	1.05E+7	1.12E+4
Np-239	1.42E+3	1.40E+2	-	4.37E+2	-	2.87E+7	7.72E+1

Table 3-7 (continued)  
 R. Vegetation Pathway Dose Factors - TEENAGER  
 (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ ) for H-3 and C-14  
 ( $\text{m}^2 \cdot \text{mrem}/\text{yr}$  per  $\mu\text{Ci}/\text{sec}$ ) for others

Nuclide	Bone	Liver	Thyroid	Kidney	Lung	GI-LLI	T-Pod
H-3	-	2.59E+3	2.59E+3	2.59E+3	2.59E+3	2.59E+3	2.59E+3
C-14	1.45E+6	2.91E+5	2.91E+5	2.91E+5	2.91E+5	2.91E+5	2.91E+5
Na-24	2.45E+5	2.45E+5	2.45E+5	2.45E+5	2.45E+5	2.45E+5	2.45E+5
P-32	1.61E+9	9.96E+7	-	-	1.35E+8	6.23E+7	-
Cr-51	-	-	3.44E+4	1.36E+4	6.05E+4	1.04E+7	6.20E+4
Mn-54	-	4.52E+8	-	1.35E+8	-	9.27E+8	8.97E+7
Mn-56	-	1.45E+1	-	1.83E+1	-	9.54E+2	2.58E+0
Fe-55	3.25E+4	2.31E+8	-	-	1.44E+8	9.98E+7	5.38E+7
Fe-59	1.81E+8	4.22E+8	-	-	1.33E+8	9.98E+8	1.63E+8
Co-57	-	1.79E+7	-	-	-	3.34E+8	3.00E+7
Co-58	-	4.38E+7	-	-	-	6.04E+8	1.01E+8
Co-60	-	2.49E+8	-	-	-	3.24E+9	5.60E+8
Ni-63	1.61E+10	1.13E+9	-	-	-	1.81E+8	5.45E+8
Ni-65	5.73E+1	7.32E+0	-	-	-	3.97E+2	3.33E+0
Cu-64	-	8.40E+3	-	2.12E+4	-	6.51E+5	3.95E+3
Zn-65	4.24E+8	1.47E+9	-	9.41E+8	-	6.23E+8	6.86E+8
Zn-69	8.19E+6	1.56E+5	-	1.02E+5	-	2.88E+5	1.09E+6
Br-82	-	-	-	-	-	-	1.33E+8
Br-83	-	-	-	-	-	-	3.01E+0
Br-84	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-
Rb-86	-	2.73E+8	-	-	-	4.05E+7	1.28E+8
Rb-88	-	-	-	-	-	-	-
Rb-89	-	-	-	-	-	-	-
Sr-89	1.51E+10	-	-	-	-	1.50E+9	4.33E+8
Sr-90	7.51E+11	-	-	-	-	2.11E+10	1.85E+11
Sr-91	2.99E+5	-	-	-	-	1.36E+6	1.19E+4
Sr-92	3.97E+2	-	-	-	-	1.01E+4	1.69E+1
Y-90	1.24E+4	-	-	-	-	1.02E+8	3.34E+2
Y-91m	5.43E+9	-	-	-	-	2.56E+7	-
Y-91	7.87E+6	-	-	-	-	3.23E+9	2.11E+5
Y-92	8.47E+1	-	-	-	-	2.32E+4	2.45E+2
Y-93	1.63E+2	-	-	-	-	4.98E+6	4.47E+0
Zr-95	1.74E+6	5.49E+5	-	8.07E+5	-	1.27E+9	3.78E+5
Zr-97	3.09E+2	6.11E+1	-	9.26E+1	-	1.65E+7	2.81E+1
Nb-95	1.92E+5	1.06E+5	-	1.03E+5	-	4.55E+8	5.86E+4
Nb-97	2.69E+6	6.67E+7	-	7.80E+7	-	1.59E+2	2.44E+7
Mo-99	-	5.74E+6	-	1.31E+7	-	1.03E+7	1.09E+6
Tc-99m	2.70E+0	7.54E+0	-	1.12E+2	4.19E+0	4.95E+3	9.77E+1
Tc-101	-	-	-	-	-	-	-
Ru-103	6.47E+6	-	-	2.42E+7	-	5.74E+8	2.94E+6
Ru-105	5.00E+1	-	-	6.31E+2	-	4.04E+4	1.94E+1
Ru-106	3.09E+8	-	-	5.97E+8	-	1.48E+10	3.90E+7
Rh-103m	-	-	-	-	-	-	-
Rh-106	-	-	-	-	-	-	-
Ag-110m	1.52E+7	1.44E+7	-	2.74E+7	-	4.04E+5	6.74E+6
Sb-124	1.55E+8	2.85E+6	3.51E+5	-	1.35E+8	3.11E+9	6.03E+7
Sb-125	2.14E+8	2.34E+6	2.04E+5	-	1.88E+8	1.66E+9	5.00E+7
Te-125m	1.48E+8	5.34E+7	4.14E+7	-	-	4.37E+8	1.98E+7
Te-127m	5.51E+8	1.96E+8	1.31E+8	2.24E+9	-	1.37E+9	6.56E+7
Te-127	5.43E+3	1.92E+3	3.74E+3	2.20E+4	-	4.19E+5	1.17E+3
Te-129m	3.67E+8	1.36E+8	1.18E+8	1.54E+9	-	1.38E+9	5.81E+7
Te-129	6.22E+4	2.32E+4	4.45E+4	2.61E+3	-	3.40E+3	1.51E+4
Te-131m	8.44E+5	4.05E+5	6.09E+5	4.22E+6	-	3.25E+7	3.38E+5
Te-131	-	-	-	-	-	-	-
Te-132	3.90E+6	2.47E+6	2.60E+6	2.37E+7	-	7.82E+7	2.32E+6
I-130	3.54E+5	1.02E+6	8.35E+7	1.58E+6	-	7.87E+5	4.09E+5
I-131	7.70E+7	1.08E+8	3.14E+10	1.85E+8	-	2.13E+7	5.79E+7
I-132	5.18E+1	1.36E+2	4.57E+3	2.14E+2	-	5.91E+1	4.87E+1
I-133	1.97E+6	3.34E+6	4.64E+8	5.86E+6	-	2.53E+6	1.02E+6
I-134	9.59E+5	2.54E+4	4.24E+3	4.01E+4	-	3.35E+6	9.13E+5
I-135	3.68E+4	9.48E+4	6.10E+6	1.50E+5	-	1.05E+5	3.52E+4
Cs-134	7.09E+9	1.67E+10	-	5.30E+9	2.02E+9	2.08E+8	7.74E+9
Cs-136	4.29E+7	1.69E+8	-	9.19E+7	1.45E+7	1.36E+7	1.15E+8
Cs-137	1.01E+10	1.35E+10	-	4.59E+9	1.78E+9	1.92E+8	4.69E+9
Cs-138	-	-	-	-	-	-	-
Ba-139	2.77E+2	1.95E+5	-	1.64E+5	1.34E+5	2.47E+1	8.08E+4
Ba-140	1.38E+8	1.69E+5	-	5.75E+4	1.14E+5	2.13E+8	8.91E+6
Ba-141	-	-	-	-	-	-	-
Ba-142	-	-	-	-	-	-	-
La-140	1.80E+3	8.84E+2	-	-	-	5.08E+7	2.35E+2
La-142	1.28E+4	5.69E+5	-	-	-	1.73E+0	1.42E+5
Ce-141	2.82E+5	1.88E+5	-	8.86E+4	-	5.38E+8	2.16E+4
Ce-143	9.37E+2	6.82E+5	-	3.06E+2	-	2.05E+7	7.62E+1
Ce-144	5.27E+7	2.18E+7	-	1.30E+7	-	1.33E+10	2.83E+6
Pr-143	7.12E+4	2.84E+4	-	1.65E+4	-	2.34E+8	3.55E+3
Pr-144	-	-	-	-	-	-	-
Nd-147	3.63E+4	3.94E+4	-	2.32E+4	-	1.42E+8	2.36E+3
W-187	3.55E+4	2.90E+4	-	-	-	7.84E+6	1.02E+4
Hf-229	1.38E+3	1.30E+2	-	4.09E+2	-	2.10E+7	7.24E+1

Table 3-7 (continued)  
 $R_{10}$  Ground Plane Pathway Dose Factors  
 ( $m^2 \cdot mrem/yr$  per  $\mu Ci/sec$ )

Nuclide	Any Organ
H-3	-
C-14	-
Na-24	1.21E+7
P-32	-
Cr-51	4.68E+6
Mn-54	1.34E+9
Mn-56	9.05E+5
Fe-55	-
Fe-59	2.75E+8
Co-58	3.82E+8
Co-60	2.16E+10
Ni-63	-
Ni-65	2.97E+5
Cu-64	6.09E+5
Zn-65	7.45E+8
Zn-69	-
Br-83	4.89E+3
Br-84	2.03E+5
Br-85	-
Rb-86	8.98E+6
Rb-88	3.29E+4
Rb-89	1.21E+5
Sr-89	2.16E+4
Sr-90	-
Sr-91	2.19E+6
Sr-92	7.77E+5
Y-90	4.48E+3
Y-91m	1.01E+5
Y-91	1.08E+6
Y-92	1.80E+5
Y-93	1.65E+5
Zr-95	2.48E+6
Zr-97	2.94E+6
Nb-95	1.34E+6
Mo-99	4.05E+6
Tc-99m	1.63E+5
Tc-101	2.04E+4
Ru-103	1.09E+8
Ru-105	6.36E+5
Ru-106	4.21E+8
Rh-103m	-
Rh-106	-
Ag-110m	3.47E+9
Te-125m	1.55E+6
Te-127m	9.17E+4
Te-127	3.00E+3
Te-129m	2.00E+7
Te-129	2.60E+4
Te-131m	8.03E+6
Te-131	2.93E+4
Te-132	4.22E+6
I-130	5.53E+6
I-131	1.72E+7
I-132	1.24E+6
I-133	2.47E+6
I-134	4.49E+5
I-135	2.56E+6
Ce-134	6.75E+9
Ce-136	1.49E+6
Ce-137	1.06E+10
Ce-138	3.59E+5
Ba-139	1.06E+6
Ba-140	2.05E+7
Ba-141	4.18E+6
Ba-142	4.49E+4
La-140	1.91E+7
La-142	7.36E+5
Ce-141	1.36E+7
Ce-143	2.32E+6
Ce-144	6.95E+7
Pr-143	-
Pr-144	1.83E+3
Hd-147	8.40E+6
W-187	2.36E+6
Hg-229	1.71E+6



DAVIS-BESSE NUCLEAR POWER STATION  
ATMOSPHERIC RADIOACTIVE RELEASE PATHWAYS

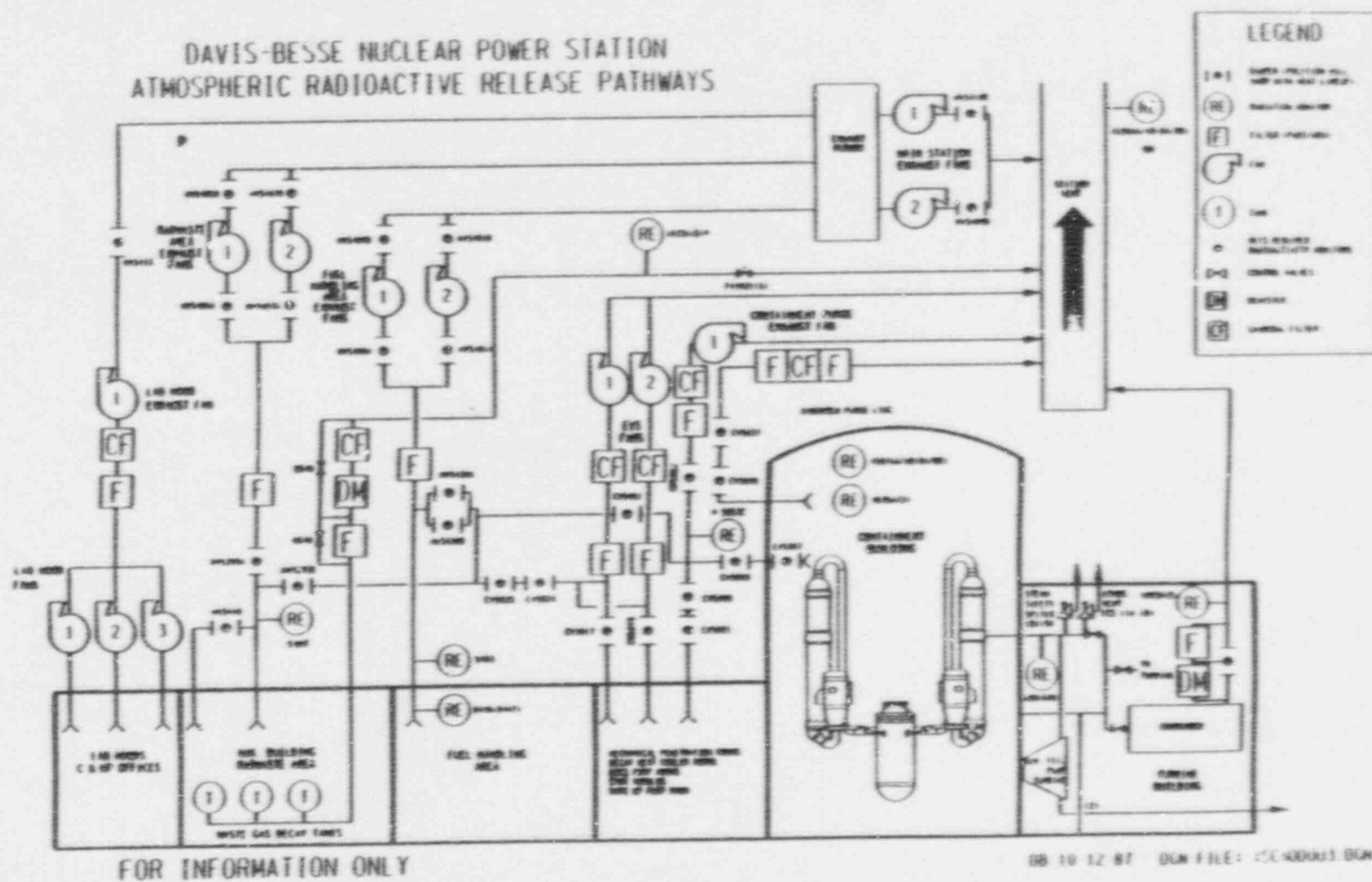


Figure 3-1  
Caseous Radioactive Effluent Monitoring and Processing Diagram



#### 4.0 SPECIAL DOSE ANALYSES

##### 4.1 Doses To Public Due To Activities Inside the SITE BOUNDARY

In accordance with Technical Specification 6.9.1.11 and ODCM Section 7.2, the Semiannual Effluent and Waste Disposal Report submitted within 60 days after January 1 and July 1 of each year shall include an assessment of radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY.

In special instances MEMBERS OF THE PUBLIC are permitted access to the radiological controlled area within the Davis-Besse station. Tours for the public are conducted with the assurance that no individual will receive an appreciable dose (i.e., small fraction of the 40 CFR 190 dose standards).

The Visitor Center located inside the Davis-Besse Administration Building (DBAB) is also accessible to MEMBERS OF THE PUBLIC. Considering the frequency and duration of the visits, the resultant dose would be a small fraction of the calculated maximum SITE BOUNDARY doses. The dose from gaseous effluents as modeled for the DBAB Visitor Center is considered the controlling factor when evaluating doses to MEMBERS OF THE PUBLIC from activities inside the SITE BOUNDARY.

For purposes of assessing the dose to MEMBERS OF THE PUBLIC in accordance with Technical Specification 6.9.1.11 and ODCM Section 7.2, the following exposure assumptions may be used:

- Exposure time for maximum exposed visitor of 20 hours (4 visits, 5 hours per visit).\*
- Annual average meteorological dispersion (conservative, default use of maximum SITE BOUNDARY dispersion) from Table 3-7.

The equations in ODCM Section 4.3 may be used for calculating the potential dose to a MEMBER OF THE PUBLIC for activities inside the SITE BOUNDARY. Based on these assumptions, this dose would be at least a factor of 400 less than the maximum SITE BOUNDARY air dose as calculated in ODCM Section 3.7.

There are no areas onsite accessible to the public where exposure to liquid effluents could occur. Therefore, the modeling of ODCM Section 2.4 conservatively estimates the maximum potential dose to MEMBERS OF THE PUBLIC.

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\* Based on a maximum conservative estimate.

##### 4.2 Doses to MEMBERS OF THE PUBLIC - 40 CFR 190

As required by Technical Specification 6.9.1.11 and ODCM Section 7.2, the Semiannual Effluent and Waste Disposal Report shall also include an assessment of the radiation dose to the likely most exposed MEMBER OF THE PUBLIC for reactor releases and other nearby uranium fuel cycle sources

(including dose contributions from effluents and direct radiation from onsite sources). For the likely most exposed MEMBER OF THE PUBLIC in the vicinity of the Davis-Besse site, the sources of exposure need consider the radioactive effluents and direct exposure contribution from Davis-Besse. No other fuel cycle facilities contribute significantly to the cumulative dose to a MEMBER OF THE PUBLIC in the immediate vicinity of the site. Fermi-2 is the closest fuel cycle facility located about 20 miles to the NNW. Due to environmental dispersion, any routine releases from Fermi-2 would contribute insignificantly to the potential doses in the vicinity of Davis-Besse.

The correlation of measured plant effluents with pathway modeling of this ODCM provide the primary method for demonstrating/evaluating compliance with the limits specified below (40 CFR 190). However, as appropriate, the results of the environmental monitoring program may be used to provide additional data on actual measured levels of radioactive material in the actual pathways of exposure. ODCM Section 4.2.3 discusses the methodology for correlating measured levels of radioactive material in environmental pathway samples with potential doses. Also, results of the land use census may be used to determine actual exposure pathways and locations.

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

With the calculated doses from the releases of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Sections 2.4.1, 3.7.1, and 3.8.1, evaluations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of this Section have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Section 7.3, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

This requirement is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The requirement requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that the dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If a dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other dose requirements for dose limitation of 10 CFR Part 20, as addressed in Sections 2.2 and 3.3.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is a part of the nuclear fuel cycle.

#### 4.2.1 Effluent Dose Calculations

For purposes of implementing the above requirements of determining the cumulative dose contribution from liquid and gaseous effluents in accordance with Sections 2 and 3 and the reporting requirements of Section 7, dose calculations for Davis-Besse may be performed using the calculational methods contained within this ODCM; the conservative controlling pathways and locations of Table 3-7 or the actual pathways and locations as identified by the land use census may be used. Liquid pathway doses may be calculated using equations in ODCM Section 2.4. Doses due to releases of radioiodines, tritium and particulates are calculated based on equations in Section 3.8.

The following equations may be used for calculating the dose to MEMBERS OF THE PUBLIC from releases of noble gases:

$$D_{tb} = 3.17E-08 * \frac{U}{8760} * X/Q * L (K_1 * Q_1) \quad (4-1)$$

and

$$D_a = 3.17E-08 * U * X/Q * L ((L_1 + 1.1 M_1) * Q_1) \quad (4-2)$$

where:

- $D_{tb}$  = total body dose due to gamma emissions for noble gas radionuclides (mrem)
- $D_s$  = skin dose due to gamma and beta emissions for noble gas radionuclides (mrem)
- $U$  = duration of exposure (hr/yr, default values in Table 4-1)
- $X/Q$  = atmospheric dispersion to the offsite location (sec/m<sup>3</sup>)
- $Q_i$  = cumulative release of noble gas radionuclide (i) over the period of interest ( $\mu$ Ci)
- $K_i$  = total body dose factor due to gamma emissions from noble gas radionuclide (i) from Table 3-6 (mrem/yr per  $\mu$ Ci/m<sup>3</sup>)
- $L_i$  = skin dose factor due to beta emissions from noble gas radionuclide (i) from Table 3-6 (mrem/yr per  $\mu$ Ci/m<sup>3</sup>)
- $M_i$  = gamma air dose factor for noble gas radionuclide (i) from Table 3-6 (mrad/yr per  $\mu$ Ci/m<sup>3</sup>)
- 8760 = hours per year
- 1.1 = mrem skin dose per mrad gamma air dose (mrem/mrad)
- $3.17E-08$  =  $1/3.15E+07$  yr/sec

Average annual meteorological dispersion parameters or meteorological conditions concurrent with the release period under evaluation may be used (e.g., quarterly averages or year-specific annual averages).

#### 4.2.2 Direct Exposure Dose Determination - Onsite Sources

Any potentially significant direct exposure contribution from onsite sources to offsite individual doses may be evaluated based on the results of the environmental measurements (e.g., TLD, ion chamber measurements) or by the use of a radiation transport and shielding calculational method. Only during atypical conditions will there exist any potential for significant onsite sources at Davis-Besse that would yield potentially significant offsite doses to a MEMBER OF THE PUBLIC. However, should a situation exist whereby the direct exposure contribution is potentially significant, onsite measurements, offsite measurements and calculational techniques will be used for determination of dose for assessing 40 CFR 190 compliance.

The following simplified method may be used for evaluating the direct dose based on onsite or site boundary measurements:

$$D_{L,\theta} = D_{B,\theta} \frac{(X_{B,\theta})^2}{(X_{L,\theta})^2} \quad (4-3)$$



where:

$D_{B, \theta}$  = direct radiation dose measured at location B (onsite or site boundary) in sector  $\theta$

$D_{L, \theta}$  = extrapolated dose at location L in same sector  $\theta$

$X_{L, \theta}$  = distance to the location L from the radiation source

$X_{B, \theta}$  = distance to location B from the radiation source

#### 4.2.3 Dose Assessment Based on Radiological Environmental Monitoring Data

Normally, the assessment of potential doses to MEMBERS OF THE PUBLIC must be calculated based on the measured radioactive effluents at the plant. The resultant levels of radioactive material in the offsite environment are so minute as to be undetectable. The calculational methods as presented in this ODCM are used for modeling the transport in the environment and the resultant exposure to offsite individuals.

The results of the radiological environmental monitoring program can provide input into the overall assessment of impact of plant operations and radioactive effluents. With measured levels of plant related radioactive material in principal pathways of exposure, a quantitative assessment of potential exposures can be performed. With the monitoring program not identifying any measurable levels, the data provides a qualitative assessment - a confirmatory demonstration of the negligible impact.

Dose modeling can be simplified into three basic parameters that can be applied in using environmental monitoring data for dose assessment.

$$D = C * U * DF \quad (4-4)$$

where:

D = dose or dose commitment

C = concentration in the exposure media, such as air concentration for the inhalation pathway, or fish, vegetation or milk concentration for the ingestion pathway

U = individual exposure to the pathway, such as hr/yr for direct exposure, kg/yr for ingestion pathway

DF = dose conversion factor to convert from an exposure or uptake to an individual dose or dose commitment

The applicability of each of these basic modeling parameters to the use of environmental monitoring data for dose assessment is addressed below:



### Concentration - C

The main value of using environmental sampling data to assess potential doses to individuals is that the data represents actual measured levels of radioactive material in the exposure pathways. This eliminates one main uncertainty in the modeling - the release from the plant and the transport to the environmental exposure medium.

Environmental samples are collected on a routine frequency (e.g., weekly airborne particulate samples, monthly vegetable samples, annual fish samples). To determine the annual average concentration in the environmental medium for use in assessing cumulative dose for the year, an average concentration should be determined based on the sampling frequency and measured levels.

$$\bar{C}_i = \Sigma(C_i * t)/365 \quad (4-5)$$

where:

- $\bar{C}_i$  = average concentration in the sampling medium for the year
- $C_i$  = concentration of each radionuclide (i) measured in the individual sampling medium
- t = period of time that the measured concentration is considered representative of the sampling medium (typically equal to the sampling frequency; e.g., 7 days for weekly samples, 30 days for monthly samples).

If the concentration in the sampling medium is below the detection capabilities (i.e., less than lower limits of detection -LLD), a value of zero should be used for  $C_i$  ( $C_i = 0$ ).

### Exposure - U

Default exposure values (U) as recommended in Regulatory Guide 1.109 are presented in Table 4-1. These values should be used only when specific data applicable to the environmental pathway being evaluated is unavailable.

Also, the routine radiological environmental monitoring program is designed to sample/monitor the environmental media that would provide early indications of any measurable levels in the environment but not necessarily levels to which any individual is exposed. For example, sediment samples are collected in the area of the liquid discharge; typically, no individuals are directly exposed. To apply the measured levels of radioactivity in samples that are not directly applicable to exposure to real individuals, the approach recommended is to correlate the location and measured levels to actual locations of exposure. Hydrological or atmospheric dilution factors can be used to provide reasonable correlations of concentrations (and doses) at other locations. The other alternative is to conservatively assume a hypothetical individual at the sampling location. Doses that are calculated in this manner should be presented as hypothetical and very conservatively determined - actual

exposure would be much less. Samples collected from nearby wells or actual water supply intake (e.g., Port Clinton) should be used for estimating the potential drinking water doses. Other water samples collected, such as near field dilution area, are not applicable to this pathway.

#### Dose Factors - DF

The dose factors are used to convert the intake of the radioactive material to an individual dose commitment. Values of the dose factors are presented in NRC Regulatory Guide 1.109. The use of the Regulatory Guide 1.109 values applicable to the exposure pathway and maximum exposed individual is referenced in Table 4-1.

#### 4.2.4 Use of Environmental TLD for Assessing Doses Due to Noble Gas Releases

Thermoluminescent dosimeters (TLD) are routinely used to assess the direct exposure component of radiation doses in the environment. However, because routine releases of radioactive material (noble gases) are so low, the resultant direct exposure doses are also very low. A study\* performed for the NRC concluded that it is possible to determine a plant contribution to the natural background radiation levels (direct exposure) of around 10 mrem per year (by optimum methods and high precision data). Therefore, for routine releases from nuclear power plants the use of TLD is mainly confirmatory - ensuring actual exposures are within the expected natural background variation.

For releases of noble gases, environmental modeling using plant measured releases and atmospheric transport models as presented in this ODCM represents the best method of assessing potential environmental doses. However, any observed variations in TLD measurements outside the norm should be evaluated.

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\* NUREG/CR-0711, Evaluation of Methods for the Determination of X- and Gamma-Ray Exposure Attributable to a Nuclear Facility Using Environmental TLD Measurements, Gail dePlanque, June 1979, USNRC.

Table 4-1

Recommended Exposure Rates  
in Lieu of Site Specific Data\*

<u>Exposure Pathway</u>	<u>Maximum Exposed Age Group</u>	<u>Exposure Rates</u>	<u>Table Reference for Dose Factors from RG 1.109</u>
<u>Liquid Releases</u>			
Fish	Adult	21 kg/y	E-11
Drinking Water	Adult	730 l/y	E-11
Bottom Sediment	Teen	67 h/y	E-6
<u>Atmospheric Releases</u>			
Inhalation	Teen	8,000 m <sup>3</sup> /y	E-8
Direct Exposure	All	6,100 h/y**	N/A (ODCM Table 3-7)
Leafy Vegetables	Child	26 kg/y	E-13
Fruits, Vegetables & Grain	Teen	630 kg/y	E-12
Milk	Infant	330 l/y	E-14

\* Adapted from Regulatory Guide 1.109, Table E-5

\*\* Net exposure of 6,100 h/y is based on the total 8760 hours per year adjusted by a 0.7 shielding factor as recommended in Regulatory Guide 1.109.

## 5.0 ASSESSMENT OF LAND USE CENSUS DATA

A land use census (LUC) is conducted annually in the vicinity of the Davis-Besse site. This census fulfills two main purposes: 1) meet requirements of TS (as required by 10 CFR 50, Appendix I, Section IV.B.3) for identifying controlling location/pathway for dose assessment of ODCM Section 3.8.1; and (2) provide data on actual exposure pathways for assessing realistic doses to MEMBERS OF THE PUBLIC.

### 5.1 Land Use Census as Required by TS

As required by TS, a land use census shall be conducted during the growing season at least once per twelve months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agricultural authorities. The land use census shall identify within a distance of 8 km (5 miles) the location, in each of the 16 meteorological sectors, of the nearest milk animal, the nearest residence and the nearest garden of greater than 50 m<sup>2</sup> (500 ft<sup>2</sup>) producing broad leaf vegetation. This requirement is provided to ensure that changes in the use of UNRESTRICTED AREAS are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m<sup>2</sup> (500 ft<sup>2</sup>) provides assurance that significant exposure pathways via leafy vegetables, will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables, assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m<sup>2</sup>.

The data from the land use census is used for updating the location/pathway for dose assessment and for updating the Radiological Environmental Monitoring Program. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Section 7.1.

With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Sections 3.8.1, in lieu of a Licensee Event Report, identify the new locations(s) in the next Semiannual Effluent and Waste Disposal Report, pursuant to Section 7.2. With a land use census identifying a locations(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than that at a location from which samples are currently being obtained in accordance with Section 6.1, add the new locations(s) if practical (and readily obtainable) to the Radiological Environmental Monitoring Program within 30 days. The sampling locations(s), excluding the control station location, having a lower calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program. In lieu of a Licensee Event Report and pursuant to Section 7.2, identify the new location(s) in the next Semiannual Effluent and Waste Disposal Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).



The following guidelines shall be used for assessing the results from the land use census to ensure compliance with this Section.

A. Data Compilation

- A.1 Locations and pathways of exposure as identified by the land use census will be compiled for comparison with the current locations as presented in Table 3-4.
- A.2 Changes from the previous year's census will be identified. Also, any location/pathway not currently included in the Radiological Environmental Monitoring Program (Table 6-2) will be identified.
- A.3 Historical, annual average meteorological dispersion parameters ( $X/Q$ ,  $D/Q$ ) for any new location (i.e., location not previously identified and/or evaluated) will be determined. All locations should be evaluated against the same historical meteorological data set.

B. Relative Dose Significance

- B.1 For all new locations, the relative dose significance will be determined by applicable pathways of exposure.
  - B.1.1 Relative dose calculations should be based on a generic radionuclide distribution (e.g., Davis-Besse USAR gaseous effluent source term or past year actual effluents). An I-131 source term dose may be used for assessment of the maximum organ ingestion pathway dose because of its overwhelming contribution to the total dose relative to the other particulates.
  - B.1.2 The pathway dose equations of the ODCM should be used.

C. Data Evaluation

- C.1 The controlling location used in the ODCM Table 3-4 will be verified. If any location/pathway(s) is identified with a higher relative dose, this location/pathway(s) should replace the previously identified controlling location/pathway in Table 3-4. If the previously identified controlling pathway is no longer present, the current controlling location/pathway should be determined.
- C.2 Any changes in either the controlling location/pathway(s) of the ODCM dose calculations (Section 3.7 and Table 3-4) or the Radiological Environmental Monitoring Program (ODCM Section 6.0 and Table 6-2) shall be reported to NRC in accordance with ODCM Section 5.1 and 7.2.

## 5.2 Land Use Census to Support Realistic Dose Assessment

The Land Use Census (LUC) provides data needed to support the special dose analyses of the ODCM Section 4.0. Activities inside the SITE BOUNDARY should be periodically reviewed for dose assessment as required by TS 6.9.1.11 (ODCM Section 4.1). Assessment of realistic doses to MEMBERS OF THE PUBLIC is required by Section 4.0 for demonstrating compliance with the EPA Environmental Dose Standard, 40 CFR 190 (ODCM Section 4.2).

To support these dose assessments, the LUC shall include (a) areas within the SITE BOUNDARY that are accessible to the public; and (b) use of Lake Erie water on and near the site. The scope of the LUC shall include the following:

- Assessment of areas onsite that are accessible to MEMBERS OF THE PUBLIC. Particular attention should be given to assessing exposure times for visits to the Davis-Besse Administration Building. Data should be used for updating ODCM Table 4-1.
- Data on Lake Erie use should be obtained from local and state officials. Reasonable efforts shall be made to identify individual irrigation and potable water users, and industrial and commercial water users whose source is Lake Erie. This data is used to verify the pathways of exposure used in ODCM Section 2.4.

## 6.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Radiological Environmental Monitoring Program (REMP) required by TS provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the higher potential radiation exposures of individuals resulting from the station operations. The sampling and analysis program described in this Section was developed to provide representative measurements of radiation and radioactive materials resulting from station operation in the principal pathways of exposure of MEMBERS OF THE PUBLIC. This monitoring program implements Sections IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent controls by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for the development of this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring.

### 6.1 Program Description

#### 6.1.1 General

The REMP shall be conducted as specified in Table 6-1. This table describes the minimum environmental media to be sampled, the sample collection frequencies, the number of representative samples required, the characteristics of the sampling locations, and the type and frequency of sample analysis. Table 6-2 provides a detailed listing of the sample locations for Davis-Besse which satisfy the requirements of Table 6-1. Maps for each site listed in Table 6-2 are contained in Appendix C. The specific locations used to satisfy the requirements of Table 6-1 may be changed as deemed appropriate by the Radiological Environmental Supervisor. The changes shall be reported in the Annual Radiological Environmental Operating Report and the Semiannual Effluent and Waste Disposal Report as required by Sections 7.1 and 7.2, respectively. If the changes are to be permanent, Table 6-2 and Appendix C shall be updated.

Note: For the purpose of implementing Section 5.1, sampling locations will be modified, to reflect the findings of the land use census as described in ODCM Section 5.1.

#### 6.1.2 Program Deviations

With the REMP not being conducted as specified in Table 6-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by TS 6.9.1.10 and Section 7.1, a description of the reasons for not conducting the program as required and plans for preventing a recurrence.

### 6.1.3 Unavailability of Milk or Broad Leaf Vegetation Samples

With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 6-1, identify locations for obtaining replacement samples and if practical add them to the REMP within 30 days. The locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to TS 6.9.1.11 and Section 7.2, identify the cause of the unavailability of samples and identify and the new locations(s) for obtaining replacement samples in the next Semiannual Effluent and Waste Disposal Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new locations(s).

### 6.1.4 Seasonal Unavailability, Equipment Malfunctions, Safety Concerns

With specimens unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons, every effort will be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule will be documented in the Annual Radiological Environmental Operating report pursuant to TS 6.9.1.10 and Section 7.1.

### 6.1.5 Sample Analysis

REMP samples shall be analyzed pursuant to the requirements of Table 6-1 and the detection capabilities required by Table 6-3. Cumulative potential dose contributions for the current calendar year from radionuclides detected in environmental samples shall be determined in accordance with the methodology and parameters in this ODCM.

## 6.2 Reporting Levels

### 6.2.1 General

The reporting levels are based on the design objective doses of 10 CFR 50, Appendix I (i.e., levels of radioactive material in the sampling media corresponding to potential annual doses of 3 mrem, total body or 10 mrem, maximum organ from liquid pathways; or 5 mrem, total body, or 15 mrem, maximum organ for gaseous effluent pathways - the annual limits of Sections 2.4.1, 3.7.1 and 3.8.1). These potential doses are modeled on the maximum exposure or consumption rates of NRC Regulatory Guide 1.109.

The evaluation of potential doses should be based solely on radioactive material resulting from plant operation.



### 6.2.2 Exceedance of Reporting Levels

With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 6-4 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Section 7.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to MEMBER OF THE PUBLIC is less than the calendar year limits of Sections 2.4.1, 3.7.1 and 3.8.1. When more than one of the radionuclides in Table 6-3 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{Concentration (2)}}{\text{reporting level (2)}} + \dots \geq 1.0$$

When radionuclides other than those in Table 6-4 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Sections 2.4.1, 3.7.1 and 3.8.1. The method described in Section 4.2.3 may be used for assessing the potential dose and required reporting for radionuclides other than those listed in Table 6-4.

A special report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

### 6.3 Interlaboratory Comparison Program

Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission. The requirement for participating in an approved Interlaboratory Comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

A summary of the results obtained as part of the required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to TS 6.9.1.10 and Section 7.1. With analyses not being performed as required, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to TS 6.9.1.10 and Section 7.1.

TABLE 6-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations<sup>a</sup></u>	<u>Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
1. DIRECT RADIATION <sup>b</sup>	<p>27 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows:</p> <p>an inner ring of stations, generally one in each meteorological sector in the general area of the SITE BOUNDARY;</p> <p>an outer ring of stations, one in each meteorological sector in the 6- to 8- km range from the site, excluding the sectors over Lake Erie;</p> <p>the balance of the stations to be placed in special interest areas such as populations centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.</p>	Quarterly	Gamma dose quarterly

TABLE 6-1 (Continued)

## RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations*</u>	<u>Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
2. AIRBORNE			
Radioiodine and Particulates	<p>Samples from 5 locations, placed as follows:</p> <p>3 samples from close to the SITE BOUNDARY, in different sectors, generally from areas of higher calculated annual average groundlevel D/Q.</p> <p>1 sample from the vicinity of a nearby community, generally in the area of higher calculated annual average groundlevel D/Q.</p> <p>1 sample from a control location, 15-30 km from the site.</p>	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	<p>Radioiodine Cannister: I-131 analysis weekly.</p> <p>Particulate Sampler: Gross beta radioactivity analysis following filter change;<sup>c</sup> Gamma isotopic analysis of composite (by location) quarterly.</p>
3. WATERBORNE			
a. Surface (untreated water)	2 samples	Weekly composite sample (Indicator location should be a composite)	Tritium and gamma isotopic <sup>d</sup> analysis of composite sample monthly.
b. Ground	Sample from one source only if likely to be affected*	Quarterly	Gamma Isotopic <sup>d</sup> and tritium analysis quarterly.

TABLE 6-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations<sup>a</sup></u>	<u>Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
c. Drinking (Treated water)	1 sample from the nearest source.  1 sample from a control location.	Weekly composite sample.	Gross beta on monthly composite. Tritium and gamma isotopic analysis on quarterly composite. I-131 analysis on each composite when the dose calculated for the composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.
d. Sediment from Shoreline	1 sample from area with existing or potential recreational value.	Semiannually	Gamma isotopic analysis <sup>d</sup> semiannually.
4. INGESTION			
a. Milk	If available, samples from animals up to 2 locations within 8 km distance having the highest dose potential.  1 sample from milking animals at a control location 15-30 km distant and generally in a less prevalent wind direction.	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic <sup>d</sup> and I-131 analysis semimonthly when animals are on pasture; monthly at other times.



TABLE 6-1 (continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations<sup>a</sup></u>	<u>Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
b. Fish	1 sample each of 2 commercially and/or recreationally important species in vicinity of site.  1 sample of same species in areas not influenced by plant discharge.	1 sample in season.	Gamma isotopic analysis <sup>d</sup> on edible portions.
c. Food Products (Broad leaf vegetation)	Samples of up to 3 different kinds of broad leaf vegetation growth in two different offsite locations of higher predicted annual average ground-level D/Q if milk sampling is not performed.  1 sample of each of the similar broad leaf vegetations grown 15-30 km distant in a less prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic <sup>d</sup> and I-131 analysis.

TABLE 6-1 (Continued)

TABLE NOTATION

<sup>a</sup> Specific parameters of distance and direction sector from the centerline of one reactor, additional description where pertinent are provided for each and every sample location in Table 6-2. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.11, and Section 7.2 identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Effluent and Waste Disposal Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

<sup>b</sup> One or more instruments, such as a pressurized ion changer, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

<sup>c</sup> Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

<sup>d</sup> Gamma isotopic analysis means the identification and quantification of gamma emitting radionuclides that may be attributable to the effluents from the facility.

<sup>e</sup> Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

Table 6-2  
Sampling Locations

Location	Appendix C Page Reference	Type of Location*	Location Description
T-1	C-3	I	Site boundary, 0.6 mile ENE of Station.
T-2	C-4	I	Site boundary, 0.9 mile E of Station.
T-3	C-5	I	Site boundary, 1.4 miles ESE of Station near mouth of Toussaint River.
T-4	C-6	I	Site boundary, 0.8 mile S of Station.
T-5	C-7	I	Main entrance to site, 0.7 mile W of Station.
T-6	C-8	I	Site boundary, 0.5 mile NNE of Station.
T-7A	C-9	I	Sand Beach main entrance, 0.9 mile NW of Station.
T-7B	C-9	I	Sand Beach residence, 0.8 mile NNW of Station.
T-8	C-10	I	Farm, 2.7 miles WSW of Station.
T-9	C-11	C	Oak Harbor substation, 6.8 miles SW of Station.
T-10	C-12	I	Site boundary, 0.5 mile SSW of Station.
T-11	C-13	C	Port Clinton Water Treatment plant, 9.5 miles SE of Station.
T-12	C-14	C	Toledo Water Treatment Plant, 23.5 miles WNW of Station. Water samples are collected 11.3 miles NW of site.
T-25	C-15	I	Farm, 3.7 miles S of Station.

\* I = Indicator locations; C = Control locations.

Table 6-2 (continued)

## Sampling Locations

Location	Appendix C Page Reference	Type of Location*	Location Description
T-27	C-16	C	Crane Creek State Park, 5.3 miles WNW of Station.
T-28	C-17	I	Davis-Besse Water Treatment Plant, onsite.
T-33	C-18	I	Lake Erie within a 5 mile radius.
T-35	C-19	C	Lake Erie greater than a 10 mile radius.
T-37	C-20	C	Farm, 13 miles SW of Station.
T-40	C-21	I	Site boundary, 0.7 mile SE of Station.
T-41	C-22	I	Site Boundary, 0.6 mile SSE of Station.
T-42	C-23	I	Site boundary, 0.8 mile SW of Station.
T-44	C-24	I	Site boundary, 0.5 mile WSW of Station.
T-46	C-25	I	Site boundary, 0.5 mile NW of Station.
T-47	C-26	I	Site boundary, 0.5 mile N of Station.
T-48	C-27	I	Site boundary, 0.5 mile NE of Station.
T-50	C-28	I	Erie Industrial Park Water Treatment Plant, 4.5 mile SE of Station.

\* I = Indicator locations; C = Control locations.



Table 6-2 (continued)

## Sampling Locations

Location	Appendix C Page Reference	Type of Location*	Location Description
T-52	C-29	I	Farm, 3.7 miles S of Station.
T-54	C-30	I	Farm, 4.8 miles SW of Station.
T-55	C-31	I	Farm, 5.0 miles W. of Station.
T-57	C-32	C	Farm, 22 miles SSE of Station.
T-67	C-33	I	Site boundary, 0.3 mile NNW of Station.
T-68	C-34	I	Site Boundary, 0.5 miles WNW of station
T-91	C-35	I	Siren Post No. 1108, 2.5 miles SSE of Station.
T-112	C-36	I	State Route 2 and Thompson Road, 1.5 miles SSW of Station.
T-151	C-51	I	State Route 2 and Humphrey Road, 1.8 miles WNW of Station.

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\* I = Incidator locations; C = Control locations.

Table 6-3

LOWER LIMITS OF DETECTION (LLD)<sup>a</sup>

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m <sup>3</sup> )	Fish (pCi/kg. wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
Gross Beta	4 <sup>b</sup>	1.0E-02				
<sup>3</sup> H	2000 <sup>c*</sup>					
<sup>54</sup> Mn	15		130			
<sup>59</sup> FE	30		260			
58, <sup>60</sup> Co	15		130			
<sup>65</sup> ZN	30		260			
<sup>95</sup> ZR	15					
<sup>131</sup> I	1 <sup>d</sup>	7.0E-02		1	60	
<sup>134</sup> , <sup>137</sup> Cs	15(10 <sup>b</sup> ), 18	6.0E-02	130	15	60	150
<sup>140</sup> Ba	15			15		

NOTE: 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.

\* If no drinking water pathway exists, a value of 3000 pCi/L may be used.

TABLE 6-3 (Continued)

TABLE NOTATION

- A. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

where

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

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

E is the counting efficiency (as counts per transformation).

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

2.22 is the number of transformations per minute per picocurie.

Y is the fractional radiochemical yield (when applicable).

$\lambda$  is the radioactive decay constant for the particular radionuclide.

$\Delta t$  is the elapsed time between end of the sample collection period and time of counting.

Typical values of E, V, Y and  $\Delta t$  should be used in the calculations.

TABLE 6-4

## REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

## Reporting Levels

Analysis	Water ( $\mu\text{Ci/l}$ )	Airborne Particulate or Gases ( $\mu\text{Ci/m}^3$ )	Fish ( $\mu\text{Ci/kg, wet}$ )	Milk ( $\mu\text{Ci/l}$ )	Vegetables ( $\mu\text{Ci/kg, wet}$ )
H-3	2.0E+04				
Mn-54	1.0E+03		3.0E+04		
Fe-59	4.0E+02		1.0E+04		
Co-58	1.0E+03		3.0E+04		
Co-60	3.0E+02		1.0E+04		
Zn-65	3.0E+02		2.0E+04		
Zr-Nb-95	4.0E+02				
I-131	2.0E+00	9.0E-01		3.0E+00	1.0E+02
CS-134	3.0E+01	1.0E+01	1.0E+03	6.0E+01	1.0E+03
CS-137	5.0E+01	2.0E+01	2.0E+03	7.0E+03	2.0E+03
Ba-La-140	2.0E+02			3.0E+02	

\* For drinking water samples, this is the 40 CFR 141 value. If no drinking water pathway exists, a value of 30,000  $\mu\text{Ci/liter}$  may be used.



## 7.0 ADMINISTRATIVE CONTROLS

### 7.1 Annual Radiological Environmental Operating Report

Routine Radiological Environmental Operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with the preoperational studies, with operational controls, as appropriate, and with previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses as required in Section 5.1.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all radiation measurements taken during the period pursuant to the locations specified in Sections 6.1 and Appendix C of this ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Section 6.3; and discussions of all analyses in which the LLD required by Table 6-3 was not achievable.

### 7.2 Semiannual Effluent and Waste Disposal Report

Routine Effluent and Waste Disposal Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Effluent and Waste Disposal Reports (Semiannual Reports) shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY during the reporting period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location, shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in this ODCM.

The Semiannual report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation."

The Semiannual report shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

- a. Container volume,
- b. Total curie quantity (specify whether determined by measurement or estimate),
- c. Principal radionuclides (specify whether determined by measurement or estimate),
- d. Source of waste and processing employed (e.g., dewatered spent resin, compressed dry waste, evaporator bottoms).
- e. Type of container (e.g., Type A, Type 3, Large Quantity), and
- f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the ODCM, as well as a listing of new locations for dose calculations and pursuant to Section 5.1.

### 7.3 Special Reports

Special Reports shall be submitted to the U. S. Nuclear Regulatory Commission (NRC) in accordance with 10 CFR 50.4 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:

- a. Dose or dose commitment exceedences to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (Section 2.4.1).
- b. The discharge of radioactive liquid waste without treatment and in excess of the limits in Section 2.5.
- c. The calculated air dose from radioactive gases exceeding the limits in Section 3.7.1.
- d. The calculated dose from the release of iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding the limits of Section 3.8.1.
- e. The discharge of radioactive gaseous waste without treatment and in excess of the limits in Section 3.9.
- f. The calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding the limits of Section 4.2.
- g. The level of radioactivity as the result of plant effluents in an environmental sampling medium exceeding the reporting levels of Table 6-4 (Section 6.2.2).

#### 7.4 Major Changes to Radioactive Liquid and Gaseous Waste Treatment Systems

Licensee initiated major changes to the radioactive waste systems (liquid and gaseous):

1. Shall be reported to the Commission in the update to the Safety Analysis Report. The discussion of each change shall contain:
  - a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;
  - b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
  - c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;
  - d. An evaluation on the change, which shows the predicted releases of radioactive materials in liquid or gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;
  - e. An evaluation of the change, which shows the expected maximum exposures to individuals in the UNRESTRICTED AREA and the general population that differ from those previously estimated in the license application and amendments thereto;



- f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents, to the actual releases for the period prior to when the changes are to be made;
  - g. An estimate of the exposure to plant operating personnel as a result of the change; and
  - h. Documentation of the fact that the change was reviewed and found acceptable by the Station Review Board.
2. Shall become effective upon review and acceptance by the Station Review Board.

#### 7.5 Definitions

- 7.5.1 BATCH RELEASE - The discharge of liquid wastes of a discrete volume.
- 7.5.2 CHANNEL CALIBRATION - A channel calibration shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameters which the channel monitors. The channel calibration shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Channel calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.
- 7.5.3 CHANNEL CHECK - A channel check shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels monitoring the same parameter.
- 7.5.4 CHANNEL FUNCTIONAL TEST - A channel functional test shall be:
- a. Analog Channels - The injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
  - b. Bistable Channels - The injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.
- 7.5.5 COMPOSITE SAMPLE - A sample in which the method of sampling employed results in a specimen which is representative of the liquids released.
- 7.5.6 GASEOUS RADWASTE TREATMENT SYSTEM - A system that is designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off gases and providing for decay for the purpose of reducing the total radioactivity prior to release to the environment.

- 7.5.7 LOWER LIMIT OF DETECTION (LLD) - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability, with 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

where

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

S<sub>b</sub> is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute);

E is the counting efficiency (as counts per transformations);

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

2.22 is the number of transformations per minute per picocurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide;

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

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

- 7.5.8 MEMBER OF THE PUBLIC - Member(s) of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreations, occupational, or other purposes not associated with the plant.

- 7.5.9 OPERABLE - OPERABILITY - A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary device to perform its function(s), are also capable of performing their related support functions(s).



- 7.5.10 PURGE-PURGING - the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.
- 7.5.11 SITE BOUNDARY - The site boundary shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.
- 7.5.12 SOURCE CHECK - A source check shall be the observation of channel upscale response when the channel sensor is exposed to a radioactive source.
- 7.5.13 UNRESTRICTED AREA - An unrestricted area shall be any area at or beyond the SITE boundary, access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation or radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes. The definition of unrestricted area used in implementing the Radiological Effluent Technical Specifications has been expended over that in 10 CFR 100.3(a), but the unrestricted area does not include areas over water bodies. The concept of unrestricted areas, established at or beyond the SITE BOUNDARY, is utilized in the Technical Specifications and the ODCM to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.
- 7.5.14 VENTILATION EXHAUST TREATMENT SYSTEM - a ventilation exhaust treatment system is a system that is designed and installed to reduce radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through HEPA filters for the purpose of removing particulates from the gaseous exhaust stream prior to release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.
- 7.5.15 VENT-VENTING - the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

APPENDIX A  
Technical Basis for Simplified Dose Calculations  
Liquid Effluent Releases

APPENDIX A  
Technical Basis for Simplified Dose Calculations  
Liquid Effluent Releases

Overview

To simplify the dose calculation process, it is conservative to identify a controlling, dose-significant radionuclide and to use its dose conversion factor in the dose calculations. Using the total release (i.e., the cumulative activity of all radionuclides) and this single dose conversion factor as inputs to a one-step dose assessment yields a dose calculation method which is both simple and conservative.

Cs-134 is the controlling nuclide for the total body dose. It has the highest total body dose conversion factor for all the radionuclides listed in Table 2-5. Therefore, the use of its dose conversion factor in the simplified dose assessment method for evaluating the total body dose is demonstrably conservative.

The selection of the maximum organ dose conversion factor for use in the simplified calculation requires consideration of the prevalence of the radionuclides in the effluents. An examination of the Table 2-5 factor will show that the Nb-95 dose factor for the GI-LLI represents the highest value ( $1.51\text{E}+06$  mrem/hr per  $\mu\text{Ci}/\text{ml}$ ); and the P-32 bone factor ( $1.39\text{E}+06$ ) is similarly high. However, neither of these two radionuclides are of significance in the Davis-Besse effluents. Nb-95 is not typically measured in the liquid effluents and P-32 analyses are not even performed. (NRC has categorically determined that P-32 is not a significant radionuclide in liquid effluents from nuclear power plants and does not require the special radiochemical analyses needed for identification and quantification.) The next highest dose conversion factor is for Cs-134, liver, with a value of  $7.11\text{E}+05$  mrem/hr per  $\mu\text{Ci}/\text{ml}$ . And, Cs-134 is a prevalent radionuclide in the liquid effluents from Davis-Besse. Therefore, it is recommended that the Cs-134 liver dose conversion factor be used for the simplified maximum organ dose assessment.

### Simplified Method

For evaluating compliance with the dose limits of Section 2.41, the following simplified equations may be used:

#### Total Body

$$D_{tb} = \frac{1.67E-02 * VOL}{DF * Z} * A_{(Cs-134, tb)} * EC_1 \quad (A-1)$$

where:

$D_{tb}$  = dose to the total body (mrem)

VOL = volume of liquid effluents released (gal)

DF = average Collection Box release flow (gal/min)

Z = 10, near field dilution

$A_{(Cs-134, tb)}$  =  $5.81E+05$  mrem/hr per  $\mu$ Ci/ml, the total body ingestion dose factor for Cs-134

$EC_1$  = total concentration of all radionuclides ( $\mu$ Ci/ml)

$1.67E-02$  = 1 hr/60 min

Substituting the values for Z and the Cs-134 total body dose conversion factor, the equation simplifies to:

$$D_{tb} = \frac{9.70 E+02 * VOL}{DF} * EC_1 \quad (A-2)$$

#### Maximum Organ

$$D_{max} = \frac{1.67E-02 * VOL}{DF * Z} * A_{(Cs-134, liver)} * C_1 \quad (A-3)$$

where:

$D_{max}$  = maximum organ dose (mrem)

$A_{(Cs-134, liver)}$  =  $7.11E+05$  mrem/hr per  $\mu$ Ci/ml, the liver ingestion dose factor for Cs-134



Substituting the values for Z and the Cs-134 liver dose conversion factor, the equation simplifies to:

$$D_{max} = \frac{1.19 \text{ E}+03 * \text{VOL}}{\text{DF}} * \text{EC}_1 \quad (\text{A-4})$$

Tritium should not be included in the simplified analysis dose assessment for liquid releases. The potential dose resulting from normal reactor releases of H-3 is relatively negligible. But, its relatively higher abundance would yield resulting simplified doses that would be overly conservative and unrealistic. Excluding tritium has essentially no impact on the conservative use of this recommended simplified method. Furthermore, the release of tritium is a function of operating history and is essentially unrelated to radwaste system operations.

APPENDIX B  
Technical Basis for Effective Dose Factors  
Gaseous Radwaste Effluents

# APPENDIX B Technical Basis for Effective Dose Factors Gaseous Radwaste Effluents

## Overview

Dose evaluations for releases of gaseous radioactive effluents may be simplified by the use of an effective dose factor rather than radionuclide-specific dose factors. These effective dose factors are applied to the total radioactive release to approximate the various doses in the environment; i.e., the total body, gamma-air, and beta-air doses. The effective dose factors are based on the typical radionuclide distribution in the gaseous radioactive effluents. This approach reduces the analyses to a single multiplication ( $K_{eff}$ ,  $M_{eff}$ , or  $N_{eff}$ ) times the quantity of radioactive gases released, rather than individual analyses for each radionuclide and summing the results to determine the dose. Yet the approach provides a reasonable estimate of the actual doses since under normal operating conditions there is expected to be minor variations in the radionuclide distribution.

## Determination of Effective Dose Factors

Effective dose transfer factors are calculated by the following equations:

$$K_{eff} = \sum (K_i * f_i) \quad (B-1)$$

where:

- $K_{eff}$  = the effective total body dose factor due to gamma emissions from all noble gases released (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ , effective)
- $K_i$  = the total body dose factor due to gamma emissions from each noble gas radionuclide (i) released (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ , from Table 3-6)
- $f_i$  = the fractional abundance of noble gas radionuclide (i) relative to the total noble gas activity.

$$(L + 1.1 M)_{eff} = \sum ((L_i + 1.1 M_i) * f_i) \quad (B-2)$$

where:

- $(L+1.1M)_{eff}$  = the effective skin dose factor due to beta and gamma emissions from all noble gases released (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ , effective)
- $(L_i+1.1M_i)$  = the skin dose factor due to beta and gamma emissions from each noble gas radionuclide (i) released (mrem/yr per  $\mu\text{Ci}/\text{m}^3$ , from Table 3-6)

$$M_{eff} = \sum (M_i * f_i) \quad (B-3)$$

where:

- $M_{eff}$  = the effective air dose factor due to gamma emissions from all noble gases released (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ , effective)  
 $M_i$  = the air dose factor due to gamma emissions from each noble gas radionuclide (i) released (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ , from Table 3-6)

$$N_{eff} = \sum (N_i * f_i) \quad (B-4)$$

where:

- $N_{eff}$  = the effective air dose factor due to beta emissions from all noble gases released (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ , effective)  
 $N_i$  = the air dose factor due to beta emissions from each noble gas radionuclide (i) released (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ , from Table 3-6)

Normally, past radioactive effluent data would be used for the determination of the effective dose factors. However, the releases of noble gases from Davis-Besse have been exceedingly insignificant. Therefore, in order to ensure overall conservatism in the modeling, the USAR estimate of radionuclide concentrations at the site boundary (summarized in Table B-1) has been used as the initial typical distribution. The effective dose factors derived from this distribution are presented in Table B-2.

#### Application

To provide an additional degree of conservatism, a factor of 2.0 is introduced into the dose calculation when the effective dose factor is used. This conservatism provides additional assurance that the evaluation of doses by the use of a single effective dose factor will not significantly underestimate any actual doses in the environment.

For evaluating compliance with the dose limits of Technical Specification 3.11.2.2 the following simplified equations may be used:

$$D\gamma = 2.0 * 3.17\text{E-}08 * X/Q * M_{eff} * EQ_i \quad (B-5)$$

and

$$D\beta = 2.0 * 3.17\text{E-}08 * X/Q * N_{eff} * EQ_i \quad (B-6)$$



where:

- $D_\gamma$  = air dose due to gamma emissions for the cumulative release of all noble gases (mrad)
- $D_\beta$  = air dose due to beta emissions for the cumulative release of all noble gases (mrad)
- $X/Q$  = atmospheric dispersion to the controlling site boundary ( $\text{sec}/\text{m}^3$ )
- $M_{\text{eff}}$  =  $5.7\text{E}+02$ , effective gamma-air dose factor (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ )
- $N_{\text{eff}}$  =  $1.1\text{E}+03$ , effective beta-air dose factor (mrad/yr per  $\mu\text{Ci}/\text{m}^3$ )
- $Q_i$  = cumulative release for all noble gas radionuclides ( $\mu\text{Ci}$ )
- $3.17\text{E}-08$  = conversion factor (yr/sec)
- 2.0 = conservatism factor to account for the variability in the effluent data.

Combining the constants, the dose calculation equations simplify to:

$$D_\gamma = 3.61\text{E}-05 * X/Q * IQ_i \quad (\text{B-7})$$

and

$$D_\beta = 7.20\text{E}-05 * X/Q * IQ_i \quad (\text{B-8})$$

The effective dose factors are used for the purpose of facilitating the timely assessment of radioactive effluent releases, particularly during periods when the computer or ODCM software may be unavailable to perform a detailed dose assessment.

TABLE B-1  
Default Noble Gas Radionuclide Distribution\*  
of Gaseous Effluents

Nuclide	Containment Vessel Purge	Fraction of Total ( $A_i / \sum A_i$ )		Total
		Station Vent	Waste Gas Decay Tank	
Ar-41	0.0003	0.004	0.004	0.003
Kr-85	0.12	0.012	0.034	0.06
Xe-131m	0.02	0.009	0.008	0.017
Xe-133m	0.005	0.011	0.011	0.008
Xe-133	0.86	0.94	0.92	0.83
Xe-135m	---	0.004	0.0034	0.06
Xe-135	0.002	0.02	0.02	0.021
Total	1.0	1.0	1.0	1.0

NOTE:

\* Data adapted from Davis-Besse USAR Section 11.3, Table 11.3-13 and Table 11.3-14. Kr-83m, Kr-85m, Kr-87, Kr-88 and Xe-138 have been excluded because of their negligible fractional abundance (i.e., < 1%).

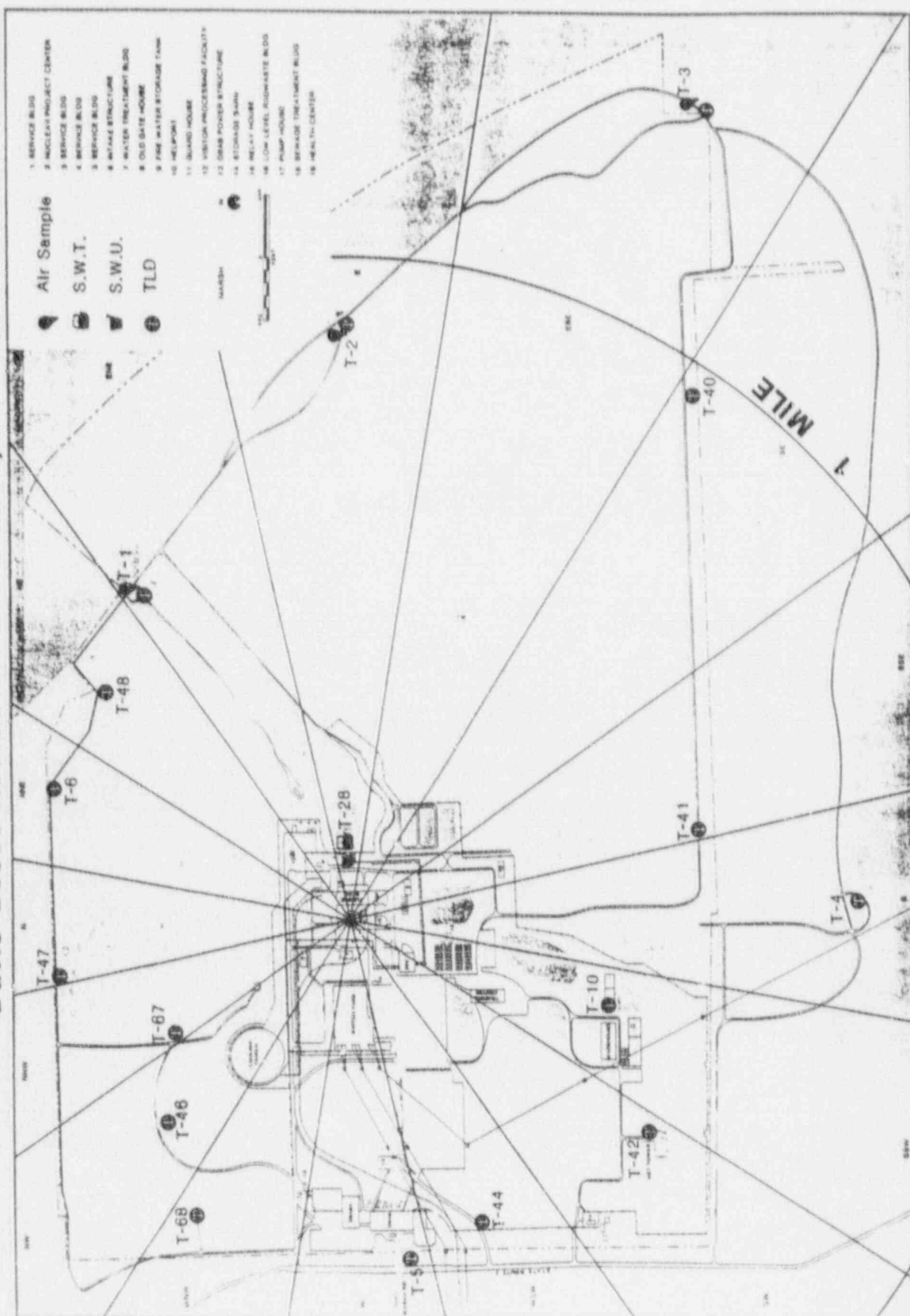
TABLE B-2  
Effective Dose Factors - Noble Gas Effluents

Isotope	Fractional Abundance	Total Body Dose Factor K (mrem/yr per $\mu\text{Ci}/\text{m}^3$ )	Skin Dose Factor (L+1.1M) (mrem/yr per $\mu\text{Ci}/\text{m}^3$ )	Gamma Air Dose Factor M (mrad/yr per $\mu\text{Ci}/\text{m}^3$ )	Beta Air Dose Factor N (mrad/yr per $\mu\text{Ci}/\text{m}^3$ )
Ar-41	0.003	2.65E+01	3.87E+01	2.79E+01	9.84E+00
Kr-85	0.06	9.96E-01	8.15E+01	1.03E+00	1.17E+02
Xe-131m	0.017	1.55E+00	1.10E+01	2.65E+00	1.88E+01
Xe-133m	0.008	2.00E+00	1.08E+01	2.61E+00	1.18E+01
Xe-133	0.83	2.44E+02	5.76E+02	2.93E+02	8.72E+02
Xe-135m	0.06	1.87E+02	2.64E+02	2.02E+02	4.43E+01
Xe-135	0.02	3.62E+01	7.94E+02	4.03E+01	5.16E+01
TOTAL	1.0	4.98E+02	9.89E+02	5.69E+02	1.12E+03

APPENDIX C  
Radiological Environmental Monitoring Program  
Sample Location Maps

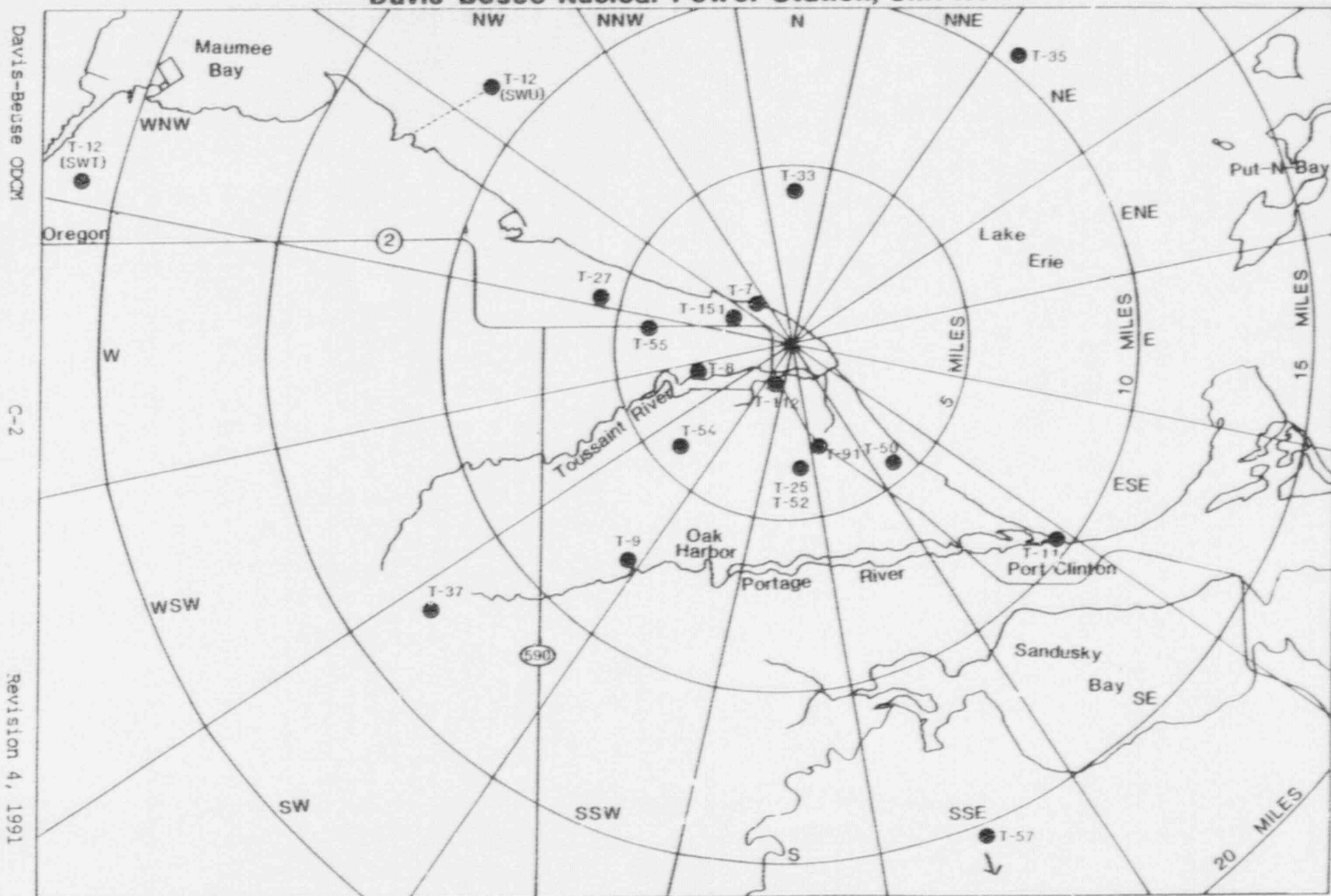


# Sampling locations on the site periphery of the Davis-Besse Nuclear Power Station, Unit No. 1.



ENVIRONMENTAL MONITORING

**Sampling locations (excepting those on the site periphery)  
Davis-Besse Nuclear Power Station, Unit No. 1.**



ENVIRONMENTAL MONITORING

Davis-Besse ODCM

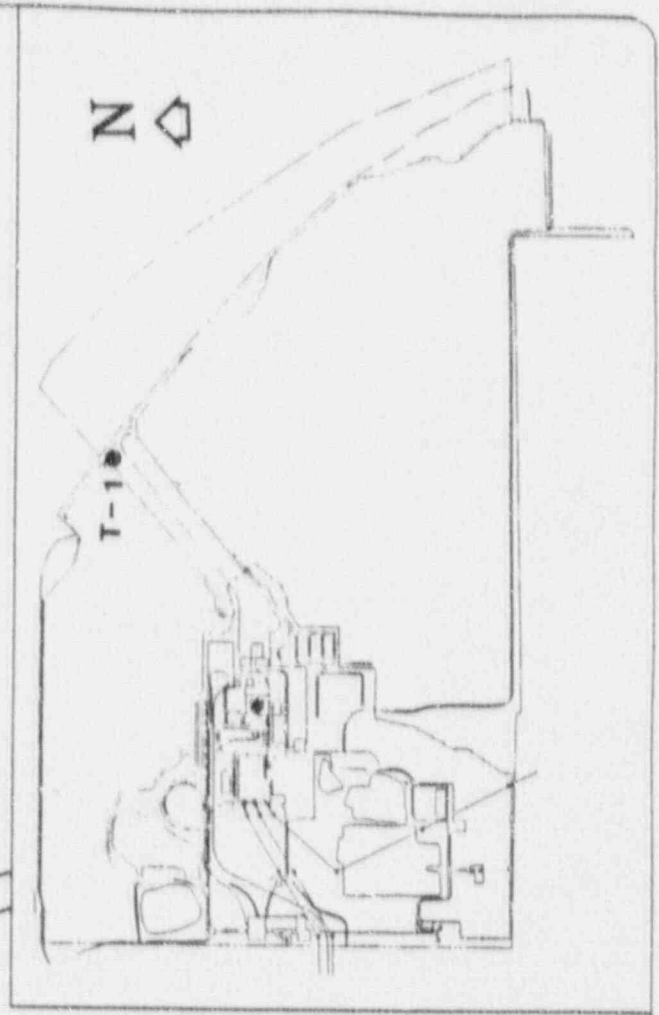
C-2

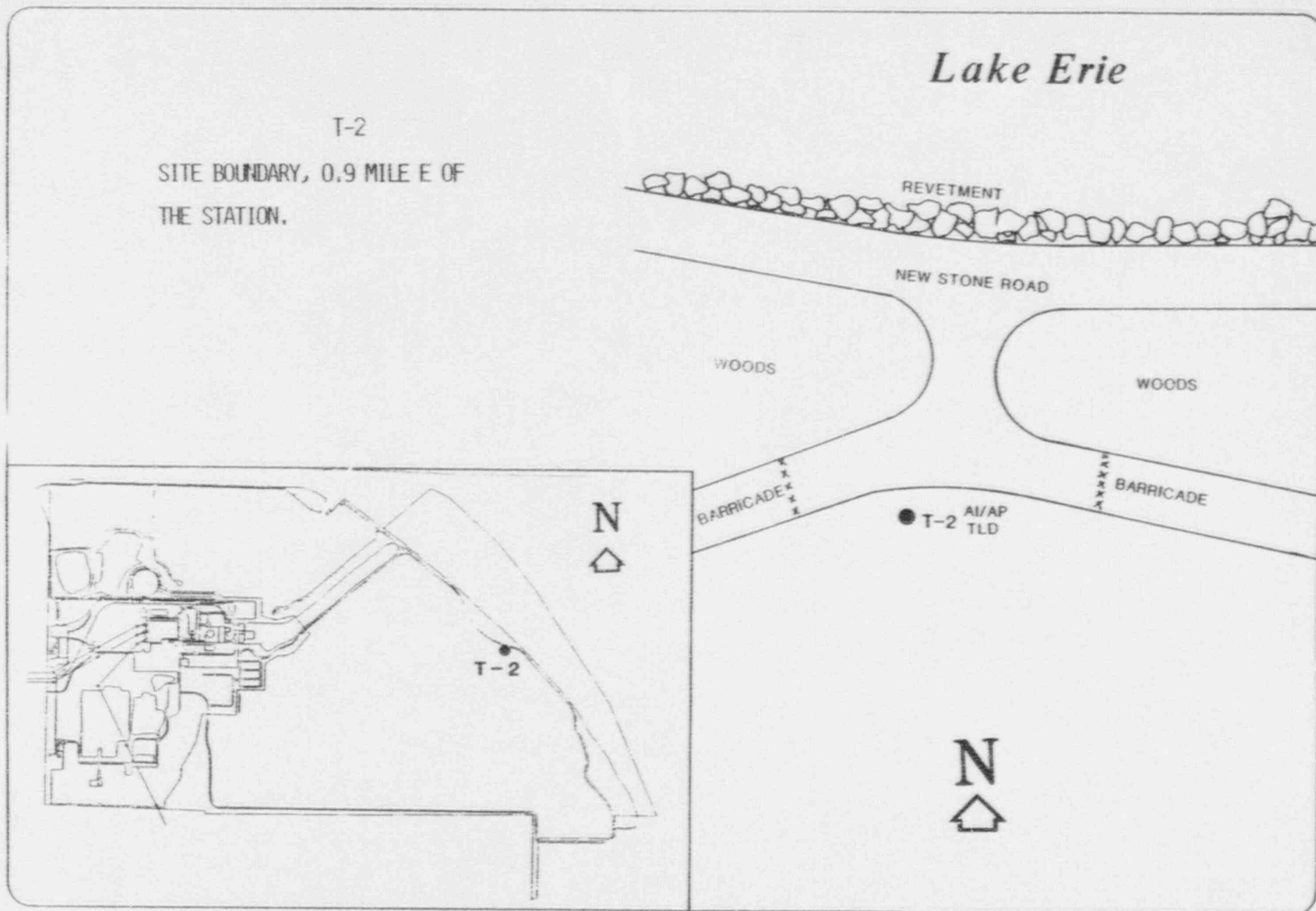
Revision 4, 1991

T-1

SITE BOUNDARY, 0.6 MILE ENE OF  
STATION, NORTH OF INTAKE CANAL.

LAKE ERIE







T-3  
 SITE BOUNDARY, 1.4 MILES ESE OF  
 STATION, NEAR MOUTH OF TOUSSAINT  
 RIVER.

Pool 2

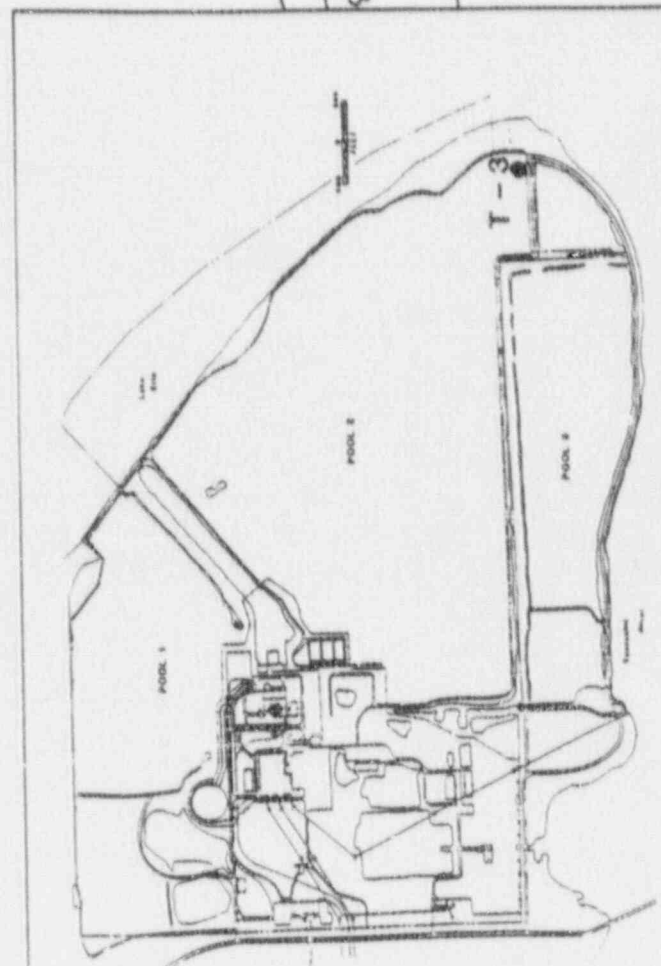
AI/AP  
 TLD  
 T-3

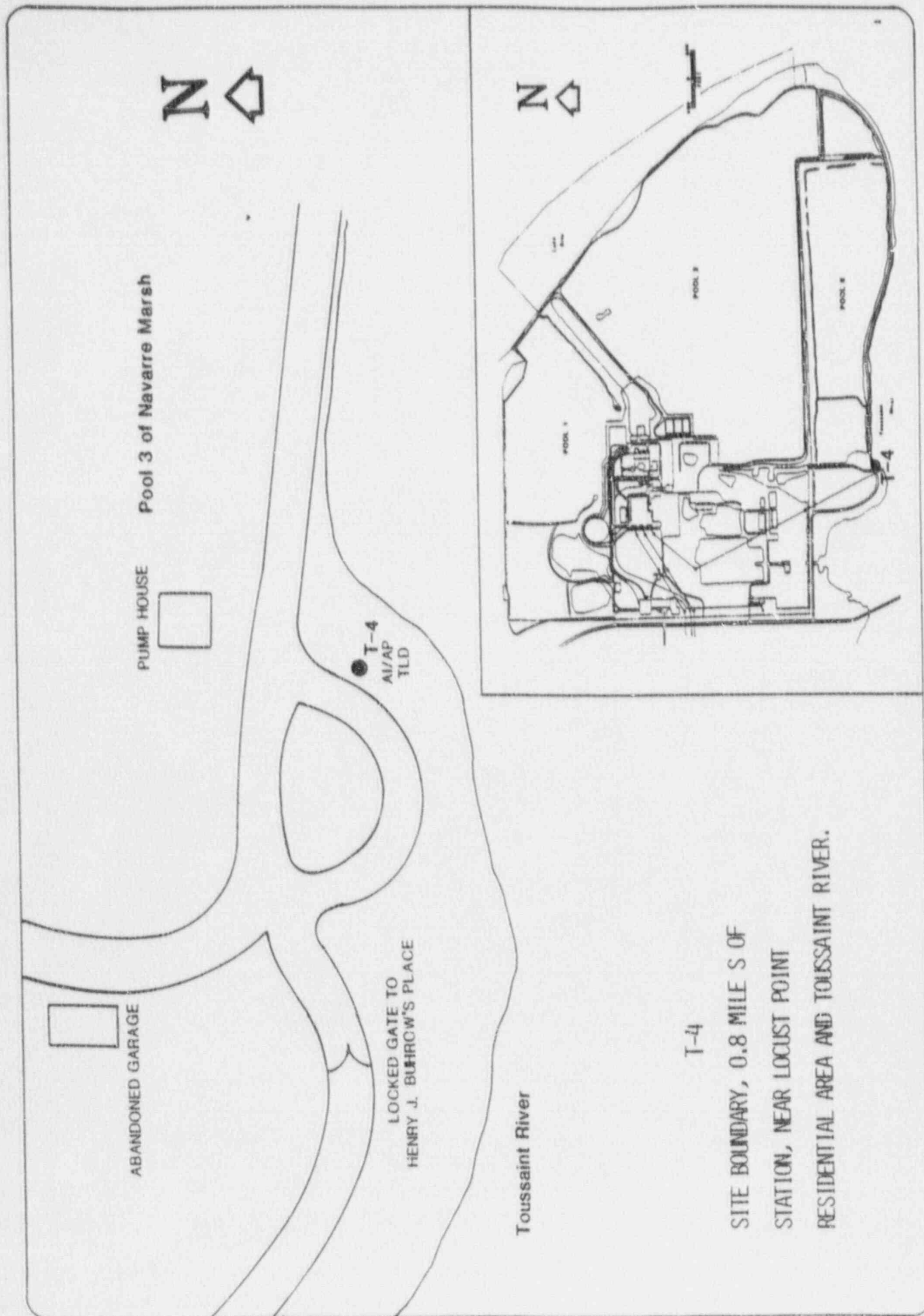
STONE DIKE ROAD

Pool 3

TOUSSAINT  
 RIVER

N







Davis-Besse ODCM

C-7

DUFF WASHA RD.

PARKING LOT

T-5 TLD  
ON LIGHT POLE

DBAB

T-5  
MAIN ENTRANCE TO SITE, 0.50 MILE  
W OF STATION.



ROUTE 2

DUFF WASHA RD.

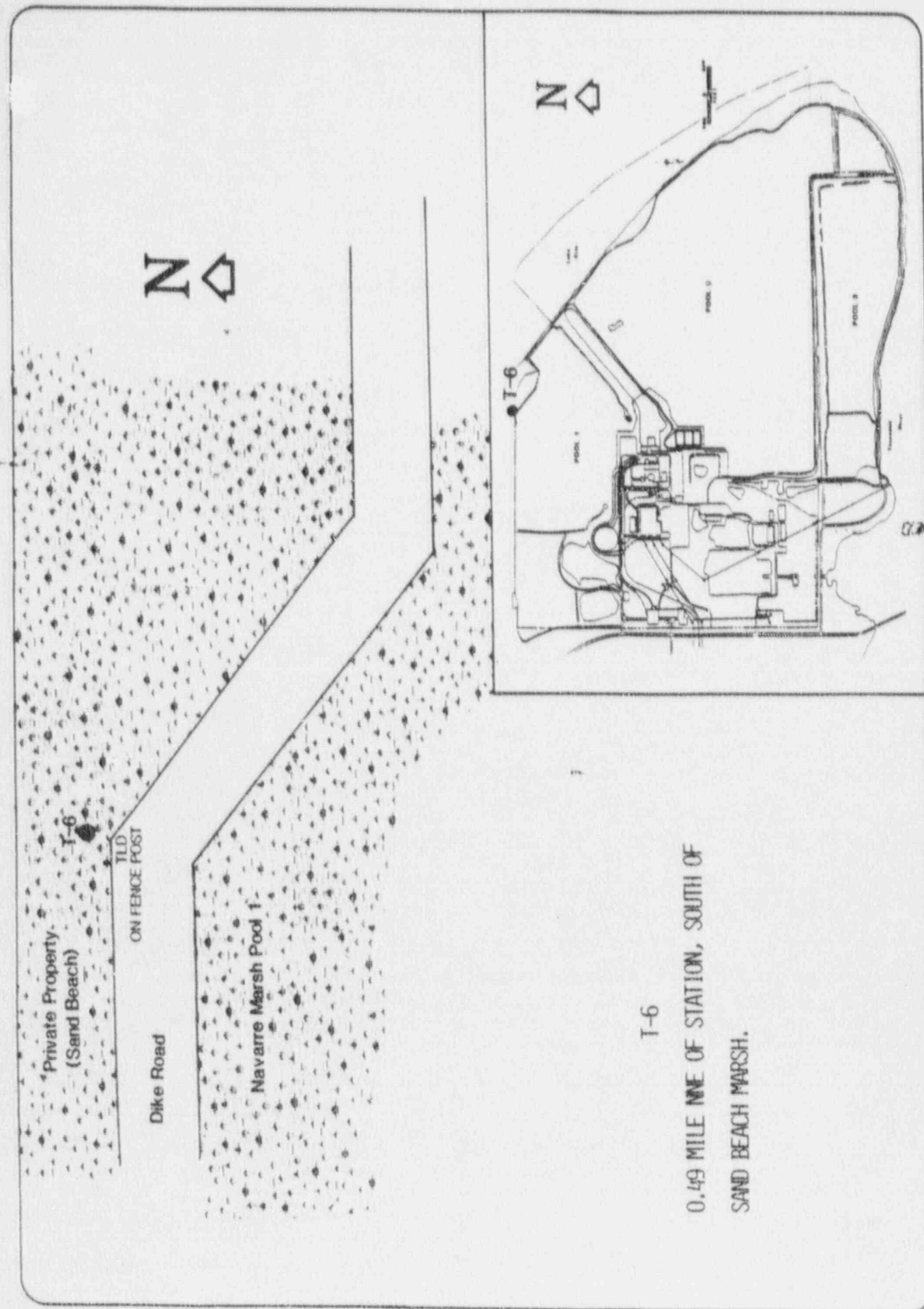
POND

COOLING  
TWR

DBAB

T-5

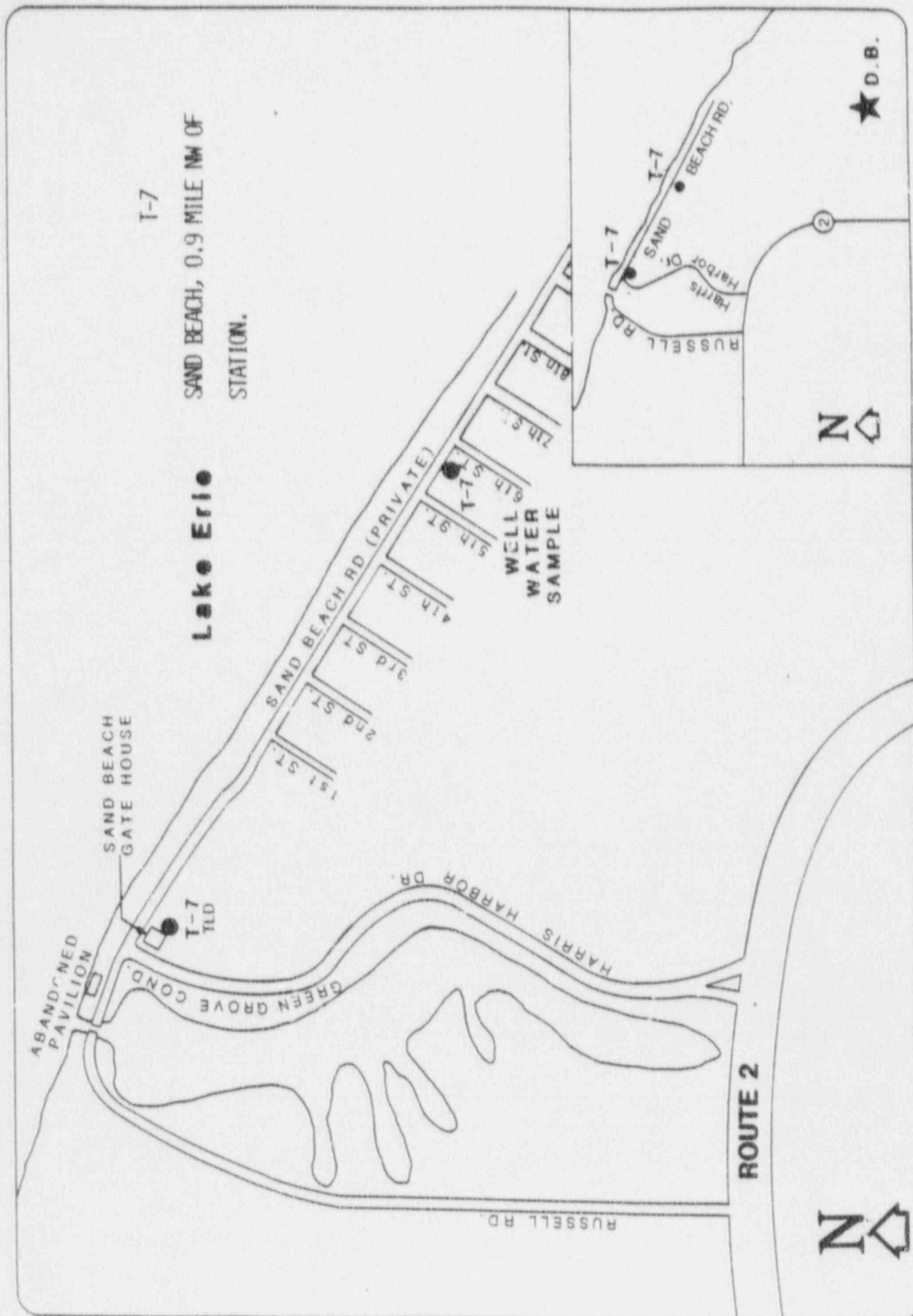
Revision 4, 1991

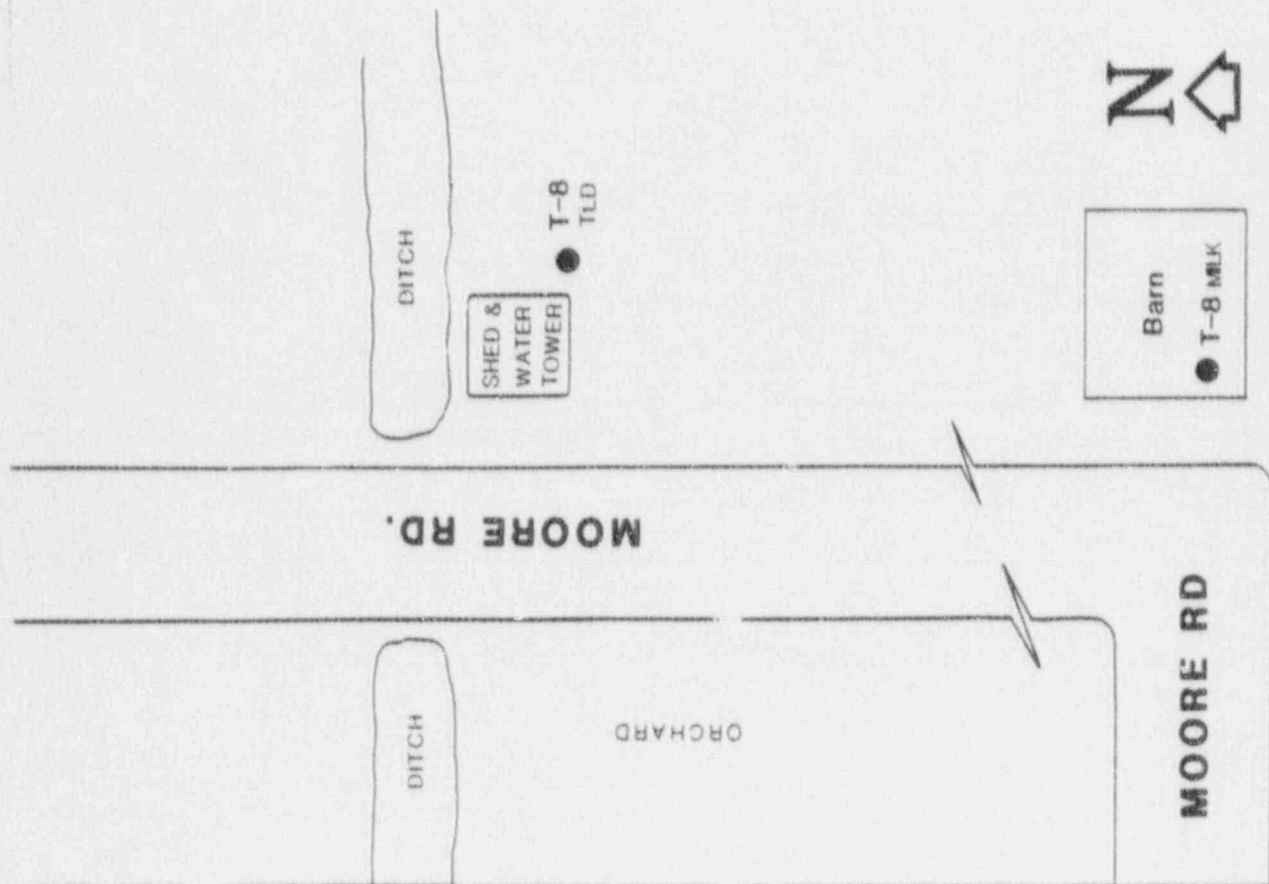


T-6

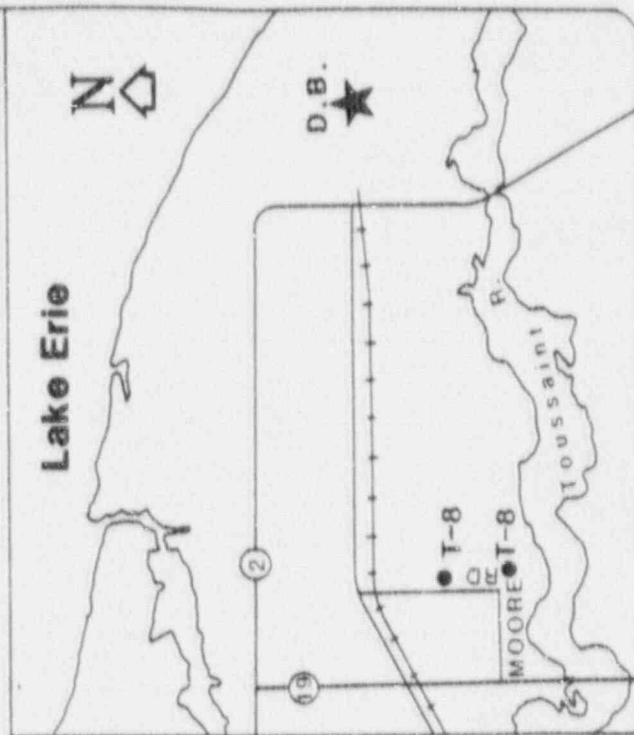
0.49 MILE NINE OF STATION, SOUTH OF  
SAND BEACH MARSH.

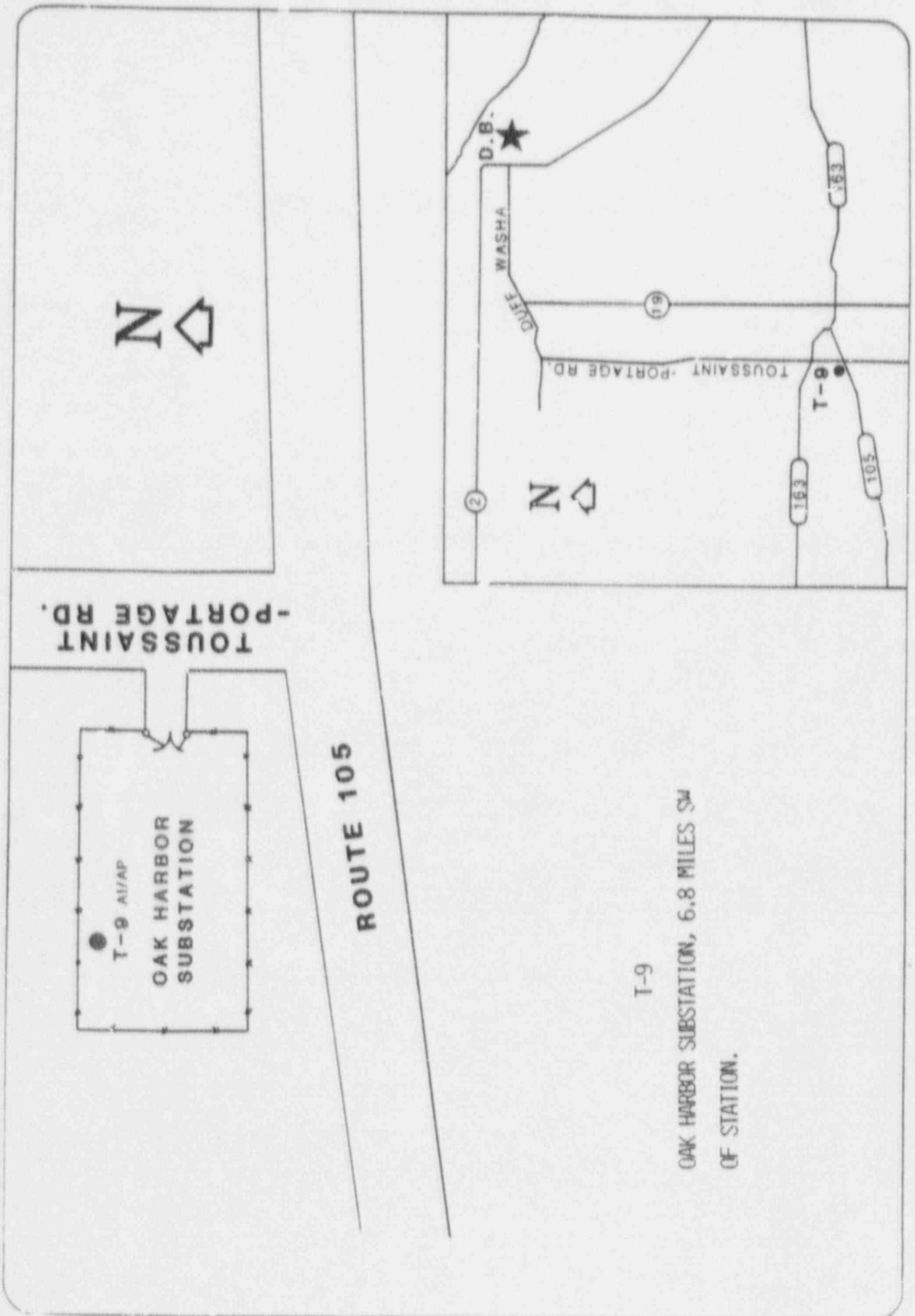




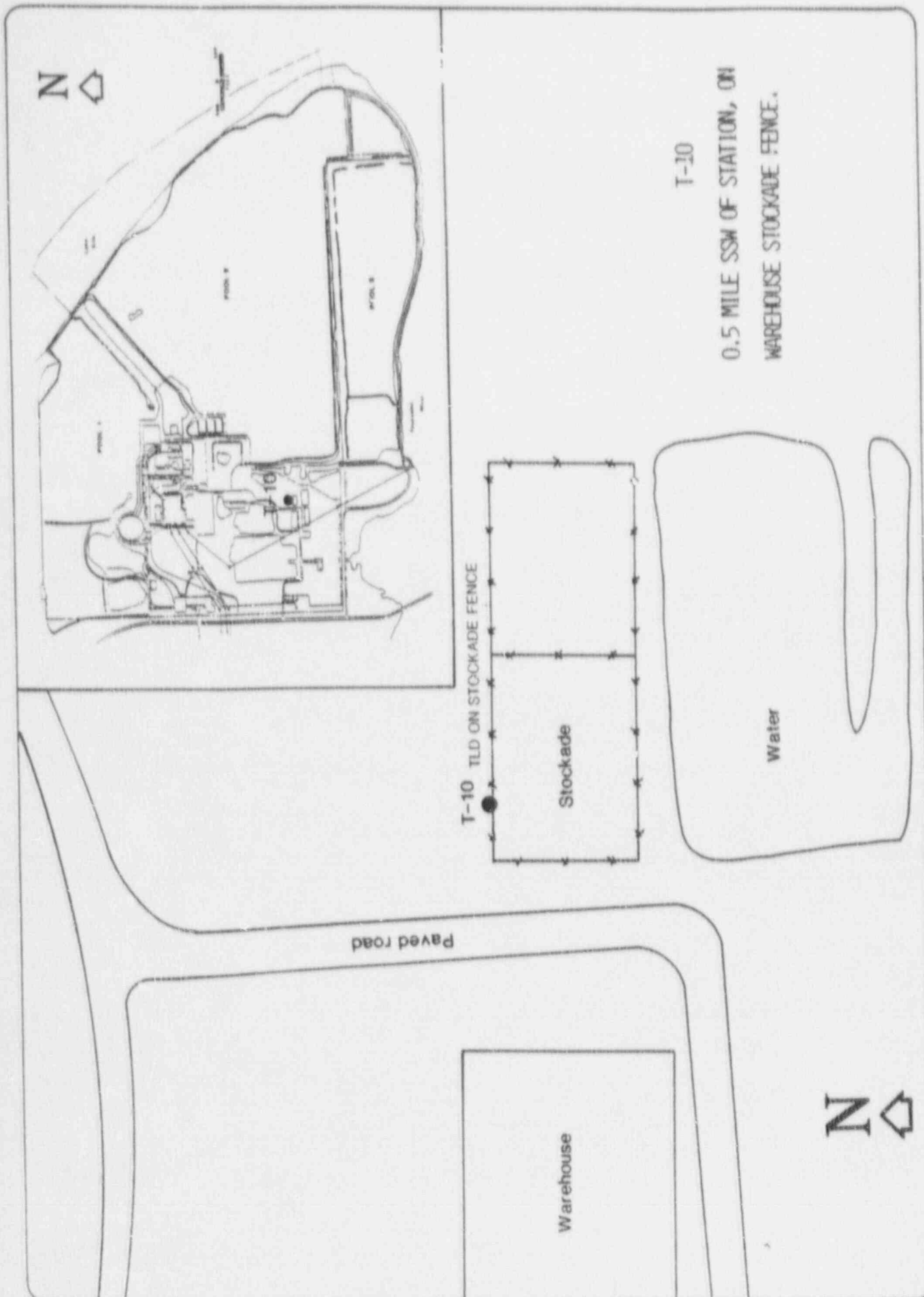


T-8  
EARL MOORE FARM, 2.7 MILES WSW  
OF STATION.



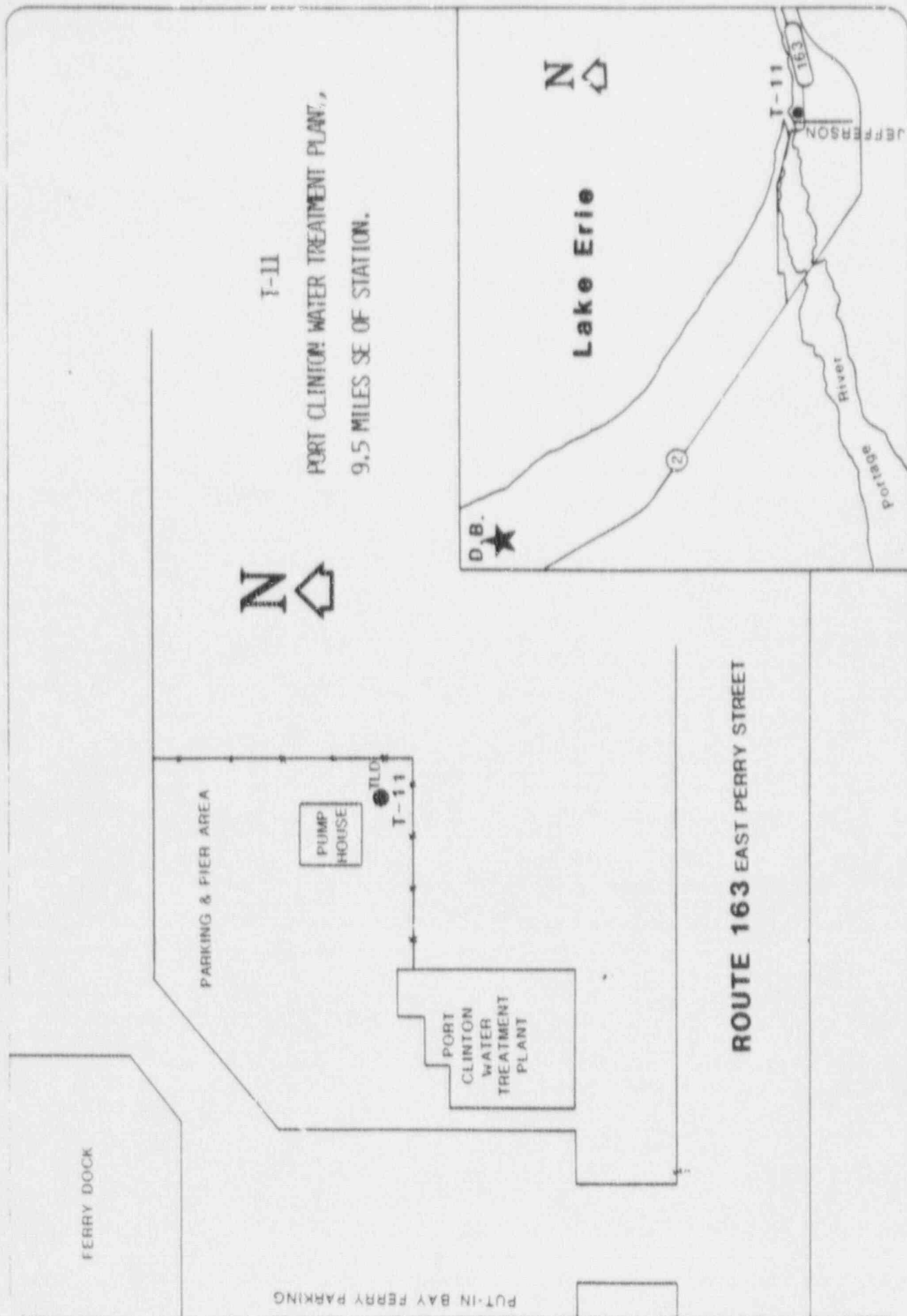


T-9  
OAK HARBOR SUBSTATION, 6.8 MILES SW  
OF STATION.



T-10  
0.5 MILE SSW OF STATION, ON  
WAREHOUSE STOCKADE FENCE.





T-11  
 PORT CLINTON WATER TREATMENT PLANT,  
 9.5 MILES SE OF STATION.

T-12

TOLEDO WATER TREATMENT PLANT, 23.5  
MILES NW OF STATION.

TO COLLINS PARK YORK ST.

T-12

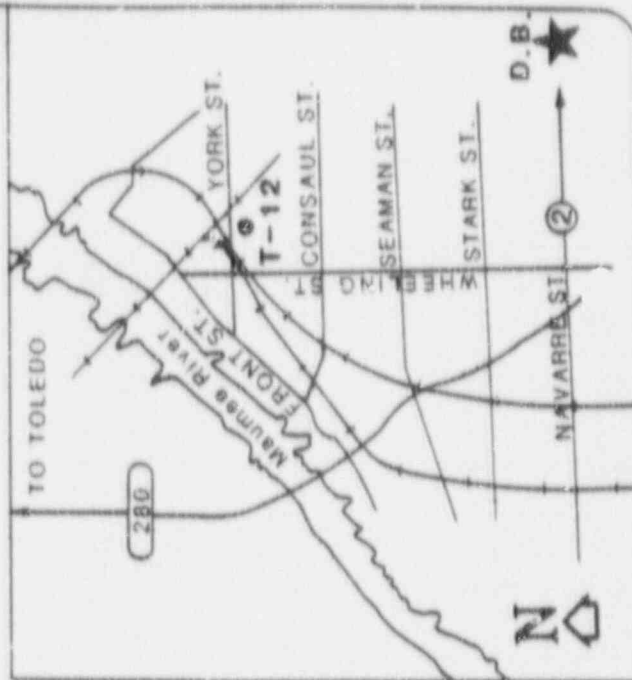
AI/AP  
TLD

SLIDING  
GATE  
#2

N

TOLEDO WATER  
TREATMENT PLANT  
600 COLLINS PARK  
OREGON, OHIO  
WATER SAMPLES

- SWT SAMPLES TAKEN INSIDE PLANT.
- S&P SAMPLES COLLECTED AT INTAKE CRIB.  
11.25 MILES NW OF STATION IN LAKE ERIE.



Davis-Besse ODCM

C-15

Revision 4, 1991

T-52

TLD is mounted on fence post in orchard

**CAMP PERRY WESTERN RD.**

T-25

BLV SAMPLES FROM GARDEN

STONE DRIVE

N

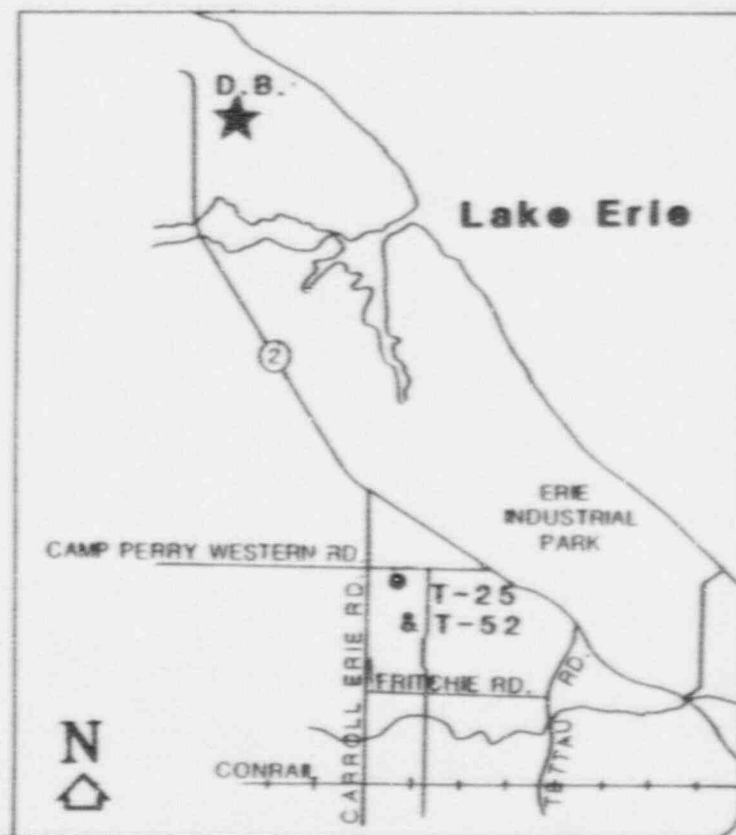
STONE DRIVE

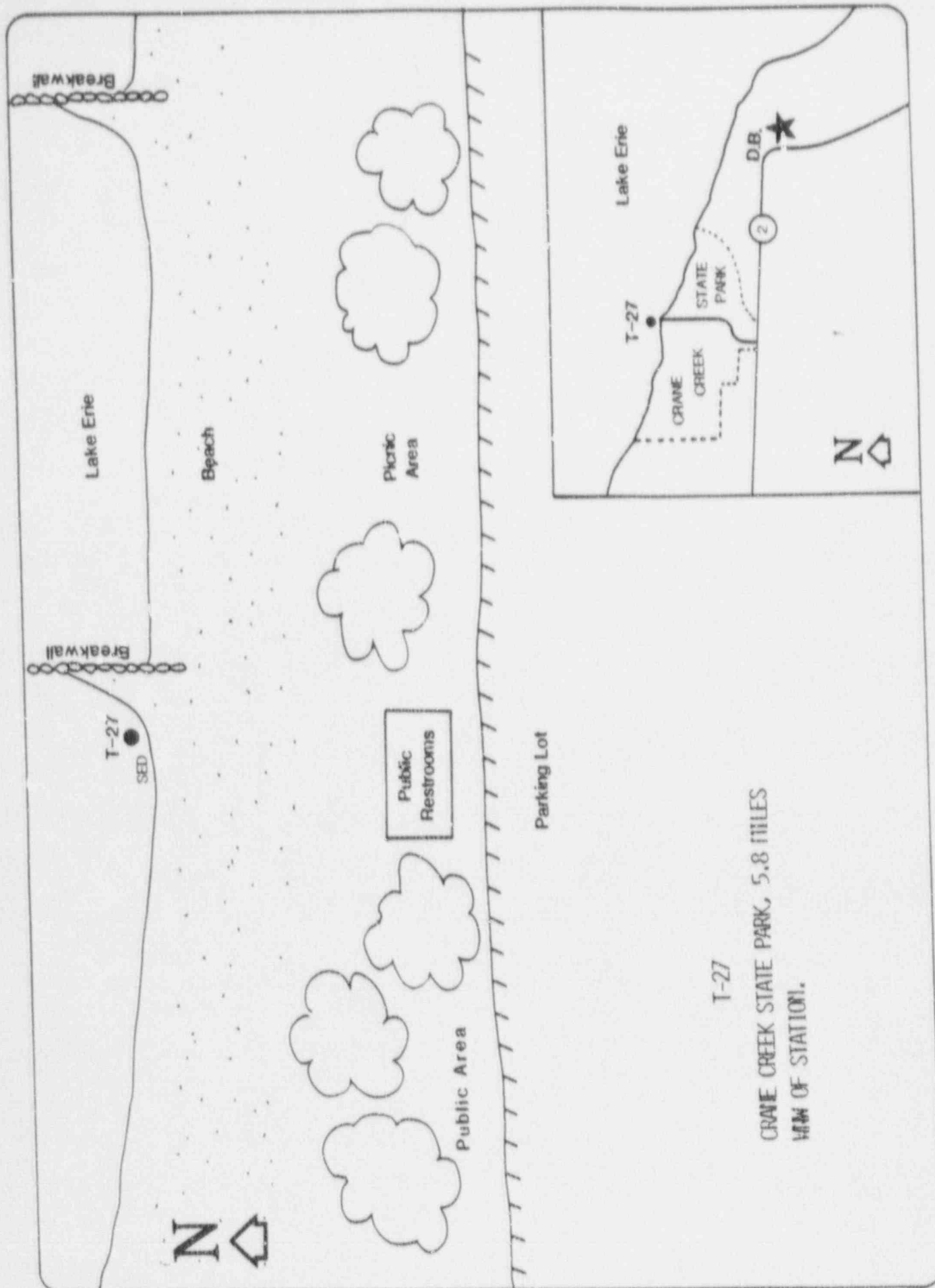
HOUSE (7898)

BARN & GARAGE

T-25 (BLV) AND T-52 (TLD)

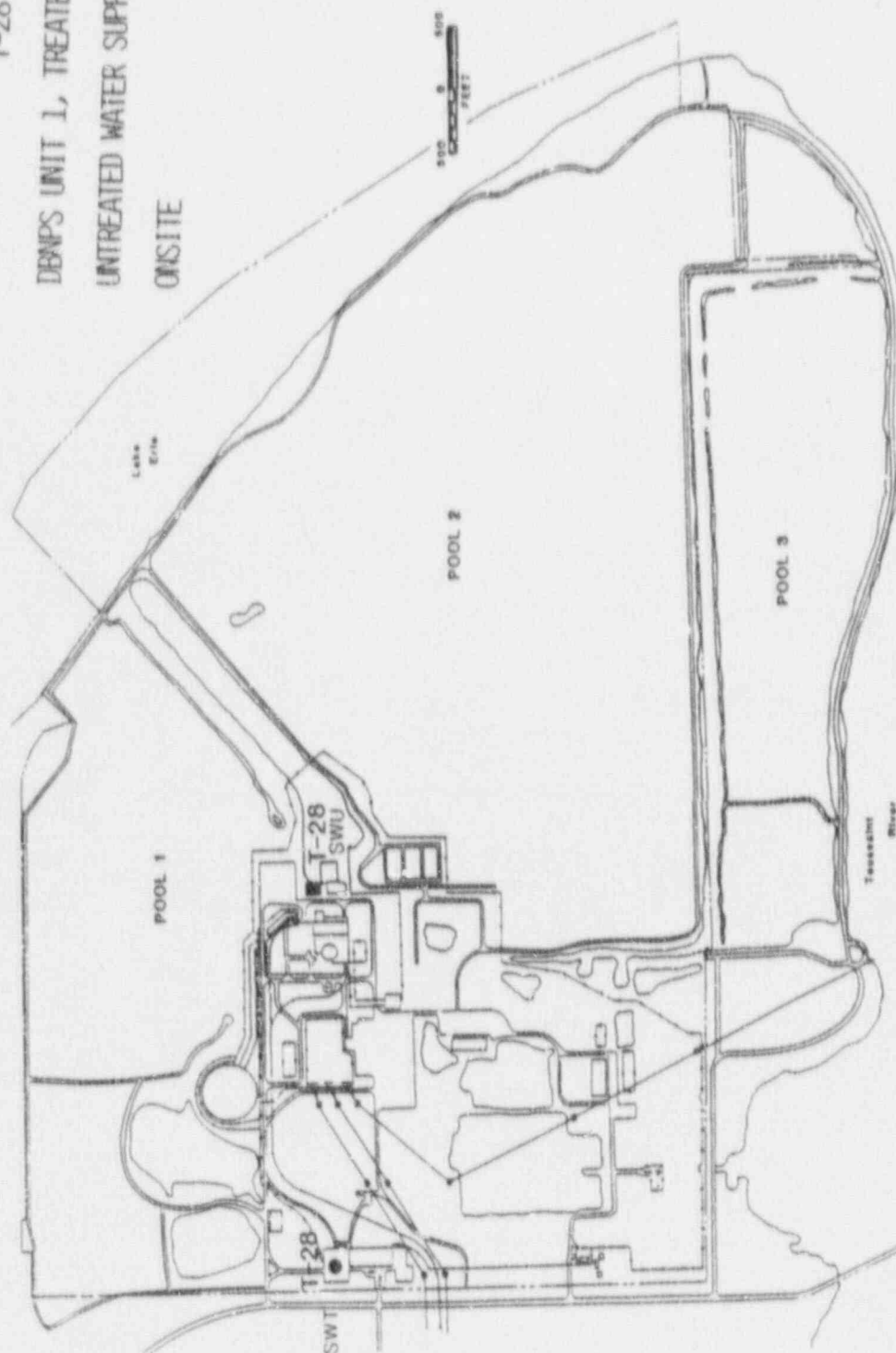
MILLER FARM, 3.7 MILES S OF  
STATION ON CAMP PERRY WESTERN ROAD.

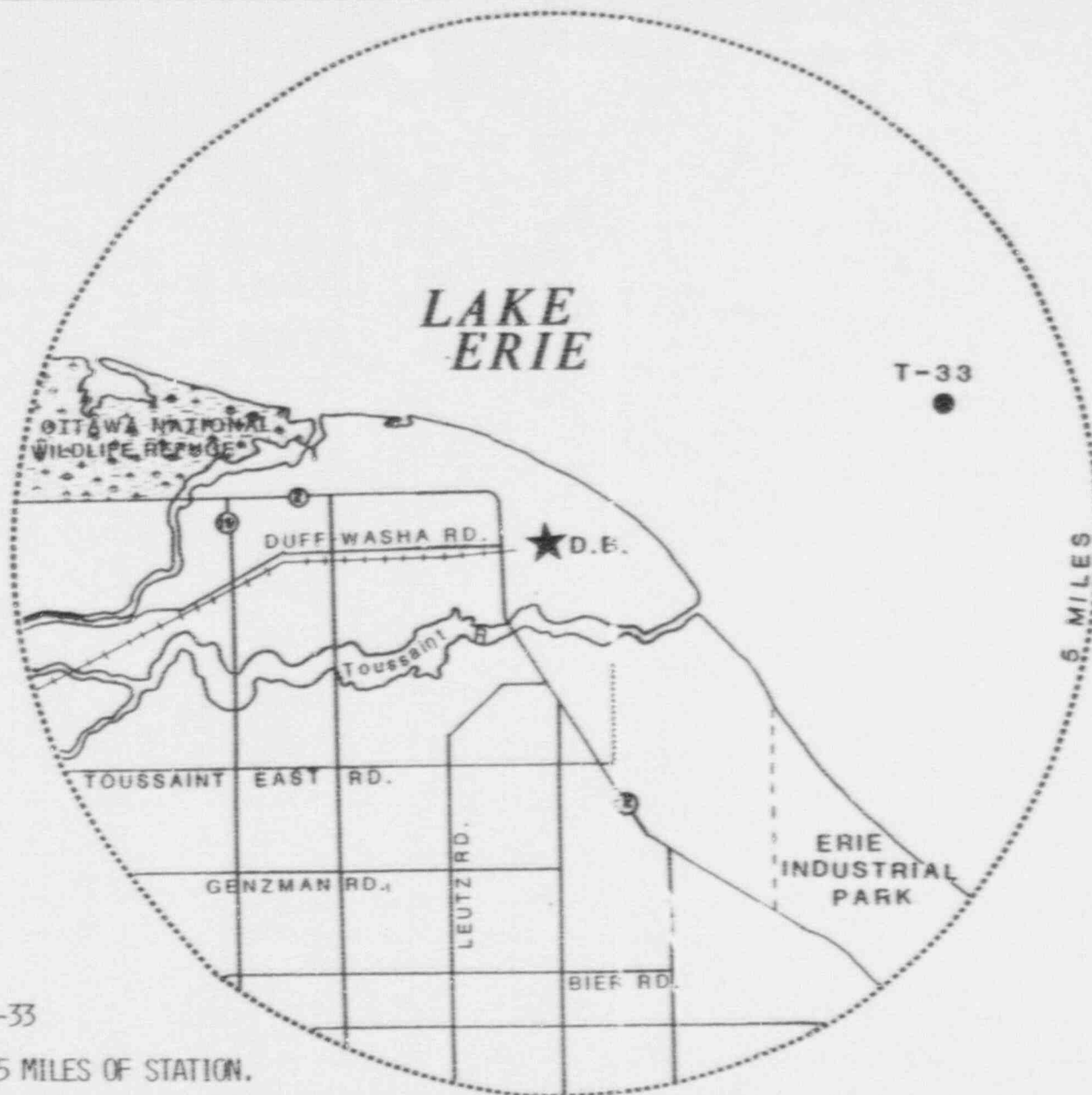




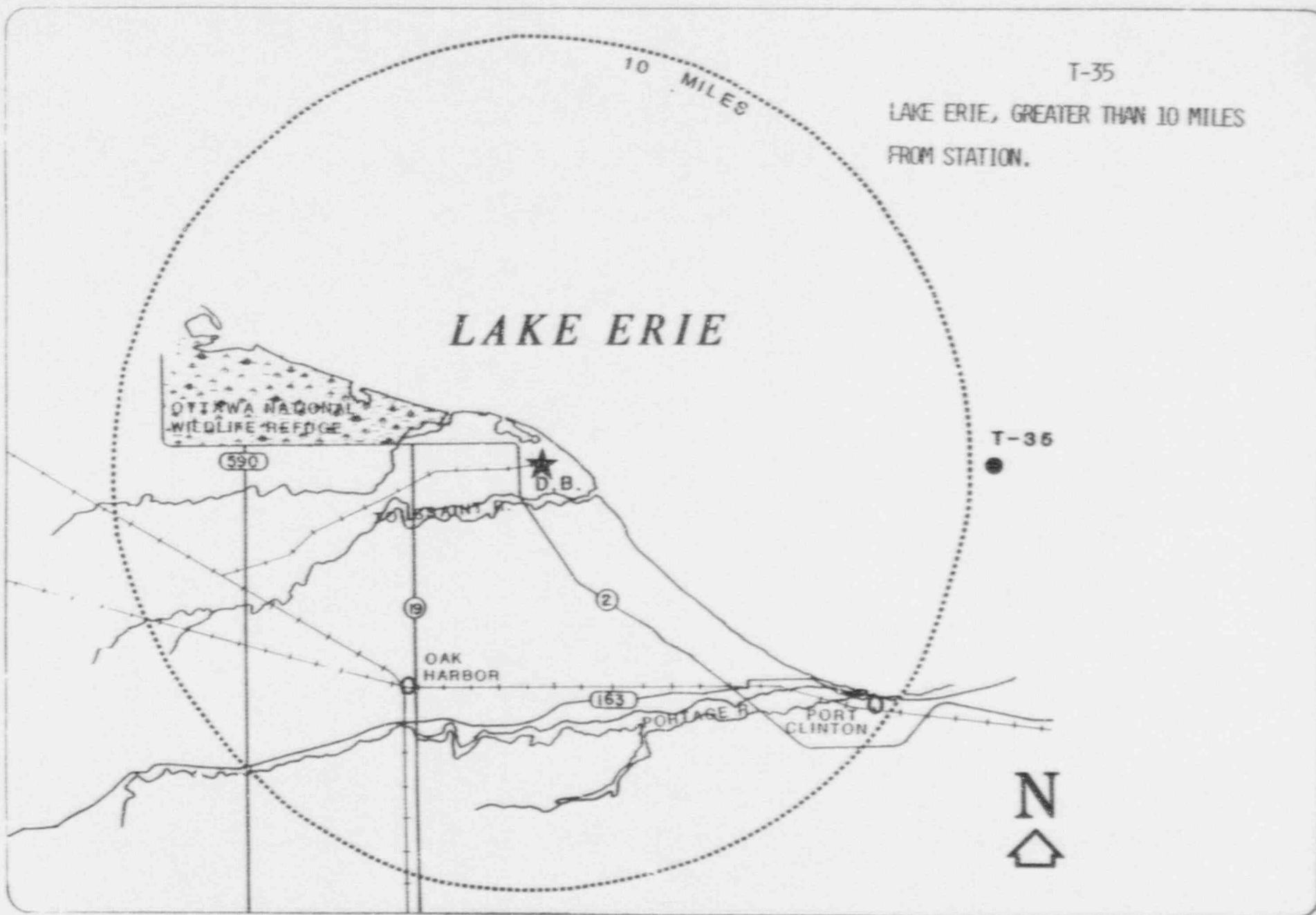


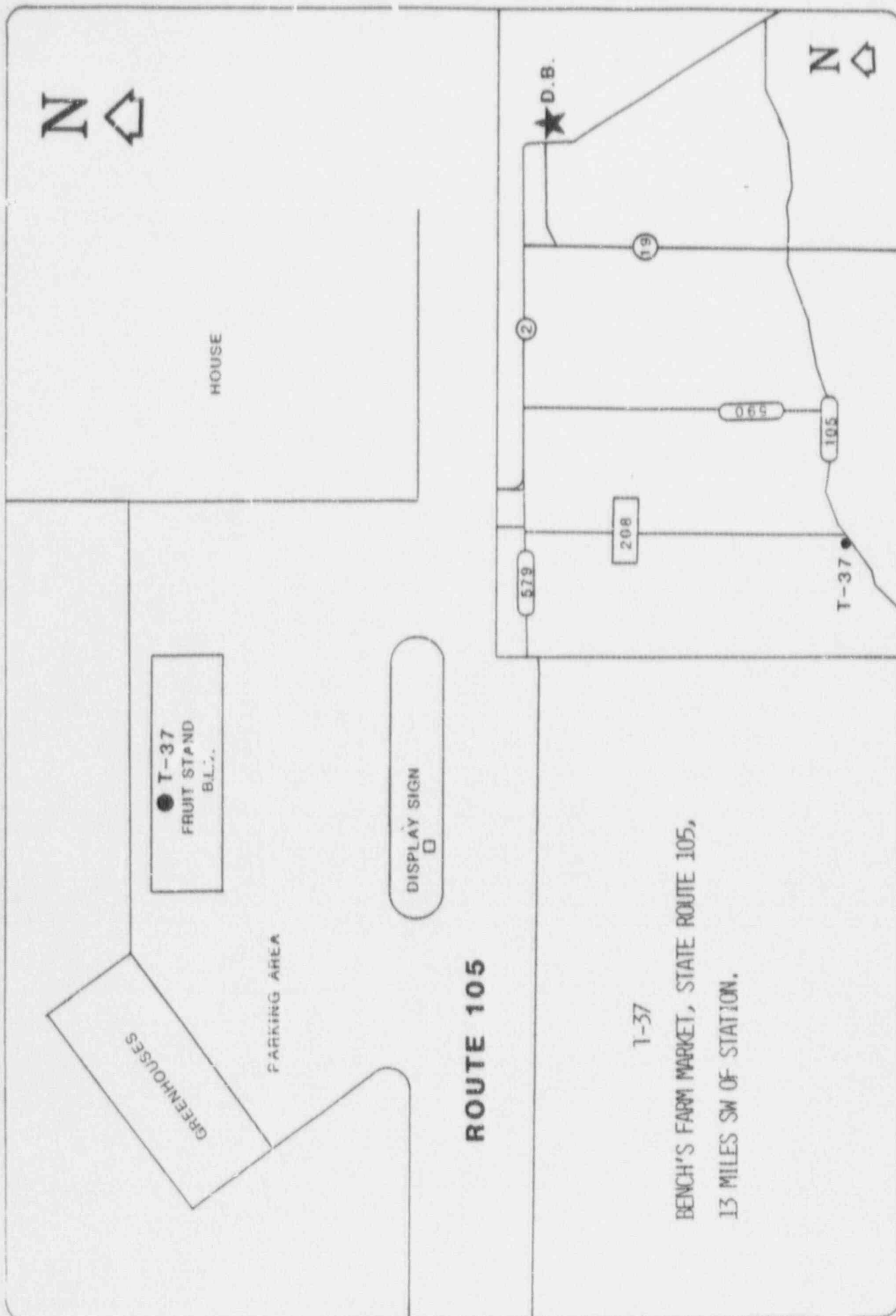
T-28  
 DWAPS UNIT 1, TREATED AND  
 UNTREATED WATER SUPPLY,  
 ONSITE



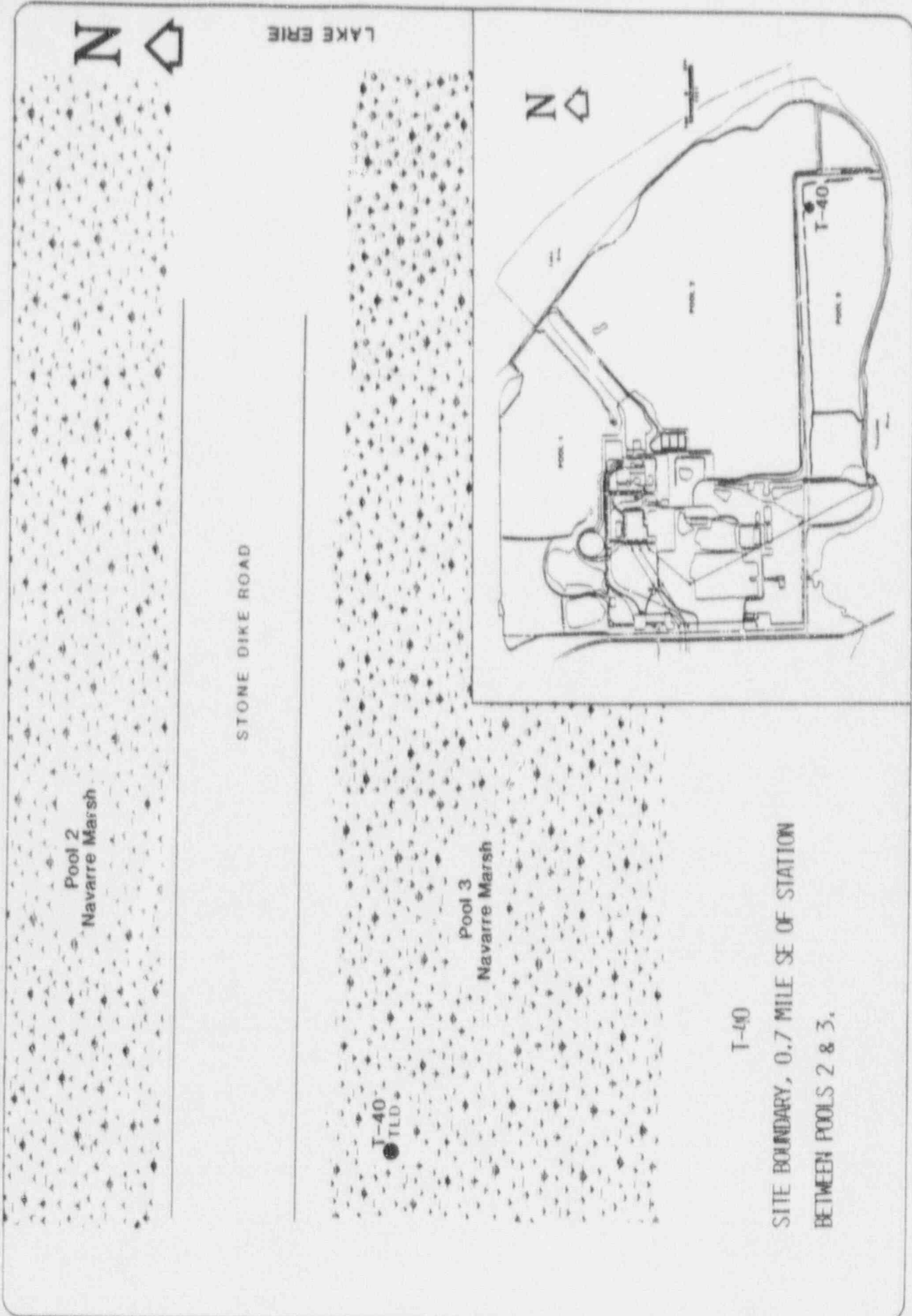


LAKE ERIE WITHIN 5 MILES OF STATION.









SITE BOUNDARY, 0.7 MILE SE OF STATION  
BETWEEN POOLS 2 & 3.

N

Pool 2

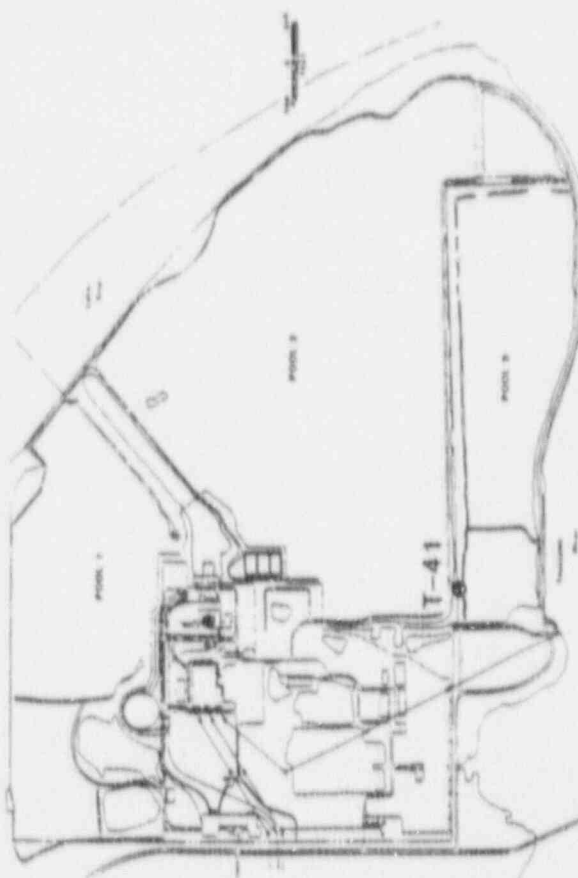
Navarre Marsh

STONE DIKE ROAD

T-41  
TLD

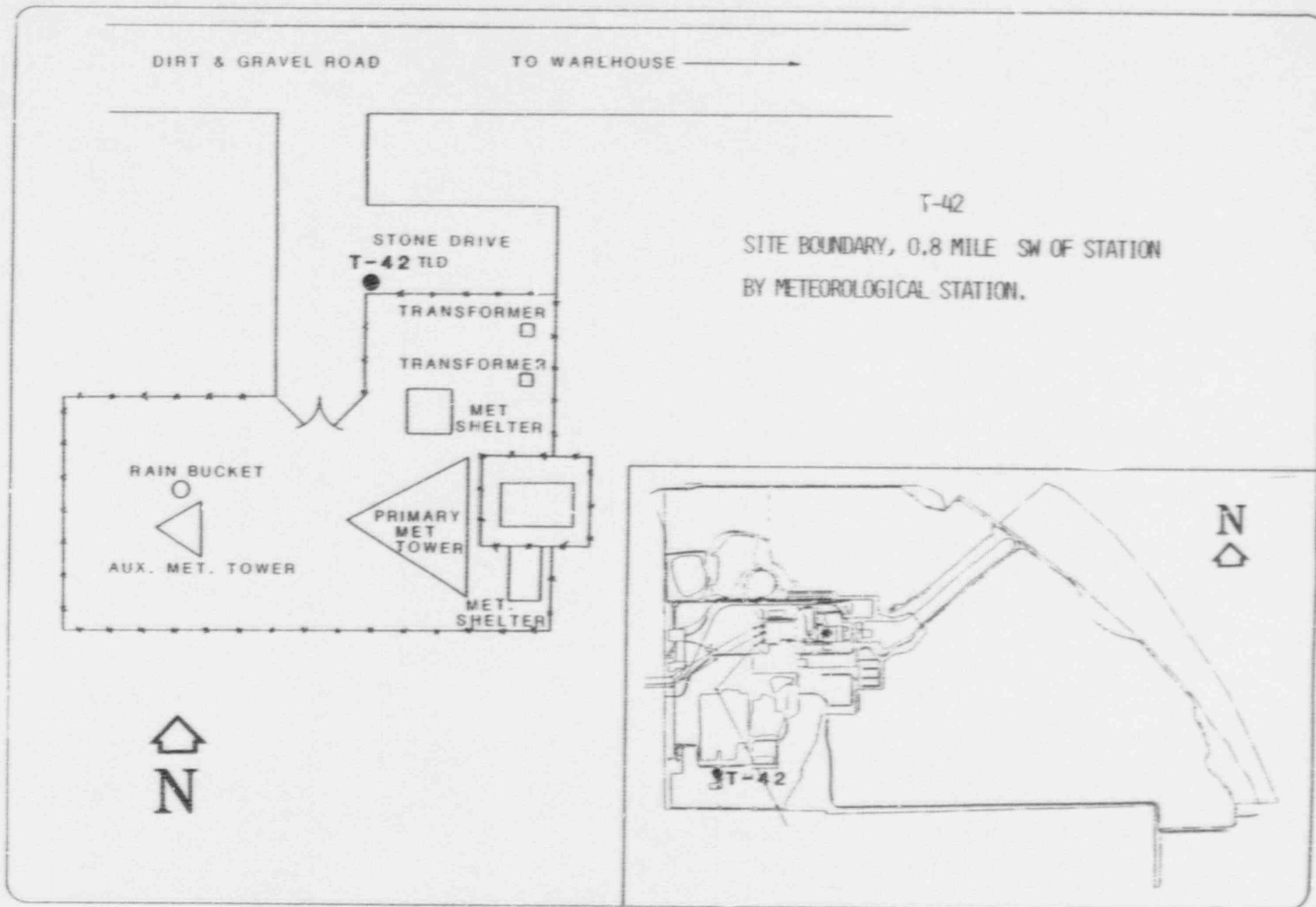
Pool 3

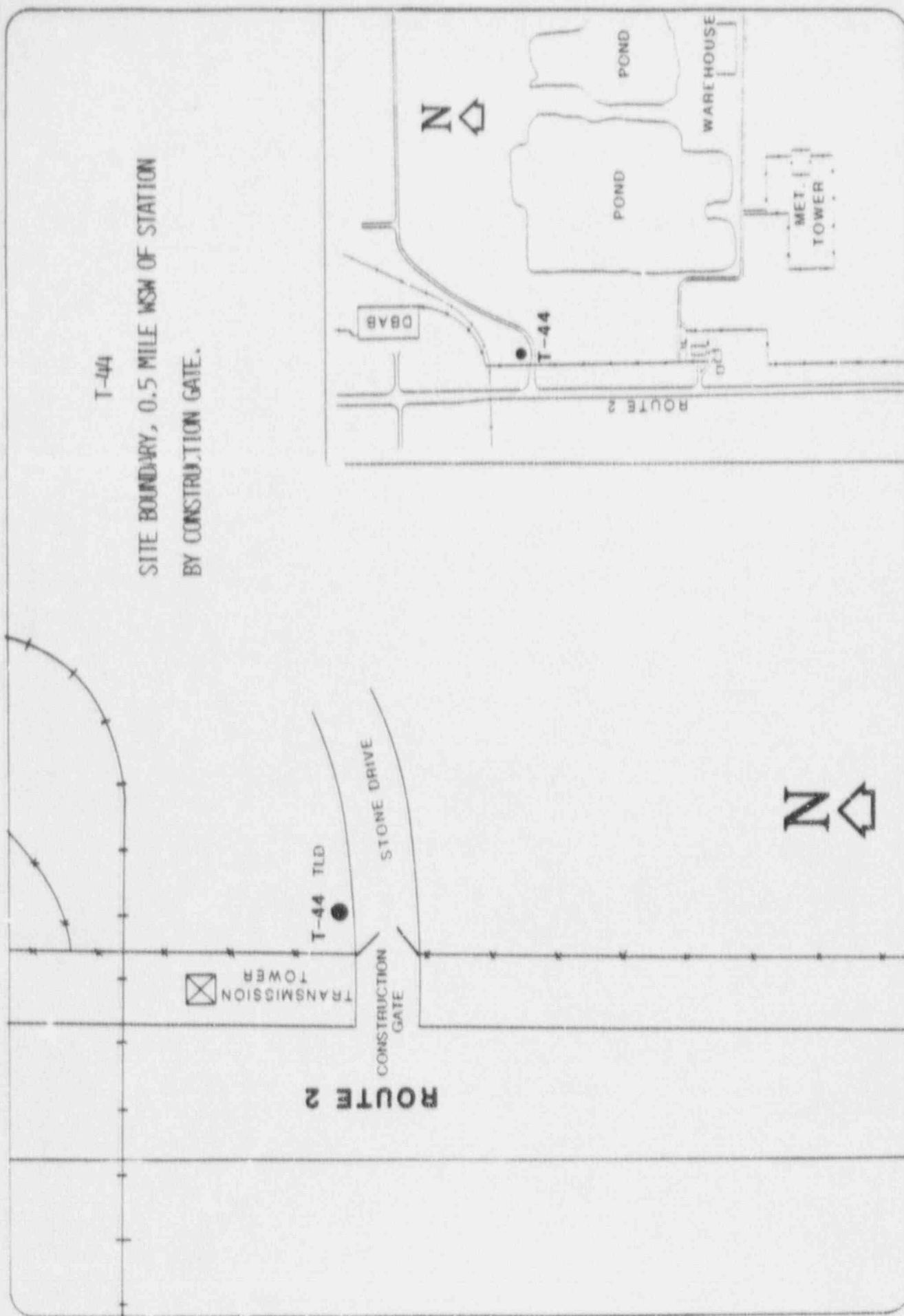
Navarre Marsh



T-41

SITE BOUNDARY, 0.6 MILE SSE OF STATION  
BETWEEN POOLS 2 & 3.

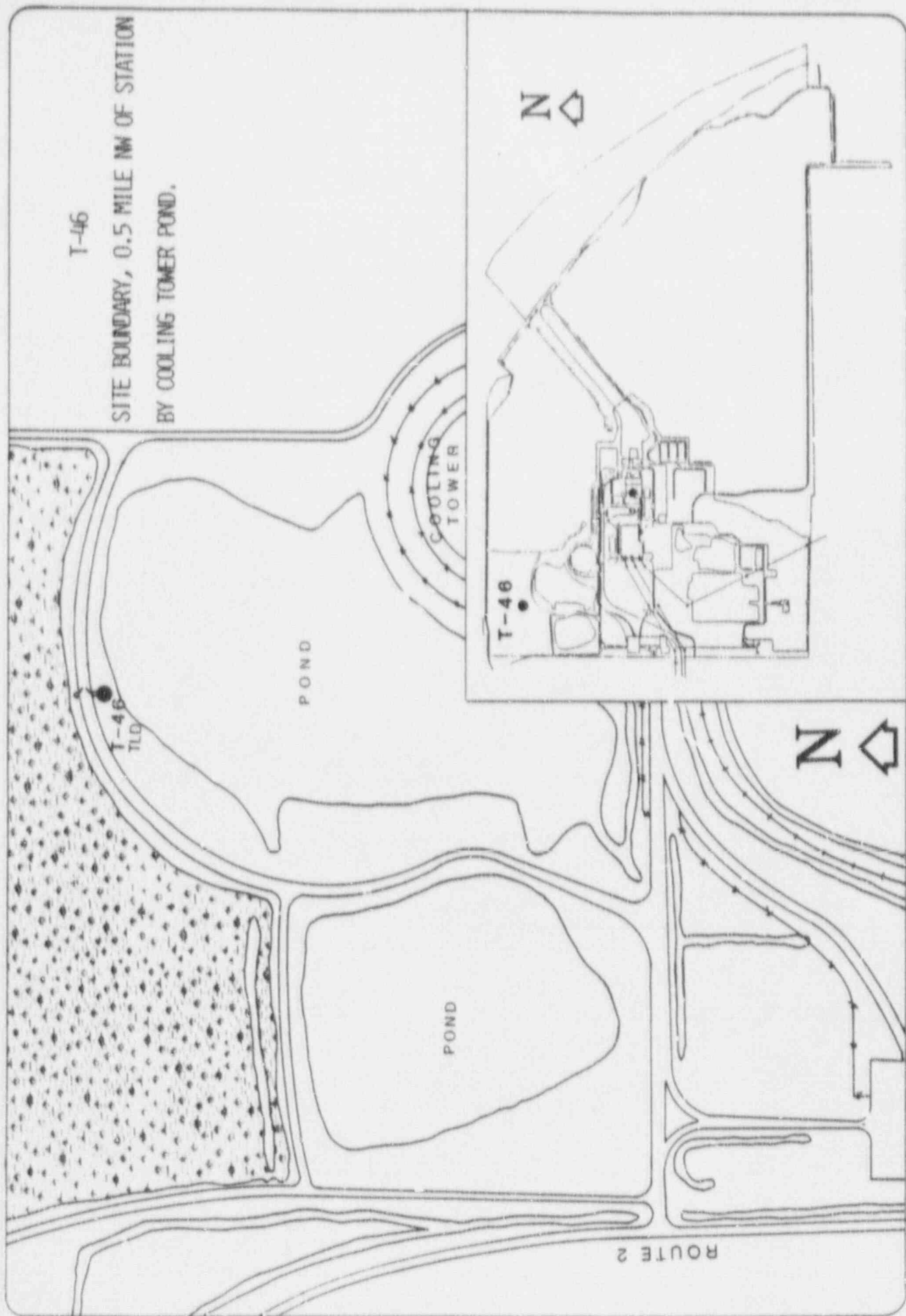




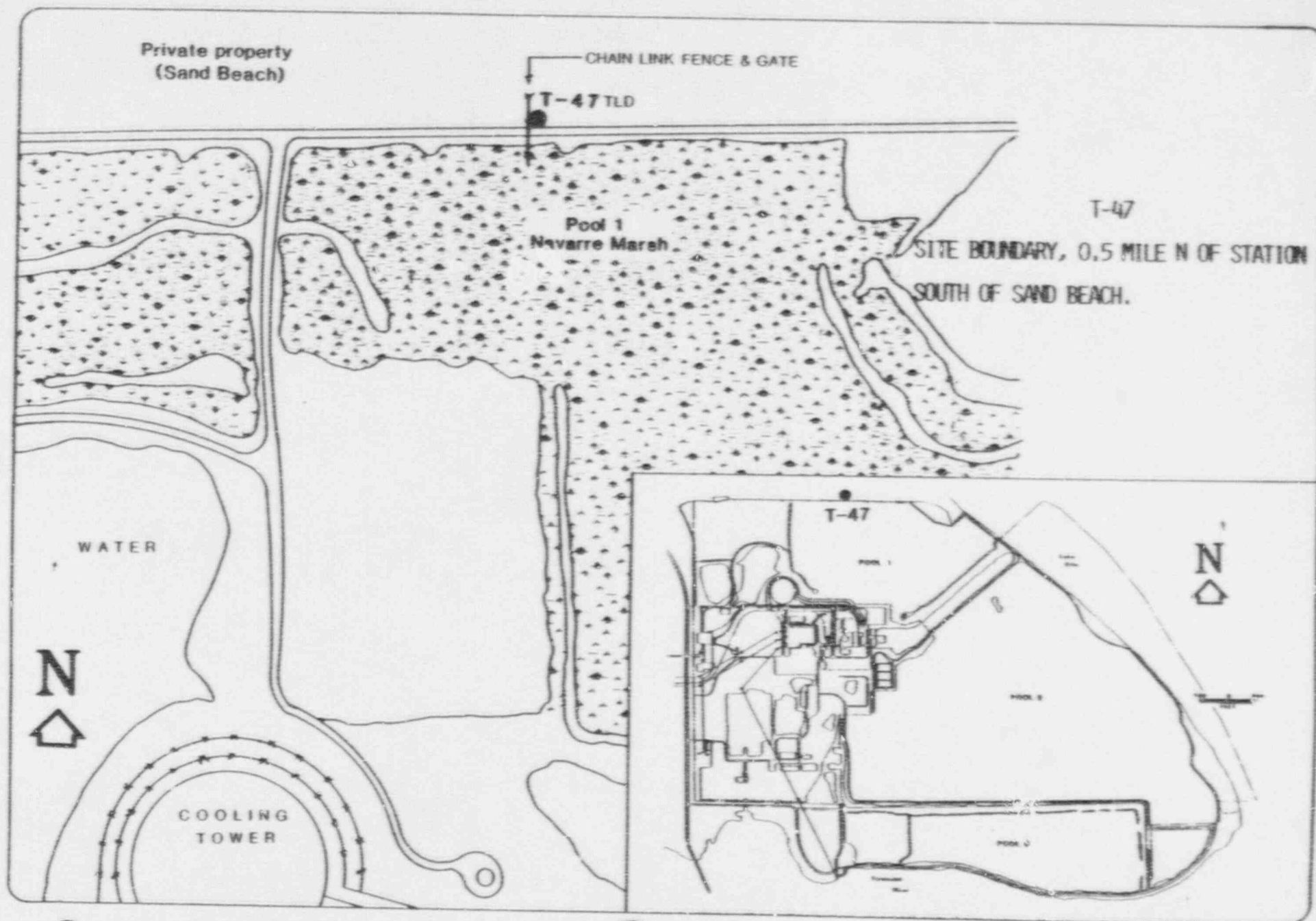
T-44

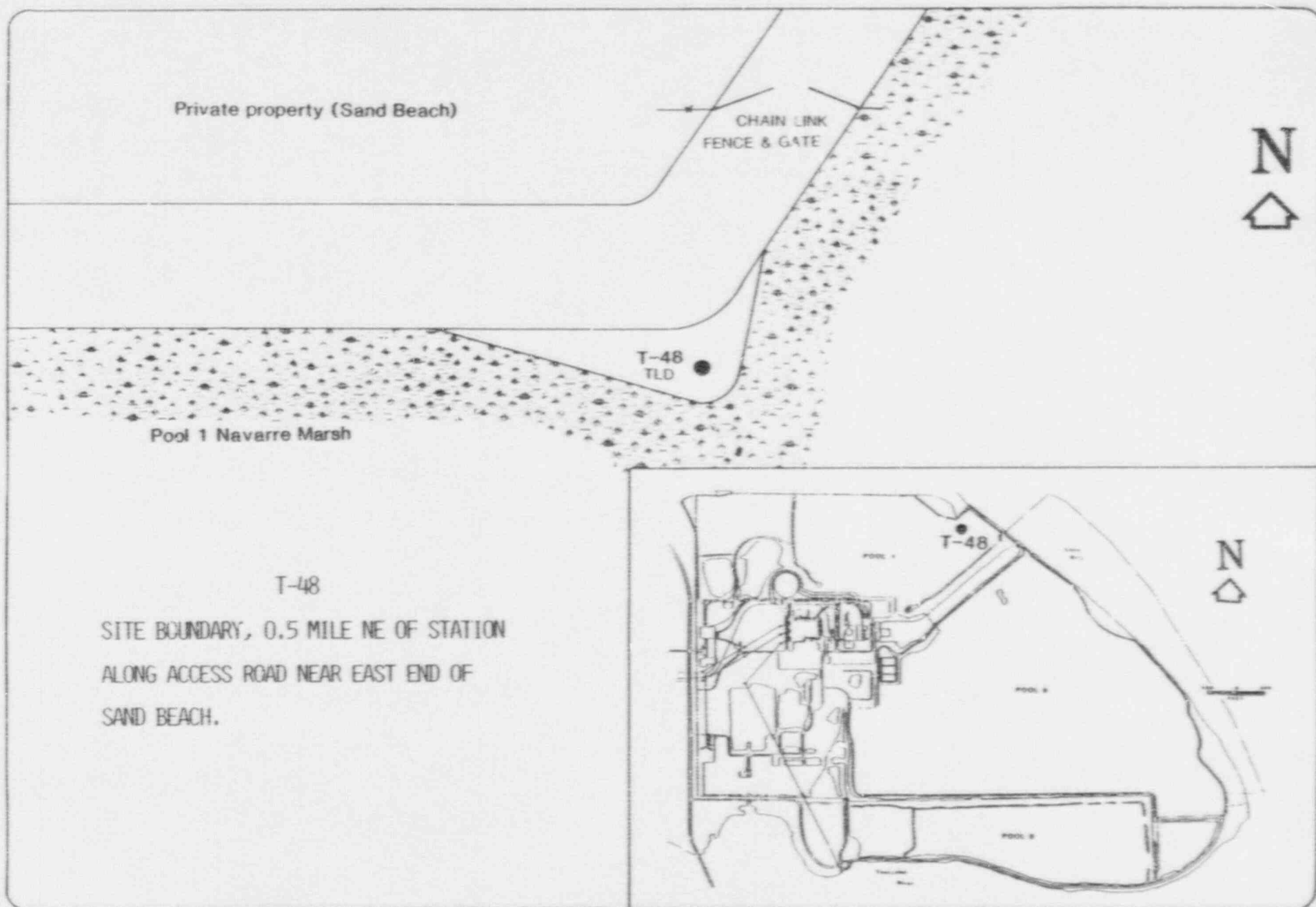
SITE BOUNDARY, 0.5 MILE WSW OF STATION  
BY CONSTRUCTION GATE.

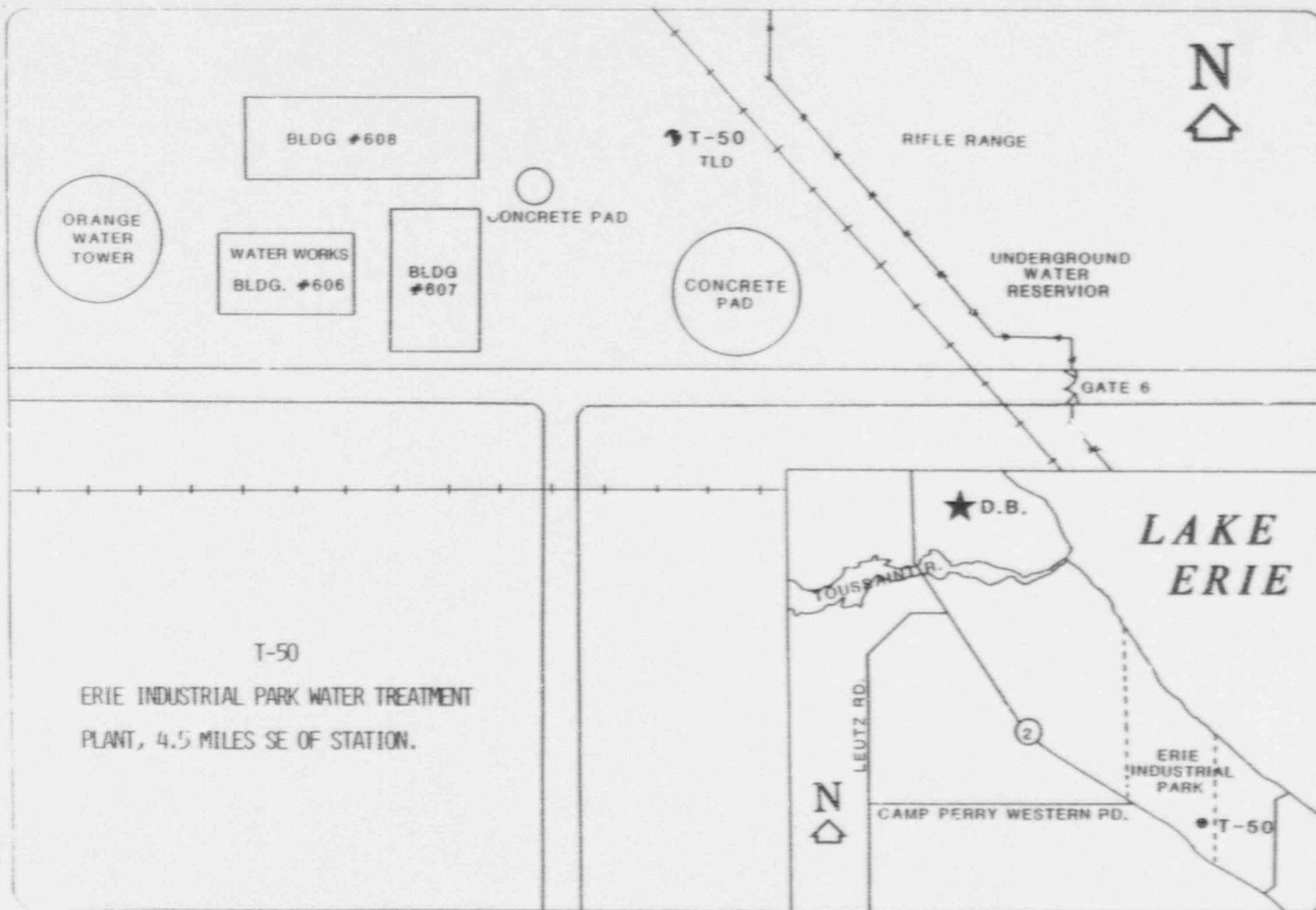




T-46  
 SITE BOUNDARY, 0.5 MILE NW OF STATION  
 BY COOLING TOWER POND.



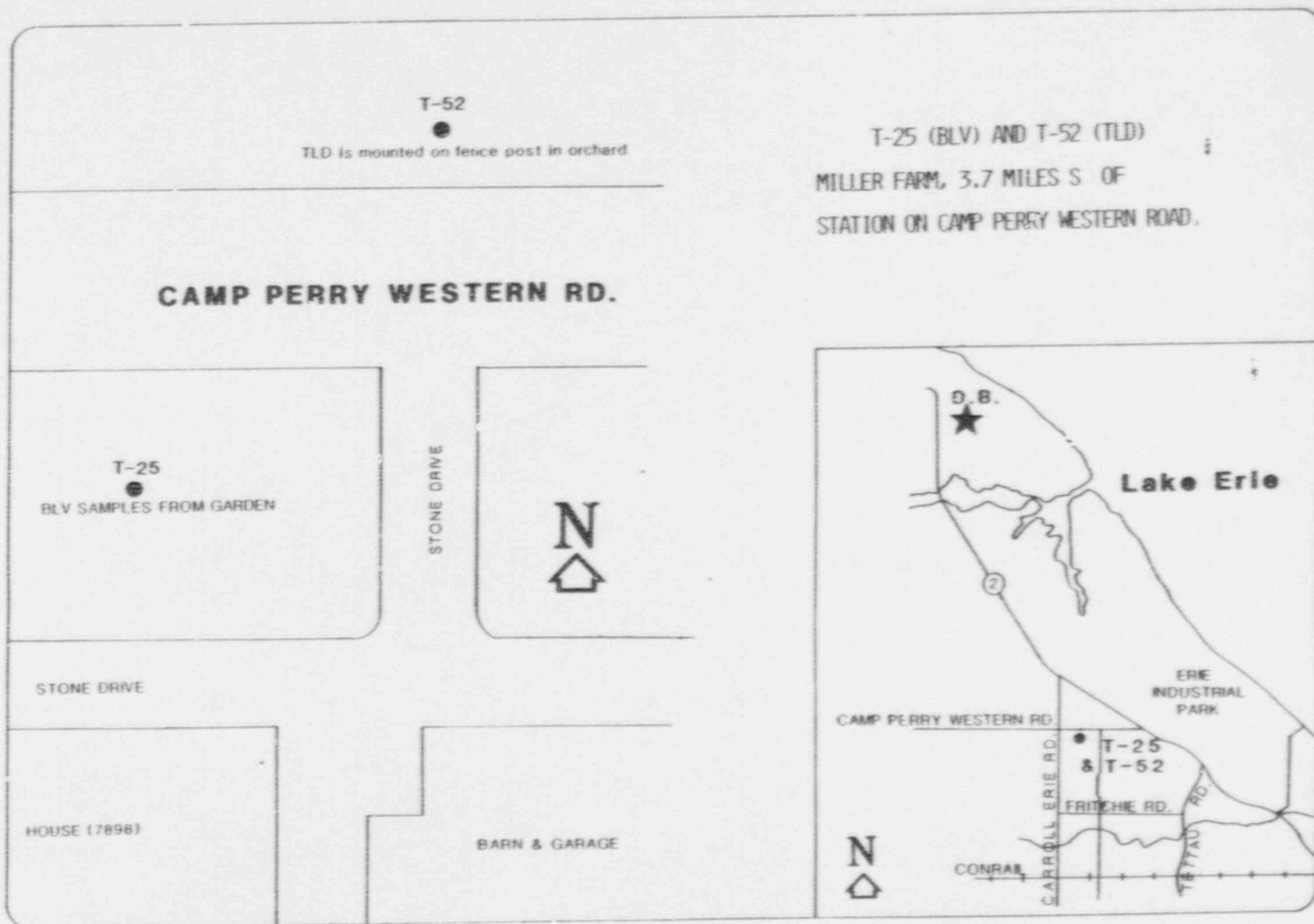




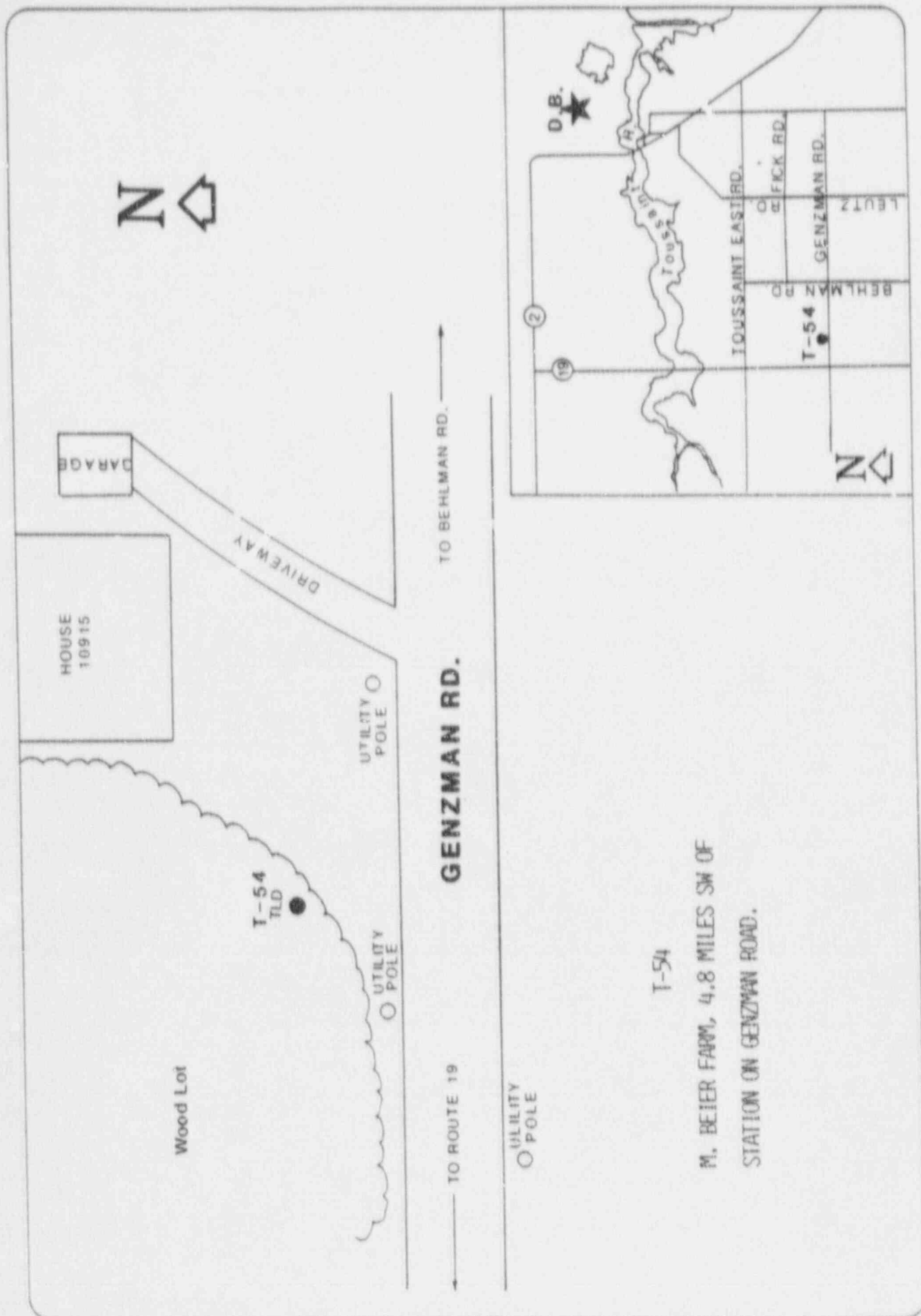
Davis-Besse OOCM

C-29

Revision 4, 1991







# STATE ROUTE 2

HOUSE #5967

Tool  
Shed

STONE DRIVE

LEMON ROAD

T-55

TLD SAMPLES MOUNTED  
ON FENCE POST

T-55

KING FARM, 5 MILES W OF STATION  
AT CORNER OF LEMON ROAD & ROUTE 2.

N

Lake Erie

T-55

2

WASHA RD.

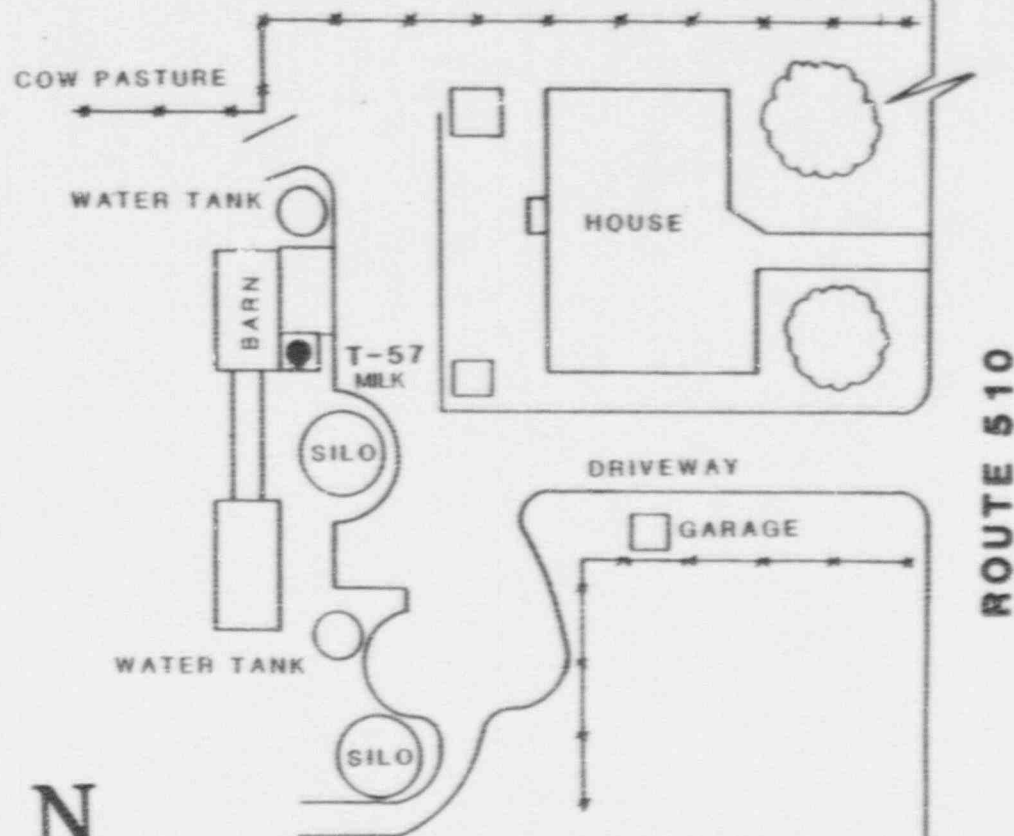
DUFF

Tousant R.

D.B.

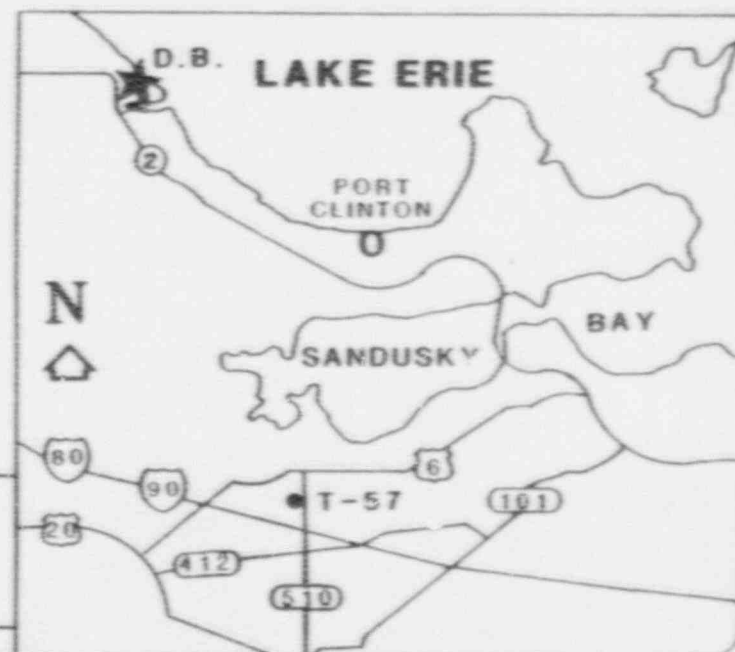
N

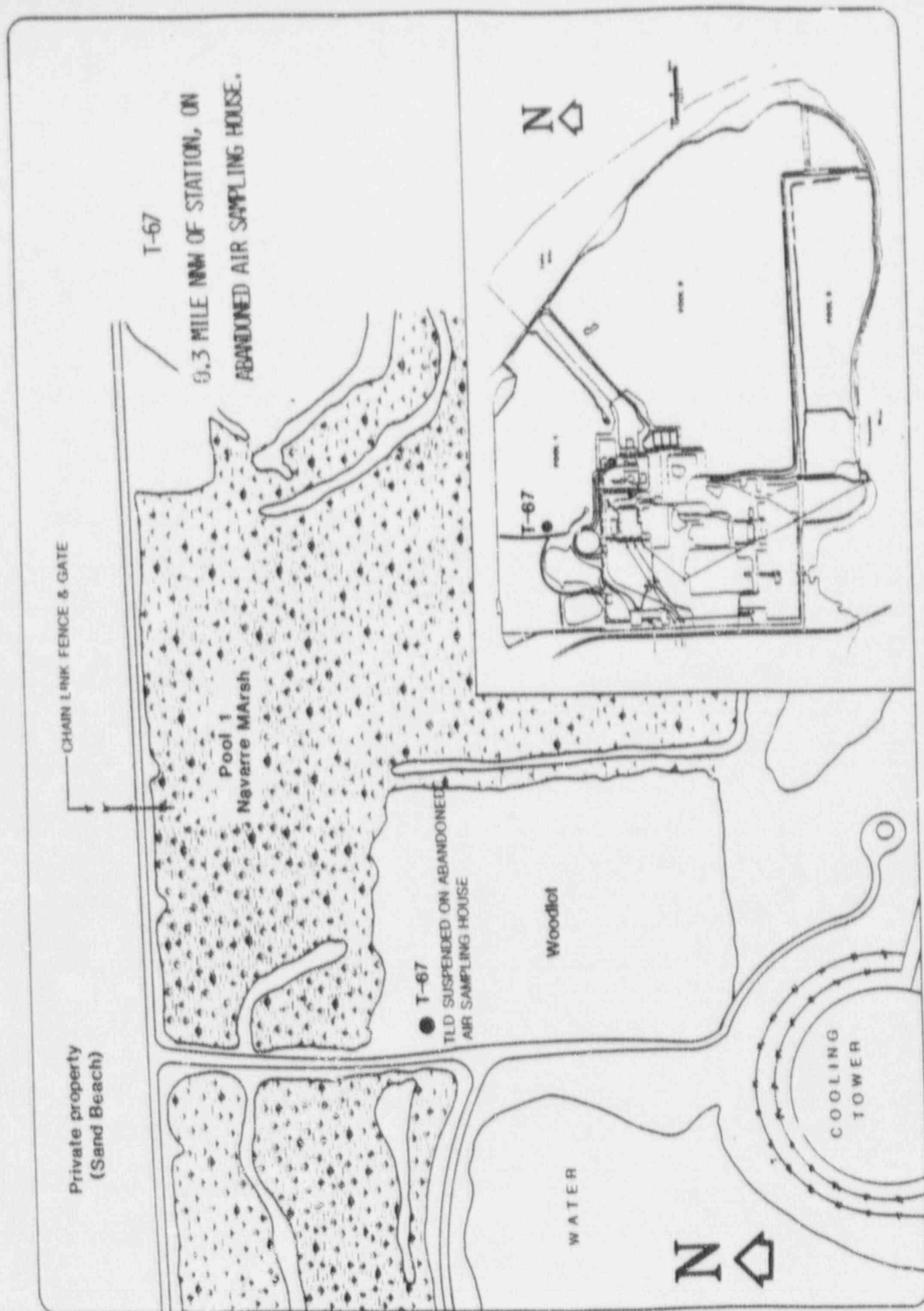
## ROUTE 6



T-57

STATE ROUTE 2 TO ROUTE 6 EXIT,  
RIGHT ON RT-6, TRAVEL ABOUT 10.6  
MILES, TURN LEFT ONTO RT 510. TRAVEL  
1.9 MILES ON RT 510. MEEK'S FARM  
IS ON THE RIGHT, 1318 N. SR 510.





N

T-68

TLD ON FENCE POST

Dike Road

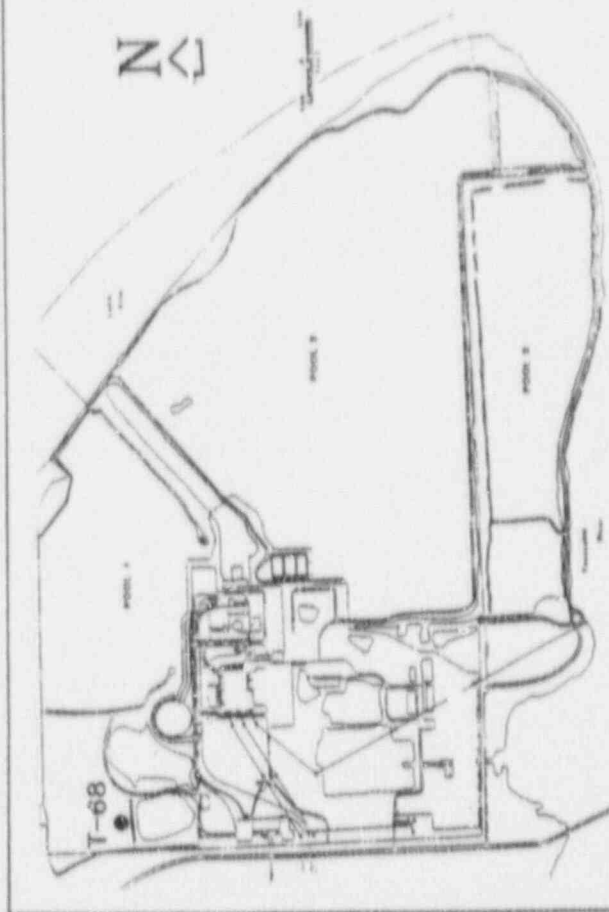
T-68

0.5 MILE WNW OF STATION, BY  
SMALLER COOLING TOWER POND.

Larger cooling  
tower pond

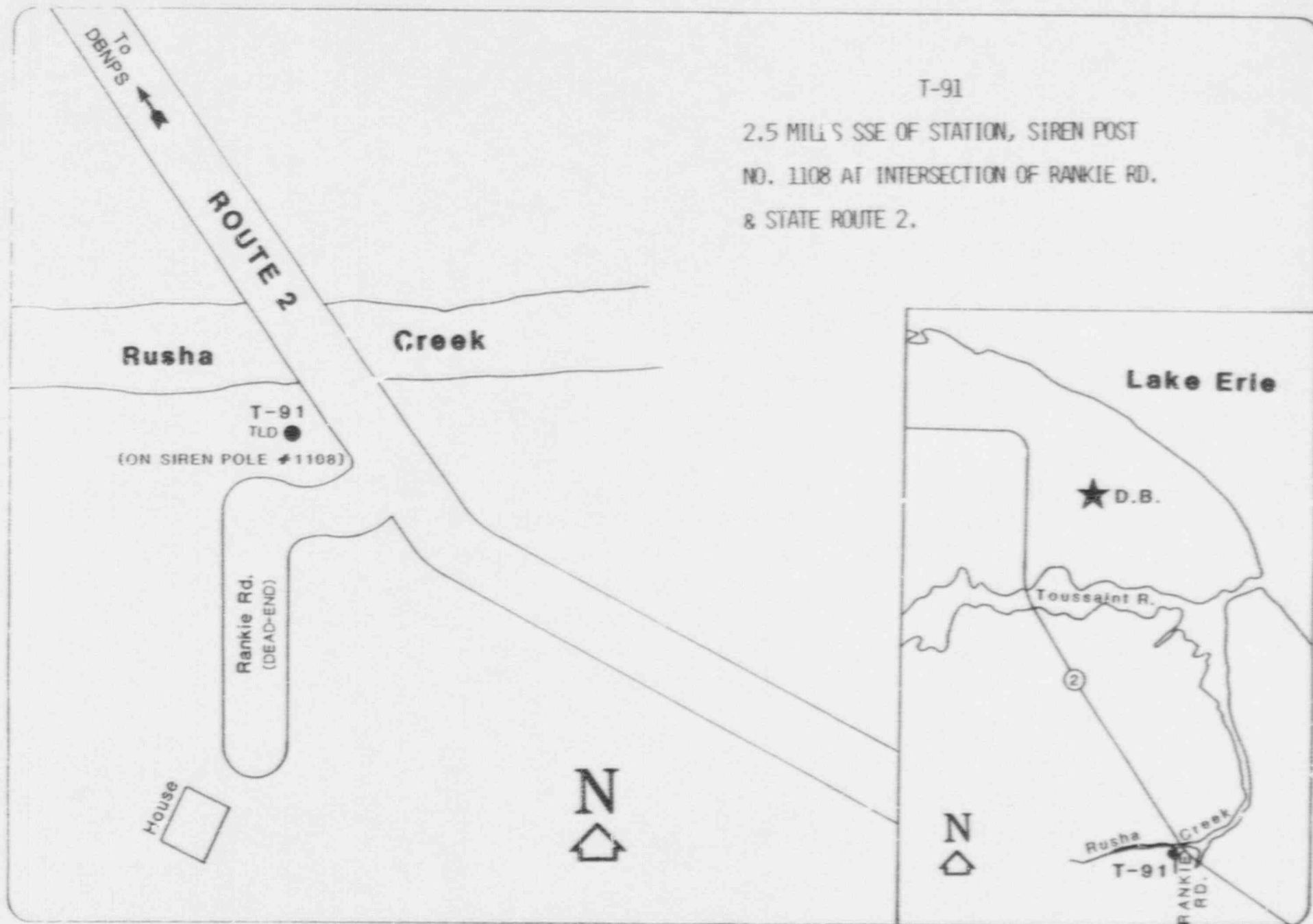
Smaller  
cooling tower  
pond

N



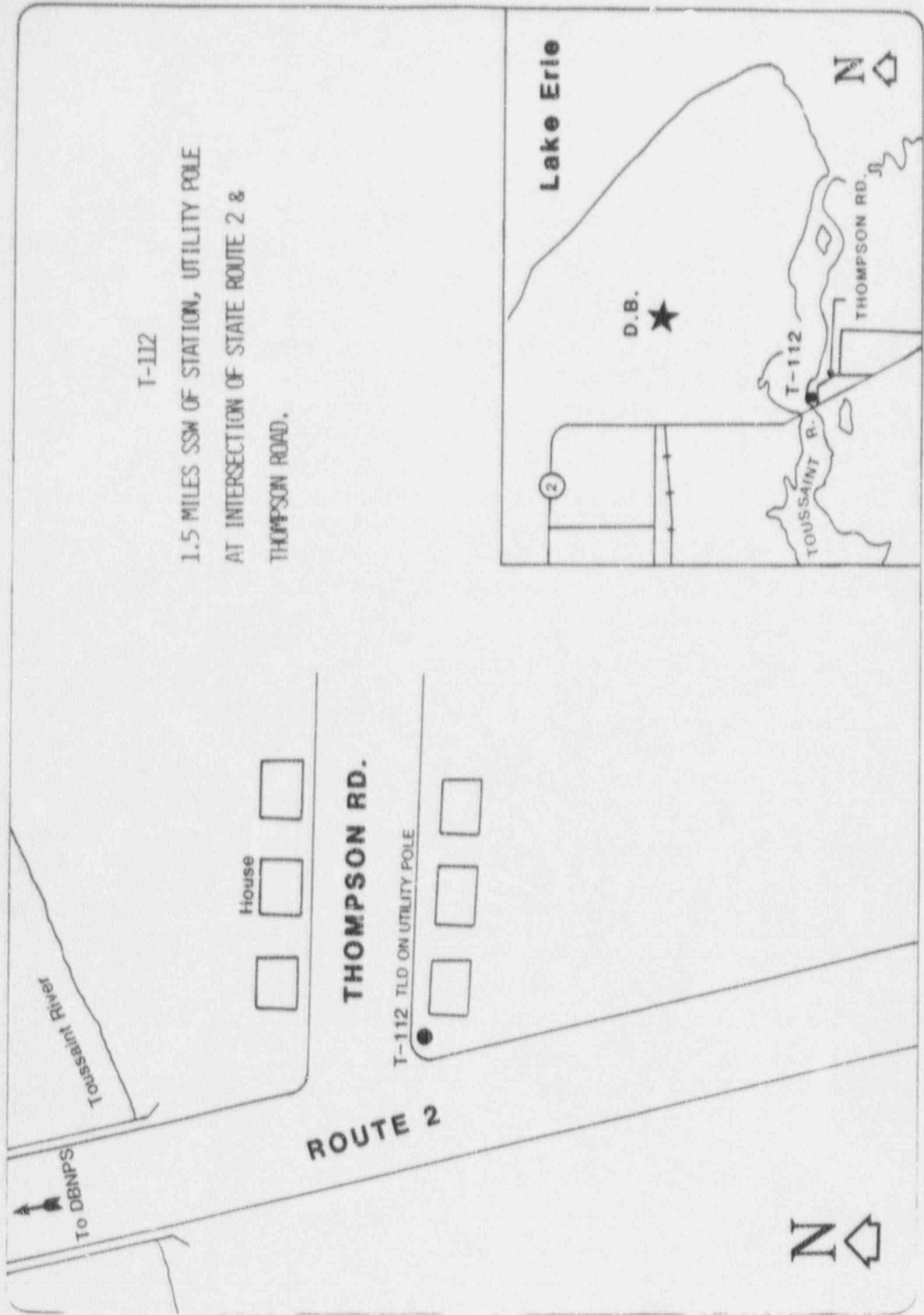
Route 2





T-91

2.5 MILES SSE OF STATION, SIREN POST  
NO. 1108 AT INTERSECTION OF RANKIE RD.  
& STATE ROUTE 2.



T-112

1.5 MILES SSW OF STATION, UTILITY POLE  
AT INTERSECTION OF STATE ROUTE 2 &  
THOMPSON ROAD.

THOMPSON RD.

T-112 TLD ON UTILITY POLE

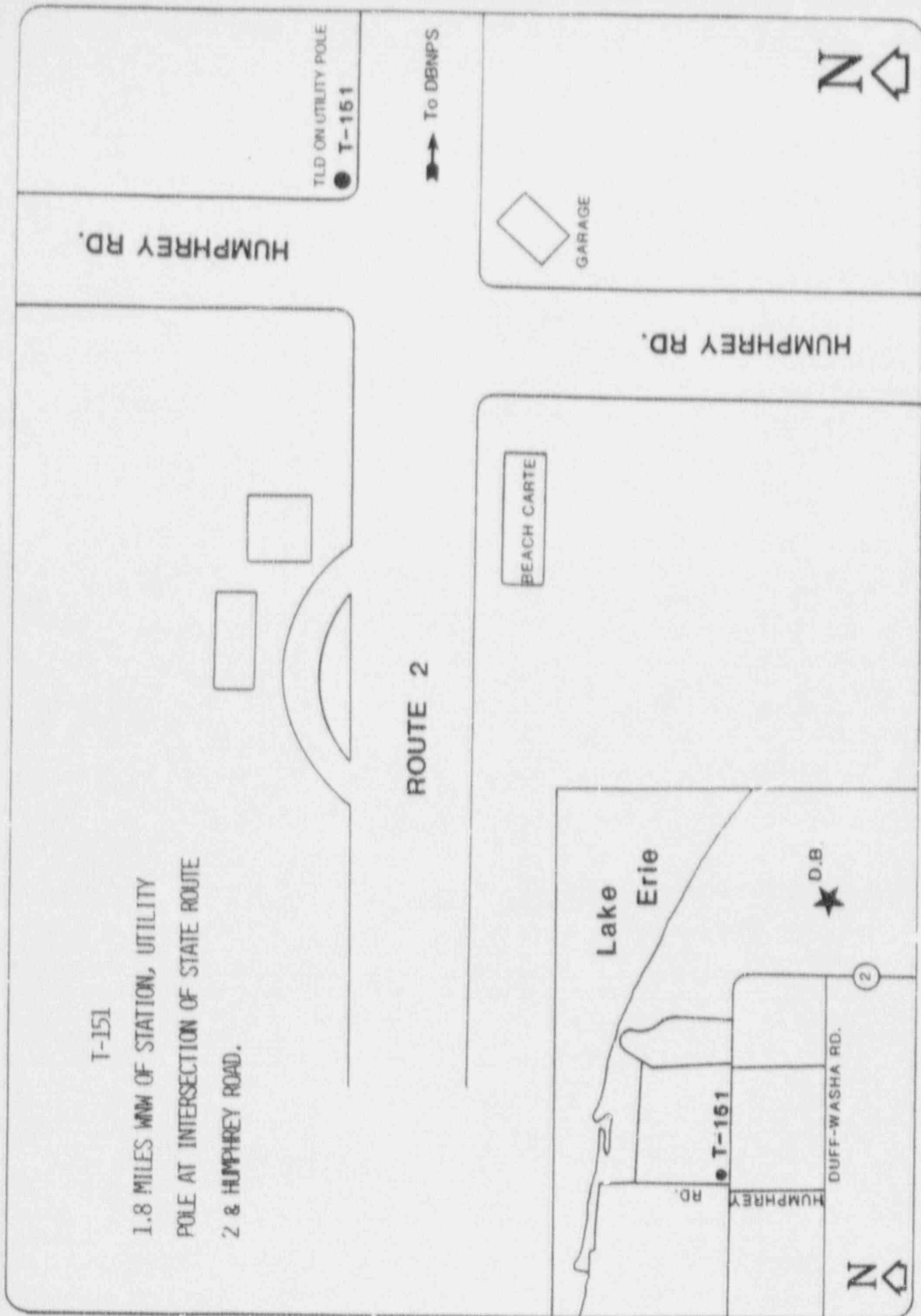
ROUTE 2

Lake Erie

D.B.

N

N



APPENDIX J  
Justifications

Safety Evaluation for the Davis-Besse  
Radiological Effluent Technical Specifications Amendment

Overview

Revision to the Davis-Besse Appendix A and Appendix B Technical Specifications are proposed which will implement the regulatory requirement of 10 CFR 50, Appendix I on ALARA for radioactive effluents and other NRC regulations and criteria on radioactive material monitoring instrumentation, radioactive material control, and radiological environmental monitoring. In keeping with NRC guidelines, all radiological requirements are being deleted from Appendix B and placed in Appendix A.

This proposed amendment is a revision to a previously submitted amendment to the NRC dated March 16, 1979 (Serial No. 488).

The major areas that are addressed in the revised submittal are as follows:

- . Liquid and gaseous effluent monitoring instrumentation -- operation and periodic operability checks;
- . Liquid and gaseous radioactive material releases--maximum release rates, quarterly dose limits and yearly dose limits;
- . Sampling and analysis requirements on batch and continuous radioactive material releases;
- . Operation requirements on the liquid radwaste treatment system;
- . Curie inventory limit on outside temporary liquid storage tanks;
- . Maximum allowable oxygen concentration in the waste gas system;



- ° Requirements to assure all solid waste meets applicable burial site requirements;
- ° Radiological environmental monitoring program--minor revisions to reflect current program and current NRC guidelines.

Changes have also been made to Section 6 of Appendix A to reflect the applicable administrative controls needed for the Section 3/4 revisions.

A notable addition to the amendment is the inclusion of a requirement for an off-site Dose Calculation Manual (ODCM) and a Process Control Program (PCP). The ODCM and PCP are not licensed documents but are referenced in the Technical Specification as presenting acceptable methods for evaluating compliance with applicable Technical Specification requirements. The ODCM provides calculational methods for determining radioactive effluent instrumentation alarm setpoints, and for evaluating releases of radioactive effluents and corresponding doses. The ODCM also includes the sampling locations for the environmental monitoring program. The PCP presents the methods used to verify that waste (dewatered resins) as processed for disposal meets appropriate shipping and burial ground regulations. Changes may be made to these documents without NRC approval; review by the SRB is required.

#### Safety Evaluation

An evaluation of the revised amendment has been performed to assure that the revisions as proposed do not involve an unreviewed safety question as defined in 10 CFR 50.59. The three criteria of 10 CFR 50.59 for the unreviewed safety question determination are addressed below.

- i) Probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

Except for the addition of the turbine building liquid effluent radiation monitor (for which an FCR has already been initiated), no plant equipment modifications are required by the proposed amendment. Certain procedural requirements will need to be developed but these address routine radioactive material effluents and controls; no accident procedures are involved.

- ii) Probability for accident or malfunction of a different type than any evaluated previously in the SAR may be created.

For reasons as stated in response to item (i) above, the proposed amendment does not directly or indirectly pose a probability for an accident or malfunction. The amendment will implement the NRC regulations for routine releases and controls of radioactive material. The amendment does not address any engineered safety features of the plant design.

- iii) Margin of safety as defined in the basis for any technical specification is reduced.

The proposed amendment does not reduce the margin of safety. The proposed amendment addresses routine releases and control of radioactive material; except as noted in item (i), no plant modifications are involved. Several operating procedure changes may be needed, but these changes will be only for routine operations and will have no impact on accident probability or consequences.

For the reasons discussed above for each of the criteria of 10 CFR 50.59, it is concluded that the amendment as proposed does not involve an unreviewed safety question.

### Service Water System--Radiological Effluent Monitoring Requirements

The service water system is classified as a non-radioactive system, being removed from radioactive systems by two boundaries. Radioactive systems are serviced by the component cooling water system interface; and, the service water system provides cooling to the component cooling water system through closed loop heat exchangers. Therefore, any leaks from radioactive systems into the plant water systems would first be identified by the monitoring of the component cooling water system prior to any additional unexpected leakage into the service water system. As a prudent measure, the service water system is monitored in accordance with the NRC guidance of Standard Review Plan, Section 11.5. However, because this system is a non-radioactive system and is separated from radioactive systems through two closed-loop boundaries, no Technical Specification requirements are needed for routine monitoring and analysis for radioactive effluents.



### Radioactive Effluent Instrumentation--Automatic Isolation Feature

The radioactive effluent monitoring instrumentation at Davis-Besse does not include provisions as called for in the NRC Standard Radiological Effluent Technical Specifications for automatic isolation should any of the following conditions exist: circuit failure, downscale failure, or instrument not set in operate mode. Even though the automatic isolation features do not exist, administrative controls have been established such that should any of these conditions exist the control of radioactive effluents would not be significantly impacted. Essentially all releases of liquid radwaste are controlled as individual batch releases with predetermined allowable release conditions. Thereby the radiation monitor serves mainly as a back-up; primary control is established by the prerelease radiological analyses and evaluations. To assure the availability of the back-up monitoring, the status of the instruments is checked once per shift by the control room operators. Indicator lights on the instrument panel are checked to verify operability. An indicator would illuminate should a failure occur such as the ones delineated above. Therefore, in addition to the administrative controls on allowable releases, the verification of instrument operability prior to releases of radioactive effluents and the "once per shift" status check by the control room operators provides adequate assurance of the proper control of the radioactive effluents.

Technical Bases for Eliminating Curie Inventory  
Limit for Gaseous Waste Decay Tanks

The NRC Standard Technical Specifications include a limit for the amount of radioactivity that can be stored in a single waste gas decay tank. This curie inventory limit is established to assure that in the event of a tank failure releasing the radioactive content to the environment the resulting total body dose at the site boundary would not exceed 0.5 rem. For Davis-Besse the inventory limit in the waste gas storage tank has been determined to be approximately 45,000 curies (Xe-133, equivalent).

An allowable primary coolant radioactivity concentration is established by the Technical Specifications which limit the primary coolant radioactivity concentrations to  $100/E$  with  $E$  being the average energy of the radioactivity in Mev. This equation yields an upper primary coolant gross activity limit of about 200  $\mu\text{Ci}/\text{ml}$ . By applying this activity concentration limit to the total liquid volume of the primary system, a total activity limit can be determined. For Davis-Besse the primary system volume is about 56,000 gallons, which yields a limiting total inventory of approximately 41,000 Ci.

By assuming a typical radionuclide distribution an equivalent Xe-133 inventory can be determined. Table 1 provides the typical radionuclide (noble gases) distribution and the Xe-133 equivalent concentration. The equivalent concentration is determined by multiplying the radionuclide concentration by the ratio of the nuclide total body dose factor to the Xe-133 total body dose factor. Summing all the individual radionuclide equivalent concentrations provides the overall Xe-133 equivalent concentration. For determining concentration in a waste gas decay tank, a conservative assumption of 48 hours decay in degassing the primary system has been used to correct the primary coolant concentrations. The data show that the equivalent concentration (decay corrected) is less than the gross concentration (i.e., 16  $\mu\text{Ci}/\text{gm}$  total in primary coolant versus 12  $\mu\text{Ci}/\text{gm}$  equivalent). The resulting Xe-133 equivalent curie inventory for WGDT input is approximately 31,000 Ci.



Therefore, even if the total primary system at the maximum Tech Spec allowable concentration was degassed to a single waste gas decay tank, the tank curie inventory would be well below the 45,000 Ci limit. Based on this evaluation, the curie inventory limit on a single waste gas storage tank has not been included as a Technical Specification requirement.

Table 1

## Xe-133 Effective Concentration

Primary*	Half-life	Concentration	Reg Guide 1.109	Ratio	Xe-133
Coolant		@ 48 hr decay	TB Dose Factor	TB DF	Effective C
( $\mu\text{Ci/GM}$ )		( $\mu\text{Ci/ml}$ )	$\frac{\text{mrem/yr}}{\text{pCi/m}^3}$	Xe-133 DF	@ 48 hr dec
					( $\mu\text{Ci/ml}$ )
<hr/>					
Kr-83M	2.0-02	1.9 hr	--	7.6x10 <sup>-8</sup>	--
Kr-85M	1.1-01	4.5 hr	--	1.2x10 <sup>-3</sup>	4.1
Kr-85	7.4-02	10.7 yr	7.4x10 <sup>-2</sup>	1.6x10 <sup>-5</sup>	0.06
Kr-87	5.8-02	76.3 min	--	5.2x10 <sup>-3</sup>	20.
Kr-88	1.9-01	2.84 hr	--	1.5x10 <sup>-2</sup>	52.
Kr-89	4.8-03	3.16 min	--	1.7x10 <sup>-2</sup>	57.
Xe-131M	8.4-02	12 days	7.5x10 <sup>-2</sup>	9.2x10 <sup>-5</sup>	0.32
133M	2.0-01	2.2 days	1.1x10 <sup>-1</sup>	2.5x10 <sup>-4</sup>	0.86
Xe-133	1.5+01	5.3 days	1.2x10 <sup>+1</sup>	2.9x10 <sup>-4</sup>	1.0
Xe-135M	1.3-02	16 min	--	3.1x10 <sup>-3</sup>	11.
Xe-135	3.3-01	9.1 hr	8.5x10 <sup>-3</sup>	1.8x10 <sup>-3</sup>	6.2
Xe-137	8.7-03	4 min	--	1.4x10 <sup>-3</sup>	4.8
Xe-138	4.3-02	17 min	--	8.8x10 <sup>-3</sup>	30
Total	1.6x10 <sup>+1</sup>		1.2x10 <sup>+1</sup>		1.2x10 <sup>+1</sup>

\*Adapted from Davis-Besse Evaluation of Compliance with Appendix T to 10 CFR 50,  
June 4, 1976.

## Lower Limit of Detection--Decay Correction Factor

The equation and definition of the lower limit of detection in the NRC Standard Radiological Effluent Technical Specification include the term  $e^{-\lambda t}$  which is used to decay correct the analysis. The LLD is further defined as an a priori (before the fact) limit representing the capabilities of a measurement system and not an a posteriori (after the fact) limit for a particular measurement.

Providing a decay correction for an evaluation of the capabilities of a system does not appear appropriate. It may be appropriate to decay correct certain analyses of specific samples to determine radionuclide concentrations at the time of release. Even in this case, such a correction is not appropriate for batch releases. Analyses are performed prior to any release; and, the sample will be decaying at the same rate as the batch from which the sample was taken. For continuous releases, decay correcting analyses of samples obtained over a specified sampling interval must take into account the accumulation of radioactivity in the sampling medium, the decay during the sampling interval and, especially for short lived radionuclides, equilibrium or quasi-equilibrium conditions that may be achieved.

Short-lived radionuclides will tend to reach an equilibrium value in the sampling medium as a function of source input and half-life. A single decay correction to adjust for sampling interval will provide an unacceptable overestimate. Equilibrium concentrations must be considered if analyses are to be indicative of actual release quantities.

Employing  $\exp(-\lambda \Delta t)$  to adjust for radioactive decay between the end of sampling and the time of analysis is straightforward. However, to attempt to use the same term to adjust the decay during the sampling period is not proper. As a practical matter, when the half-life of a radionuclide is long relative to the sampling time and the time between sampling and analysis, i.e., minimal decay, the correction term will be near unity. In that event, the correction term is relatively unimportant.

At the other extreme, when the half-life of a radionuclide is much shorter than the sampling time or the time between the end of sampling and the analysis, the term  $\exp(-\lambda \Delta t)$  could be used to adjust for decay between the end of sampling and the analysis. However, it would not be appropriate in that case to use the same term to attempt to adjust for decay during sampling.

The relationship between the radioactivity in a sample at the end of sampling and activity concentration in the medium being sampled is somewhat more involved. To explain this in the simplest condition, assume the radionuclide concentration is constant in the medium being sampled and that the medium is sampled at a constant rate.

In the instance of water sampling, the relationship between the activity concentration in the water being sampled and the activity concentration in the water sample at the end of sampling is:

$$C_1 = C_2 \frac{\lambda t}{1 - e^{-\lambda t}} \quad (1)$$

where

$C_1$  = radionuclide concentration in the water being sampled

$C_2$  = radionuclide concentration in the water sample at the end of sampling

$t$  = duration of sampling

$\lambda$  = radionuclide decay constant

when  $\lambda t \gg 1$ ,  $C_1 \approx C_2 \lambda t$

In the separate case of sampling a radionuclide in air by filtering the air and analyzing radioactive material collected on the filter, the radionuclide of interest is concentrated. Absent diluent air in the sample being analyzed, the relation between radioactivity on the sample media and radionuclide concentration in the air being sampled is:

$$q = \frac{C_1 F}{\lambda} (1 - e^{-\lambda t}) \quad (2)$$



where

$C_i$  = radionuclide concentration in the air being sampled

$q$  = radioactivity on the sample media (assuming 100% collection efficiency)

$F$  = sampler flow rate (volume/time)

$\lambda$  = radionuclide decay constant

$t$  = duration of sampling

when  $\lambda t \gg 1$ ,  $C_i = q \lambda / F$ .

This merely recognizes that the rate of loss from the filter by radioactive decay equals the rate of collection onto the filter at equilibrium.

The NRC proposed equation appears to incorporate an adulterated way of encouraging analysis soon after the end of sampling and to encourage efficient sample concentration or radiochemical extraction. Although not rigorous, it combines both objectives in a simple and thus practical way, provided the decay correction is not extrapolated to a time earlier than the end of sampling.

A more nearly rigorous way of determining the activity concentration (or minimum detectable activity) in the medium being sampled is to assess the LLD in the sample at the time of analysis. Then the activity concentration in the medium being sampled can be calculated with the product of  $\exp(-\lambda \Delta t)$  for decay between the end of sampling and the analysis and one of the equations derived herein for the relation between the medium being sampled and the activity in the sample at the end of sampling.

However, this method is not very practical or necessary considering the types of sampling and analysis at nuclear power plants, the significant radionuclides, and the offsite potential doses. The bulk of radioactivity is released as batch releases with all sampling and analysis performed prior to release. Therefore, no decay corrections are applicable. It is in the sampling and analysis of continuous releases that the accumulation and decay of the radioactive material may need to be considered. The use of NRC's guidance for decay correction to the mid-point of the sampling period can grossly overestimate actual release quantities of short-lived radionuclides, while providing little improvement for



the quantification of the longer half-life radionuclides that are the major dose contributors.

Overall, it may be appropriate to decay correct a certain analysis to account for radionuclide decay during the sampling period. However, simple decay correction to the mid-point of sampling will grossly overestimate any short-lived radionuclides that may be detected. More consideration needs to be given by the NRC to address this problem. In any case, the use of a decay correction factor in defining a lower limit of detection is inappropriate. The LLD is a measurement of the capability of the measurement system and should not be used to try to establish a regulatory position on sampling and decay correction for quantification of releases.

## Waste Gas Decay System and Ventilation System--Operability Requirements

At Davis-Besse, the operation of the waste gas decay system is essentially continuous, similar to the routine operation of such a system at other PWRs. The system consists of a surge tank which receives the waste gases from the primary system, dual compressors (one in-service and the other in reserve), and three waste gas hold-up tanks (one in-service, one isolated for gas decay, and the third in reserve). Once the system is on-line with a waste gas decay tank receiving primary system gases for the surge tank, operation is automatic; no operator actions are required.

The operating philosophy at Davis-Besse is to essentially operate the waste gas system continuously. Not only is this philosophy prudent from an ALARA standpoint, but it is also conservative and protective from an operational standpoint. Having to periodically evaluate primary system off-gas activity levels and anticipate unexpected increases in radioactivity would be an unnecessary burden in determining needed waste gas system operation.

For the ventilation systems, the operating philosophy is similar to that for the waste gas system; operation is continuous. But for the ventilation systems, the reasons for continuous operation are even more straightforward. Areas within the plant must be provided with outside air in order to provide an inside environment suitable for continued occupancy. Without continuous ventilation system operation, heat, humidity, and airborne radioactive material levels would increase and worker occupancy would be jeopardized.

As described in the Davis-Besse Appendix I evaluation, the ventilation systems contain HEPA filters for removal of airborne radioactive particulate material prior to release to the outside environment. (As evaluated for Appendix I compliance, only the waste gas vent includes charcoal filters for removal of radioiodines) The operation of the systems can essentially be considered a passive operation. No active operational

procedures are required for normal system operation for removal of airborne radioactive material.

Davis-Besse's operating philosophy (and operating procedures) for the waste gas system and ventilation systems is a commitment in itself to the routine continuous operation of the systems. Having to commit to such a requirement (in lieu of a technical specification requirement on operation) without appropriate consideration of system down-time and plant shut-down (where operation may not be needed or feasible) is unacceptable and not in keeping with the principles of ALARA. Including special technical specifications that would impose additional procedures and periodic surveillance requirements in excess of those already established (which at present assure appropriate operation) is unnecessary and excessive.

Radiological Effluent Dose Analysis--  
Meteorology for Short Term Releases

Except for the waste gas decay tank (WGDT) releases and the containment purges releases, gaseous effluents from the Davis-Besse Station are from ventilation systems and are considered continuous releases. Most of the radioactive material in gaseous effluents is released from the WGDT. However, because of the essentially random nature of WGDT releases (i.e., no prescribed diurnal time, frequency or duration), the dose analysis of these releases is better modeled by the use of annual average meteorological conditions rather than short term meteorology. Containment purges are so infrequent that special meteorological analyses are not warranted; reasonable evaluations of off-site doses can be provided by the use of annual average meteorological conditions.



### Radiological Environmental Reporting Levels

Only the radionuclides listed in Table 3.12-2 of the proposed Radiological Effluent Technical Specifications (see note) for Davis-Besse are considered in the reporting requirements for elevated levels of radioactive material in environmental sampling media. The radionuclides listed are those that are dominant in the plant effluents and contribute essentially all of the environmental dose. Other radionuclides will be present in plant effluents, but their contribution to the calculated total environmental dose will be minor compared to the contribution of the radionuclides listed in Table 3.12-2. (see note) Even the contents of the NRC's Standard RETS reflect this position; not all pathways include reporting levels for all the radionuclides listed (e.g., no reporting levels are presented for Co-58, Co-60, or Fe-59 for the milk, airborne particulate, or vegetable pathway). This very selective identification of pathway and important radionuclides reflects the very well defined concept of significant radionuclides for each particular pathway.

Based on past experience in monitoring plant effluents and environmental sampling media, it can be stated with confidence that for the routine operation of Davis-Besse the radionuclides listed in Table 3.12-2 (see note) with applicable reporting levels by the identified pathways are the only radionuclides that need be considered when evaluating potential doses in the offsite environment. Also, even if reporting levels were included for other radionuclides, the values would be higher than those for the significant radionuclides and would have a very minor role in determining actual reporting requirements. The reporting levels for the significant radionuclides would be reached well before any identified levels of other radionuclides would even be controlling.

Note: Table 3.12-2 has been incorporated into Section 6 of the ODCM, Revision 4, Table 6-4.



Technical Basis for Eliminating Curie Inventory  
Limit of Outside Liquid Tanks

At Davis-Besse, outside liquid tanks that potentially contain radioactive material are limited to the borated water storage tanks (2 tanks @ 550,000 gallons) and the primary water storage tank (1 tank, @ 140,000 gallons).

The borated water storage tanks are part of ECCS and are of seismic design. These tanks are designed to withstand extremely adverse environmental conditions and for purposes of this evaluation can be considered rupture-proof. Also, overflow from the tanks is piped back to radwaste. For these reasons, it is considered unnecessary to impose curie inventory limits on these tanks.

The primary water storage tank is used for normal make-up and letdown to the primary system. Water contained within the PW storage tank is typically processed primary coolant or clean (non-radioactive) water. Prior to adding primary system water to the PW storage tank, the water is processed by evaporation and demineralization. This processing limits the levels of radioactivity in the tank. Past sampling and analysis of the tank has indicated only detectable levels of tritium, no other radionuclides have been identified. Also, the overflow on the PW storage tank is piped to radwaste. Therefore, due to the processing of any radioactive waste prior to addition to the PW storage tank and the piping of the overflow to radwaste, the probability of any abnormal discharges to the environment that could exceed the concentrations of 10 CFR 20, Appendix B, Table II, Column 2 at the nearest drinking water supply is extremely remote.

Because of the design of the BWST and the design and operating conditions of the PW storage tank, it is considered unnecessary to impose curie inventory limits on these tanks. Having to routinely sample and analyze for radioactivity concentration imposes an undue burden on plant personnel without providing any additional assurance of the public health and safety.

### Sampling Frequency for I-131: Significance of Power Changes and Increases in Coolant Activity Levels

The NRC guidance on effluent monitoring for I-131 (RETS Table 4.11-1, footnote c) calls for increased sampling frequency for I-131 during increases (or decreases) in reactor power level and increases in primary coolant level or noble gas effluent activity level. By system design, releases of radioactive material from plant operation are minor. Trying to identify small increases in I-131 releases that may (or may not) be associated with power changes is unnecessary. To evaluate the potential significance of increases in I-131 releases associated with power changes and the effect that may (or may not) be associated with power changes and the effect that sampling time may have on actual quantification of releases, the following example situation is evaluated.

Consider a power increase (or decrease) on the first day of a seven (7) day sampling period that leads to an increase in I-131 release rate by a factor of 10 for one (1) day. After this one day increase, the release rate returns to the steady-state condition for the remaining 6 days of the sampling period. To evaluate the amount of I-131 on the sampling cartridge as a function of sampling time and concentration, the following equation is used:

$$Q_i = \frac{C_i F (1 - e^{-\lambda_i t})}{m \lambda_i}$$

where:

$Q_i$  = quantity of activity on collection medium

$C_i$  = air concentration of radionuclide i

$\lambda_i$  = decay constant for radionuclide i

t = sample time

m = correction factor for collection efficiency

Assuming 100% collection efficiency, at the end of the one day increase the total amount of activity (I-131) on the collection cartridge is determined to be 9.54 C<sub>i</sub>F. (For this example, the steady-state I-131 concentration is designated as C<sub>i</sub> and the one day increase is 10 C<sub>i</sub>.) For the remainder of the sampling period with a concentration equal to C<sub>i</sub>, the I-131 activity on the collection cartridge is equal to 4.66 C<sub>i</sub>F.

By decaying the activity on the collection cartridge for the one day increase to the end of the sampling period and adding this quantity to 4.66 C<sub>i</sub>F, the total I-131 activity is determined to be 10.3 C<sub>i</sub>F.

\* Incorporated into ODCM, Revision 4, Table 3-3.

If this value is decay corrected to the mid-point of the sampling period in accordance with the guidance of Regulatory Guide 1.21, the I-131 activity which is used to determine the release quantity is equal to 14.0 C<sub>i</sub>F.

If a similar analysis is performed for the case of analyzing the collection cartridge at the end of the one day increase and analyzing a new cartridge at the end of 6 days sampling (constituting a 7 day sampling period), the total activity (decay corrected to mid-point of sampling periods) is determined to be 16.0 C<sub>i</sub>F.

By not analyzing the collection cartridge at the end of the one day increase, the total quantity of I-131 is underestimated by 14%. This analysis represents a somewhat worse case situation. The later into the sampling period that the increase occurs, the less the error. If the increase in release rate occurs after the mid-point of the 7 day sampling period, the actual release will be overestimated.

Over a period of time involving numerous increases and decreases in effluent level, the rules of probability dictate that the overestimations and underestimations will tend to cancel out, providing an overall closer approximation to actual releases.

Both the NRC in-plant measurement program and a study by EPRI\* have indicated that minor increases in I-131 releases may be associated with reactor power changes and the iodine spiking phenomenon. However, these studies also indicate that overall such increases are minor, not being a significant contributor to the total releases of I-131. As was concluded by the EPRI study for other PWRs, the main source of I-131 releases at Davis-Besse is associated with containment purges.

Regardless of the source, the total I-131 releases are negligible. Since initial start-up of Davis-Besse, the annual releases of I-131 have been less than 0.06 C<sub>i</sub> and calculated maximum individual doses less than 0.01 mrem. Even considering a hypothetical 14% increase for sampling periods that may include iodine spiking in the primary coolant, the effect on

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\*EPRI NP-939, "Sources of Radioiodine at Pressurized Water Reactors". Science Applications, Inc., November 1978



total releases and calculated doses is still negligible. The actual increase will be even more insignificant considering the fact that the major source of I-131 at Davis-Besse is from containment purges.

Based on a review of plant operating data and the above analysis of the I-131 release quantification as a function of concentration and sampling time, it is concluded that for Davis-Besse, a sampling frequency based on power changes and increases in primary coolant I-131 concentrations is not justified. Determining the releases (and the insignificant environmental doses of these releases) on a weekly basis is sufficient verification of the negligible impact of plant operation. Trying to "fine tune" these releases is not justified considering the manpower and material costs associated with the additional sampling and analysis.

## Condensate Demineralizer Backwash Receiving Tank - Radioactivity Control

The discharge from the condensate demineralizer backwash receiving tank is controlled on a batch-by-batch basis in lieu of continuous radioactive effluent monitoring. This method of operation has been determined to provide better control over the discharge of the backwash receiving tank, preventing any unanticipated, unevaluated releases of radioactively contaminated secondary-side clean-up resins to the on-site settling basin. Prior to discharge, the contents of the backwash receiving tank are sampled and analyzed for radioactivity. As required, radioactively contaminated resins are transferred to radwaste for processing and disposal as radioactive material.

The condensate demineralizer backwash receiving tank discharge line as originally designed included a radiation monitor. However, because of the nature of the resin-slurry mixture and the accumulation of resin beads in the monitor line, the radiation monitor has failed to provide the reliable indication of radioactivity and control as originally intended. For this reason, it has been determined that the sampling and analysis of each batch prior to discharge is needed to identify and evaluate radioactive contamination resulting from minor steam generator tube leaks (or residual radioactive material from previous leaks) that might otherwise go undetected and unevaluated by a gross radiation effluent monitor.

The condensate demineralizer backwash receiving tank discharges to an on-site settling basin. No resin discharges are made directly to the off-site environment. Therefore, even in the event of personnel error resulting in the inadvertent discharge of unacceptably radioactive, contaminated resins to the settling basin, no off-site releases would occur. All resins and radioactive material would be retained on-site within the settling basin. Appropriate follow-up measures could then be initiated to control the radioactive material and prevent any potential for releases to the off-site environment in excess of the regulatory limits.



Controlling the discharge of the condensate demineralizer backwash receiving tank on a batch-by-batch basis provides adequate control over the releases of any radioactive material to the off-site environment from this pathway. Also, the discharge is to an on-site settling basin, representing an additional passive barrier from release off-site. Even in the unlikely event of personnel error, by discharging to an on-site settling basin and its isolation from the off-site environment, the probability of unwanted, unevaluated releases of radioactive material to the off-site environment is exceedingly remote. Any additional protective measures provided by a continuous radiation monitor (for which operational performance and reliability are unlikely, based on past experience) are not considered needed.

### Lower Limit Of Detection

#### Definition And Application To Detection Capabilities For Ce-144

The lower limit of detection (LLD), as defined in the Radiological Effluent Technical Specifications (RETS) is an "... a priori (before the fact) limit representing the capabilities of a measurement system and not as a posteriori (after the fact) limit for a particular measurement." As defined by this definition applicable to the detection capability for radioactive effluent analysis, the LLD is a statistical analysis of a background spectrum and represents the detection limits for a radionuclide if it is the only radionuclide present above background. LLDs should be determined based on an analysis of a blank (or background) sample.

However, even with this definition and application of LLD, it can be increasingly difficult to achieve a predesignated LLD value for particular radionuclides as the photon abundance (i.e., decay yield) decreases. To address this problem, specific radionuclides have been identified in the RETS as being the principal radionuclides for which the required LLD must be met. For the analysis of samples of liquid radioactive effluents, an LLD of  $5 \times 10^{-7}$   $\mu\text{Ci/ml}$  is required. For the principal gamma emitters listed, all have characteristic gammas with energy levels and abundances that provide for sufficient analytical sensitivity yielding LLDs within the required value of  $5 \times 10^{-7}$   $\mu\text{Ci/ml}$  - except Ce-144. With a 10.8% abundance and an energy level of 133.5 KeV, being able to meet the LLD of  $5 \times 10^{-7}$   $\mu\text{Ci/ml}$  requires optimum conditions--conditions which cannot be repeatedly achieved for an operational radiochemistry program at Davis-Besse. The low gamma yield is a major factor; however, with an energy level which is located within the Compton continuum, the detection capability for Ce-144 even for a blank, background sample is significantly higher compared with other so-called principal gamma emitters.

The equation for LLD in the Davis-Besse RETS is:

$$\text{LLD} = \frac{4.66 S_b}{E \cdot V \cdot 2.22 \cdot Y}$$

where:

$S_b$  = the standard deviation of the background counting rate

$$= \sqrt{R/T}$$

$R$  = background counting rate

$T$  = counting time

$E$  = counting efficiency

$V$  = sample size

2.22 = conversion factor (transformations per minute per picocurie)

$Y$  = fractional chemical yield (when applicable)

By substitution of typical values in this equation, the LLDs for different principal gamma emitters can be compared. For analysis of a typical background sample at Davis-Besse, the ratio of the LLDs for Ce-144 and Co-60 is about 5.35; for Ce-144 and Mn-54 the ratio is 8.34. These large ratios are demonstrative of some of the relative difficulties in achieving an LLD of  $5 \times 10^{-7}$   $\mu\text{Ci/ml}$  for Ce-144 compared with other principal gamma emitters.

Examining the equation of LLD, two main factors can be altered in an attempt to improve the detection capability - counting time and detector efficiency. (Altering sample size is not considered realistic since larger samples would pose operational and standard calibration problems. It can also be shown that increasing sample volume does not strongly influence efficiency for counting on contact with the detector face due in part to sample self-shielding and decreased relative efficiency for the increased volume).

LLD improves at best as the square root of the counting time. Therefore, increasing the counting time from 2000 seconds to 5000 seconds would only provide a 1.6 reduction in LLD. A 5000 second count is considered to be a reasonable maximum for radioactive effluent analysis. Having to extend to longer counting times would introduce a potential operational delay without commensurate improvement in detection capability.

An improvement in the efficiency is negated in part by the corresponding increase in background count rate. A comparison of 5 GeLi detectors with relative efficiencies ranging from 7.2% to 22% was performed at the University of Michigan\*. For a 500 ml sample contact with the detectors, the 15% relative efficiency detector demonstrated the highest photopeak efficiency in this energy than did the 21% and 22% relative efficiency detectors. Some unexplainable differences may be due to inherent manufacturers specifications; however, a valid conclusion is that increasing the detector efficiency provides little if any improvement in detection capability, especially in the low energy range (<200 KeV).

Therefore, the analysis of effluent samples at Davis-Besse with a 10% relative efficiency GeLi and a 5000 second counting time provides a detection system that is not only practical for an operational radio-chemistry program but can also be considered as representative of state-of-the-art for routine, general purpose radionuclide detection. Since the required LLD of  $5 \times 10^{-7}$   $\mu\text{Ci}/\text{ml}$  can not be met on a routine basis for Ce-144, therefore the LLD Ce-114 will be  $2.0 \times 10^{-6}$   $\mu\text{Ci}/\text{ml}$  (Table 4.11-1\*\* footnote b.).

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\* D. M. Minnema, C. G. Hudson and J. D. Jones. "A Comparison of Ge(Li) Detectors with Different Efficiencies for Low-Level General Purpose Counting"; University of Michigan, 1978.

\*\* Incorporated into ODCM, Revision 4, Table 3-3.