APPENDIX C

CATAWBA NUCLEAR STATION

SITE SPECIFIC INFORMATION

APPENDIX C - TABLE OF CONTENTS

C1.0 CATAWBA NUCLEAR STATION RADWASTE SYSTEMS

C1.1 LIQUID RADWASTE PROCESSING

The liquid radwaste system at Catawba Nuclear Station (CNS) is used to collect and treat fluid chemical and radiochemical by-products of unit operation. The system produces efflucents which can be reused in the plant or discharged in small, dilute quantities to the environment. The means of treatment vary with waste type and desired product in the various systems:

A) Filtration - All waste sources are filtered during processing. In some cases, such as the Floor Drain Tank (FDT) Subsystem of the Liquid Waste (WL) System, filtration may be the only treatment required.

B) Adsorption - Adsorption of halides and organic chemicals by activated charcoal (Carbon Filter) is used primarily in treating waste in the Laundry and Hot Shower Tank (LHST) Subsystem of the WL System. FDT waste may also be treated by this method.

C) Ion Exchange - Ion exchange is used to remove radioactive cations from solution, as in the case of either LHST or FDT waste in the WL System after removal of organics by carbon filtration (adsorption). Ion exchange is also used in removing both cations (cobalt, manganese) and anions (chloride, fluoride) from evaporator distillates in order to purify the distillates for reuse as makeup water. Distillate from the Waste Evaporator in the WL System and the Boron Recycle Evaporator in the Boron Recycle System (NB) can be treated by this method, as well as FDT, LHST waste, and letdown.

D) Gas Stripping - Removal of gaseous radioactive fission products is accomplished in both the WL Evaporator and the NB Evaporator.

E) Distillation - Production of pure water from the waste by boiling it away from the contaminated solution which originally contained it is accomplished by both evaporators. Proper control of the process will yield water which can be reused for makeup. Polishing of this product can be achieved by ion exchange as pointed out above.

F) Concentration - In both the WL and NB Evaporators, dissolved chemicals are concentrated in the lower shell as water is boiled away. In the case of the WL Evaporator, the volume of water containing waste chemicals and radioactive cations is reduced so that the waste may be more easily and cheaply solidified and shipped for burial. In the NB Evaporator, the dilute boron is concentrated to 4% so that it may be reused for makeup to the reactor coolant system.

Figure C1.0-1 is a schematic representation of the liquid radwaste system at Catawba.

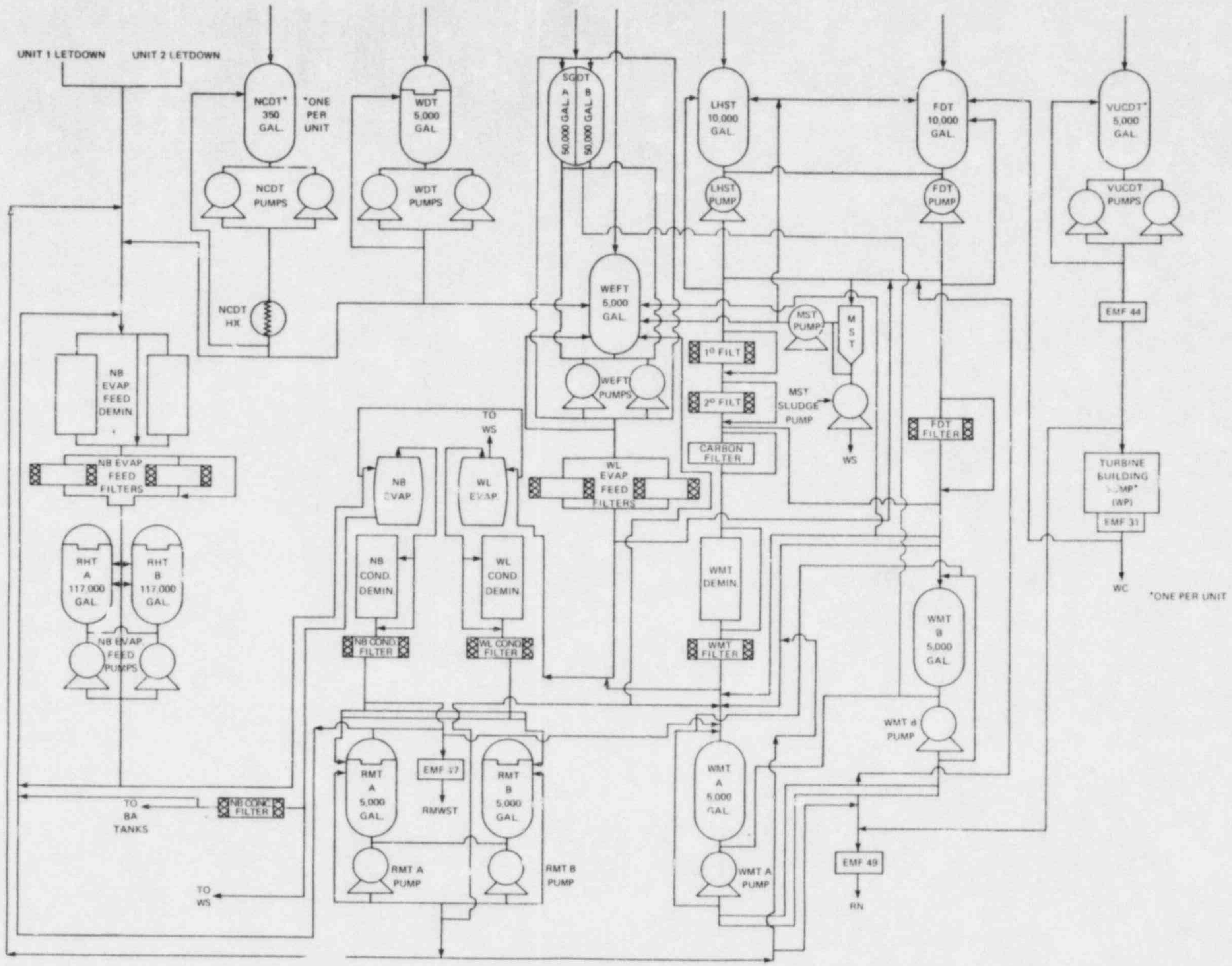


FIGURE C1.0.1
CATAWBA NUCLEAR STATION
LIQUID RADWASTE SYSTEM
DATE COMPILED 8/3/83

ABBREVIATIONS

Systems:

- KC - Component Cooling
- NB - Boron Recycle
- NC - Reactor Coolant
- ND - Residual Heat Removal
- NI - Safety Injection
- NR - Boron Thermal Regeneration
- NV - Chemical Volume and Control
- RL - Low Pressure Service Water
- RN - Nuclear Service Water System
- WC - Conventional Waste Water Treatment
- WG - Waste Gas
- WL - Liquid Waste Recycle
- WS - Nuclear Solid Waste Disposal

Terms:

- BOL - Beginning of Core Life
- BTRS - Boron Thermal Regeneration System
- CDT - Chemical Drain Tank
- ECST - Evaporator Concentrates Storage Tank
- EOL - End of Core Life
- FDT - Floor Drain Tank
- FWST - Fueling Water Storage Tank (formerly Refueling Water Storage Tank)
- LHST - Laundry and Hot Shower Tank
- MST - Mixing and Settling Tank
- NCDT - Reactor Coolant Drain Tank
- RBT - Resin Batching Tank
- RHT - Recycle Holdup Tank
- RMT - Recycle Monitor Tank
- RMWST - Reactor Makeup Water Storage Tank

TABLE C1.0-1

(2 of 2)

ABBREVIATIONS

Terms: (continued)

SRST - Spent Resin Storage Tank

VUCDT - Ventilation Unit Condensate Drain Tank

WDT - Waste Drain Tank

WEFT - Waste Evaporator Feed Tank

WMT - Waste Monitor Tank

C1.2 GASEOUS RADWASTE SYSTEMS

The gaseous waste disposal system for Catawba is designed with the capability of processing the fission-product gases from contaminated reactor coolant fluids resulting from operation. The system shown schematically in Fig. C1.0-2 is designed to allow for the retention, through the plant lifetime, of all the gaseous fission products to be discharged from the reactor coolant system to the chemical and volume control system or the boron recycle system, to limit the need for intentional discharge of radioactive gases from the waste gas holdup tanks. Thus, the only unavoidable sources of low-level radioactive gaseous discharge to the environment will be from periodic purging operations of the containment, from the auxiliary building ventilation system, and through the secondary system air ejector. With respect to the former, the potential contamination is expected to arise from uncollectable reactor coolant leakage. With respect to the air ejector, the potential source of contamination will be from leakage of the reactor coolant to the secondary system through defects in steam generator tubes. The gaseous waste disposal system includes two waste gas compressors, two catalytic hydrogen recombiners, six gas decay storage tanks for use during normal power generation, and two gas decay storage tanks for use during shutdown and startup operations.

C1.2.1 Gas Collection System

The gas collection system combines the waste hydrogen and fission gases from the volume control tanks and that from the boron recycle gas stripper evaporator produced during normal operation with the gas collected during the shutdown degasification (high percentage of nitrogen) and will cycle it through the catalytic recombiners to convert all the hydrogen to water. After the water vapor is removed, the resulting gas stream will be transferred from the recombiner into the gas decay tanks, where the accumulated activity may be contained in six approximately equal parts. From the decay tanks the gas will flow back to the compressor suction to complete the loop circuit.

C1.2.2 Containment and Auxiliary Building Ventilation

Nonrecyclable reactor coolant leakage occurring either inside the containment or inside the auxiliary building will generate gaseous activity. Gases resulting from leakage inside the containment will be contained until the containment air is released through the VQ or VP system. The containment atmosphere will be discharged through a charcoal adsorber and a particulate filter prior to release to the atmosphere.

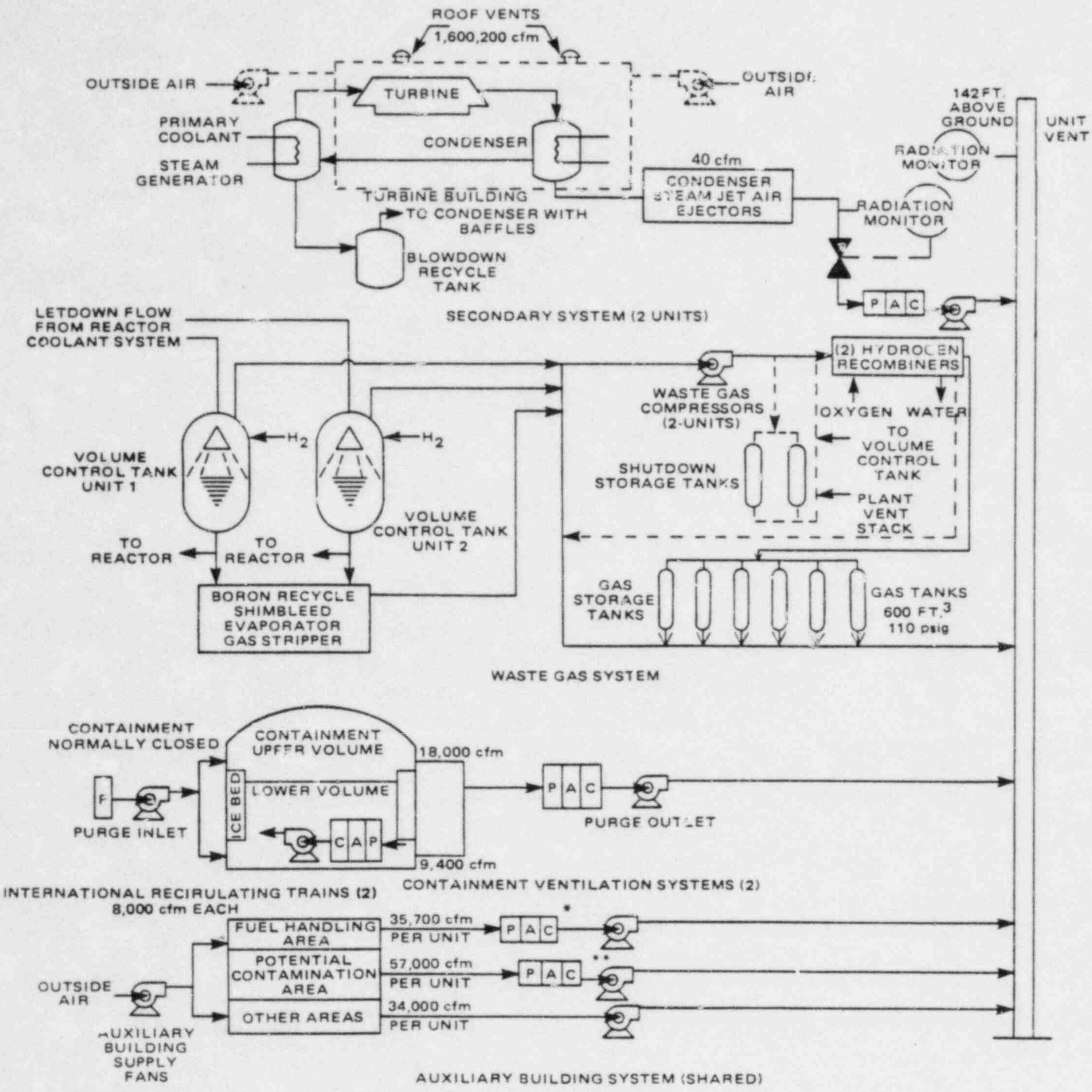
Gases resulting from leakage inside the auxiliary building are released, without further decay, to the atmosphere via the auxiliary building ventilation system. The ventilation exhaust from potentially contaminated areas in the auxiliary building is normally unfiltered. However, on a radiation monitor alarm, the exhaust is passed through charcoal adsorbers to reduce releases to the atmosphere.

C1.2.3 Secondary Systems

Normally, condensate flow and steam generator blowdown will go parallel through 4 of the 5 condensate polishing demineralizers to remove activity and harmful ions from the water. Noncondensable gases will be taken from the

secondary system by the condenser steam air ejector and are passed through a radiation monitor to the unit vent.

Figure C1.0-2 is a schematic representation of the gaseous radwaste system at Catawba.



* FUEL HANDLING AREA IS NORMALLY UNFILTERED. UPON A RADIATION ALARM BY EMF – 42, THE EXHAUST WILL BE DIVERTED TO THE FILTERED MODE.

** POTENTIALLY CONTAMINATED AREAS OF THE AUXILIARY BUILDING ARE NORMALLY UNFILTERED. UPON A RADIATION ALARM BY EMF – 41. THE EXHAUST WILL BE DIVERTED TO THE FILTERED MODE.

FIGURE C1.0-2
CATAWBA NUCLEAR STATION
GASEOUS RADWASTE SYSTEM
DATE COMPILED 8/3/83

C2.0 RELEASE RATE CALCULATION

Generic release rate calculations are presented in Section 1.0; these calculations will be used to calculate release rates for Catawba Nuclear Station.

C2.1 LIQUID RELEASE RATE CALCULATIONS

There are two potential release points at Catawba. They are as follows:

1. Liquid Waste Effluent Discharge Line
2. Conventional Waste Water Treatment System Effluent Line

C2.1.1 Liquid Waste Effluent Discharge Line

There are three low-pressure service water pumps with a minimum flow rate of 16,500 gpm each and four nuclear service water pumps with a minimum flow rate of 9,000 gpm each which provide the required dilution water needed for a release. The flow rate monitor has a variable setpoint which terminates the release by closing the isolation valve, 1 WL124 should the dilution flow fall below the setpoint. The following equation shall be used to calculate a discharge flow, in gpm.

$$f \leq F_{RL} \div \left[\sigma \sum_{i=1}^{n} \frac{C_i}{MPC_i} \right]$$

where:

f = the undiluted effluent flow, in gpm.

F_{RL} = actual low pressure service water flowrate, in gpm, from the sum of the flowrate monitors located in the Control Room.

σ = the recirculation factor at equilibrim (dimensionless), 1.027.

$$\sigma = 1 + \frac{Q_R}{Q_H} = 1 + \frac{120 \text{ cfs}}{4400 \text{ cfs}} = 1.027$$

where:

Q_R = average dilution flow (120 Cfs)

Q_H = average flow past Wylie Dam (4400 cfs)

C_i = the concentration of radionuclide, i, in undiluted effluent as determined by laboratory analyses, in µCi/ml.

MPC_i = the concentration of radionuclide, i, from 10CFR20, Appendix B, Table II, Column 2. If radionuclide, i, is a dissolved noble gas, the MPC_i = 2.0E-04 µCi/ml.

C2.1.2 Conventional Waste Water Treatment System Effluent Line

The conventional waste water treatment system effluent is normally considered nonradioactive; that is, it is unlikely the effluent will contain measurable activity above background. It is assumed that no activity is present in the effluent until indicated by radiation monitoring measurements and by periodic analyses of the composite sample collected on that line. These three sources of water that are normally discharged into the conventional waste water system will have their flow diverted if they become radioactive.

a. Containment Ventilation Unit Condensate Effluent Line

Normally the containment ventilation unit condensate effluent line would discharge into the Turbine Building sump, but if radiation is detected above background. the discharge will be terminated and an alarm actuated. The containment ventilation unit condensate tank will then be recirculated, sampled and then discharged through the liquid waste effluent line and monitored or processed thru the WL system.

b. Clean Area Floor Drain Pump Sumps

Normally the discharge line coming from these sumps will discharge into the Turbine Building sump, but if radiation is detected above background, the discharge flow will automatically be routed to the floor drain tank for processing and later be discharged through the liquid waste effluent line.

c. Turbine Building Sump Discharge Line

Normally the discharge from the Turbine Building sump will go into the conventional waste water treatment system, but if radiation is detected above background, the sump pumps A, B, and C will stop and an alarm actuated. The Turbine Building sump discharge line can either be routed to the floor drain tank for processing or routed directly to the liquid waste effluent discharge line.

C2.2 GASEOUS RELEASE RATE CALCULATIONS

The unit vent is the release point for waste gas decay tanks, containment air releases, the condenser air ejector, and auxiliary building ventilation. The condenser air ejector effluent is normally considered nonradioactive; that is, it is unlikely the effluent will contain measurable activity above background. It is assumed that no activity is present in the effluent until indicated by radiation monitoring measurements and/or by analyses of periodic samples collected on that line. Radiation monitoring alarm/trip setpoints in conjunction with administrative controls assure that release limits are not exceeded; see section C.3.0 on radiation monitoring setpoints.

The following calculations, when solved for flowrate, are the release rates for noble gases and for radioiodines, particulates and other radionuclides with half-lives greater than 8 days; the most conservative of release rates calculated in C2.2.1 and C2.2.2 shall control the release rate for a single release point.

C2.2.1 Noble Gases

$$\sum_i K_i \; [(\overline{X/Q})\tilde{Q}_i] < 500 \text{ mrem/yr, and}$$

$$\sum_i (L_i + 1.1\, M_i) \; [(\overline{X/Q})\tilde{Q}_i] < 3000 \text{ mrem/yr}$$

where the terms are defined below.

C2.2.2 Radioiodines, Particulates, and Other Radionuclides With T 1/2 > 8 Days

$$\sum_i P_i \; [W \tilde{Q}_i] < 1500 \text{ mrem/yr}$$

where:

K_i = The total body dose factor due to gamma emissions for each identified noble gas radionuclide, in mrem/yr per $\mu Ci/m^3$ from Table 1.2-1.

L_i = The skin dose factor due to beta emissions for each identified noble gas radionuclide, in mrem/yr per $\mu Ci/m^3$ from Table 1.2-1.

M_i = The air dose factor due to gamma emissions for each identified noble gas radionuclide, in mrad/yr per $\mu Ci/m^3$ from Table 1.2-1 (unit conversion constant of 1.1 mrem/mrad converts air dose to skin dose).

P_i = The dose parameter for radionuclides other than noble gases for the inhalation pathway, in mrem/yr per $\mu Ci/m^3$ and for the food and ground plane pathways in $m^2 \cdot$(mrem/yr) per μCi/sec from Table 1.2-2. The dose factors are based on the critical individual organ and most restrictive age group (child or infant).

$\tilde{Q}_i$ = The release rate of radionuclides, i, in gaseous effluent from all release points at the site, in μCi/sec.

$(\overline{X/Q})$ = 2.60E-07 sec/m^3. The highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary.

W = The highest calculated annual average dispersion parameter for estimating the dose to an individual at the controlling location:

W = 1.00E-07 sec/m^3, for the inhalation pathway. The location is the unrestricted area in the S sector.

W = 1.70E-09 meter^{-2}, for the food and ground plane pathways. The location is the unrestricted area boundary in the S sector (nearest residence, and vegetable garden).

$$\tilde{Q}_i = k_1 C_i f \div k_2 = 4.72E{+}2 C_i f$$

where:

C_i = the concentration of radionuclide, i, in undiluted gaseous effluent, in μCi/ml.

f = the undiluted effluent flow, in cfm

k_1 = conversion factor, 2.83E4 ml/ft^3

k_2 = conversion factor, 6E1 sec/min

C3.0 RADIATION MONITOR SETPOINTS

Using the generic calculations presented in Section 2.0, radiation monitoring setpoints are calculated for monitoring as required by the Technical Specifications.

All radiation monitors for Catawba are off-line. These monitors alarm on low flow; the minimum flow alarm level for the liquid monitors is 2 gallons per minute and for the gas monitors is 6 standard cubic feet per minute. These monitors measure the activity in the liquid or gas volume exposed to the detector and are independent of flow rate if a minimum flow rate is assured.

Radiation monitoring setpoints calculated in the following sections are expressed in activity concentrations; in reality the monitor readout is in counts per minute. The relationship between concentration and counts per minute is established by a station procedure using the following relationship:

$$c = \frac{r}{2.22 \times 10^6 e V}$$

where:

c = the gross activity, in μCi/ml
r = the count rate, in cpm
2.22×10^6 = the disintegration per minute per μCi
e = the counting efficiency, cpm/dpm
V = the volume of fluid exposed to the detector, in ml.

C3.1 LIQUID RADIATION MONITORS

C3.1.1 Waste Liquid Effluent Line

As described in Section C2.1.1 on release rate calculations for the waste liquid effluent, the release is controlled by limiting the flow rate of effluent from the station. Although the release rate is flow rate controlled, the radiation monitor setpoint shall be set to terminate the release if the effluent activity should exceed that determined by laboratory analyses and that used to calculate the release rate. This setpoint is:

$$c \leq \frac{\text{MPC} \times F}{\sigma f} \leq 2.48\text{E-05 } \mu\text{Ci/ml}$$

where:

c = the gross activity in undiluted effluent, in μCi/ml

f = the flow from the tank may vary from 0-100 gpm but, for this calculation, is assumed to be 100 gpm.

MPC = 1.0E-07 μCi/ml, the MPC for an unidentified mixture

σ = 1.027 (See Section C2.1.1)

F = the dilution flow may vary as described in section C2.1.1, but is conservatively estimated at 25,500 gpm, the minimum flow available.

C3.1.2 Containment Ventilation Unit Condensate Effluent Line

As described in Section C2.1.2, on release rate calculations for the containment ventilation unit condensate effluent, it is possible but unlikely that the effluent will contain measurable activity above background. It is assumed that no activity is present in the effluent until indicated by radiation monitoring. Since the tank contents are discharged automatically, a maximum tank concentration, which also is the radiation monitor setpoint, is calculated to assure that release limits are not exceeded. The monitor setpoint and maximum tank concentration is:

$$c \leq \frac{MPC \times F}{\sigma f} \leq 4.97E\text{-}05\ \mu Ci/ml$$

where:

c = the gross activity in undiluted effluent, in μCi/ml

f = the flow from the tank may vary from 0-50 gpm but, for this calculation, is assumed to be 50 gpm

MPC = 1.0E-07 μCi/ml, the MPC for an unidentified mixture

σ = 1.027 (see Section C2.1.1)

F = the dilution flow may vary as described in C2.1.1, but is conservatively estimated at 25,500 gpm, the minimum flow available.

C3.1.3 Feedwater Pump Sumps

As described in Section C2.1.2, release rate calculations for the feedwater pump sumps and floor drain tank effluents, it is possible but unlikely that the effluents will contain measurable activity above background. It is assumed that no activity is present in the effluent until indicated by radiation monitoring. Since the sumps are discharged automatically, a maximum sump concentration, which is also the radiation monitor setpoint, is calculated as follows to assure that release limits are not exceeded:

$$c \leq \frac{MPC \times F}{\sigma f} \leq 2.48E\text{-}05\ \mu Ci/ml$$

where:

c = the gross activity in undiluted effluent, in μCi/ml

f = the flow from the sumps may vary from 0-100 gpm but, is assumed to be 100 gpm.

MPC = 1.0E-07 μCi/ml, the MPC for an unidentified mixture

σ = 1.027 (see Section C2.1.1)

f = the dilution flow may vary as described in Section C2.1.1, but is conservatively estimated at 25,500 gpm, the minimum flow available.

C3.1.4 Turbine Building Sump Discharge Line

As described in Section C2.1.2, release rate calculations for the turbine building sumps, it is possible but unlikely that the effluents will contain measurable activity above background. It is assumed that no activity is present in the effluent until indicated by radiation monitoring Since the sumps are discharged automatically, a maximum sump concentration, which is also the radiation monitor setpoint, is calculated as follows to assure that release limits are not exceeded:

$$c \leq \frac{MPC \times F}{\sigma f} \leq 1.84E\text{-}06,\ \mu Ci/ml$$

where:

c = the gross activity in undiluted effluent, in μCi/ml

f = the flow rate of the sumps may vary from 0-1350 gpm, but for this calculation, is assumed to be 1350 gpm.

MPC = 1.0E-07 μCi/ml, the MPC for an unidentified mixture

σ = 1.027 (see Section C2.1.1)

F = the dilution flow may vary as described in Section C2.1.1, but is conservatively estimated at 25,500 gpm, the minimum flow available.

C3.2 GAS MONITORS

The following equation shall be used to calculate noble gas radiation monitor setpoints based on Xe-133:

$$K(X/Q)\tilde{Q}_i < 500 \qquad \text{(see Section C2.2.1)}$$

$$\hat{Q}_i = 4.72E{+}02\ C_i f \quad \text{(see Section C2.2.2)}$$

$$C_i < 1.39E{+}04/f$$

where:

C_i = the gross activity in undiluted effluent, in μCi/ml

f = the flow from the tank or building and varies for various release sources, in cfm

K = from Table 1.2-1 for Xe-133, 2.94E+2 mrem/yr per $\mu Ci/m^3$

(X/Q) = 2.60E-07 sec/m^3, the highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary for long term releases.

As stated in Section C2.2, the unit vent is the release point for the containment purge ventilation system, the containment air release and addition system, the condenser air ejector, and auxiliary building ventilation. Since all of these releases are through the unit vent, the radiation monitor on the unit vent may be used to assure that station release limits are not exceeded.

For releases from the containment purge ventilation system, the radiation monitor setpoint may be:

$$C_i < 1.39E{+}04/f = 7.28E{-}02$$

where:

f = 151,000 cfm (auxiliary building ventilation) + 40,000 cfm (containment purge) = 191,000 cfm

For release from the containment air release and addition system, the waste gas decay tanks, the condenser air ejectors, and the auxiliary building ventilation, the radiation monitor setpoint may be:

$$C_i < 1.39E{+}04/f = 9.21E{-}02$$

where:

f = 151,000 cfm (auxiliary building ventilation)

C4.0 DOSE CALCULATIONS

C4.1 FREQUENCY OF CALCULATIONS

Dose contributions to the maximum exposed individual shall be calculated every 31 days, quarterly, semiannually, and annually (as required by Technical Specifications) using the methodology in the generic information sections. This methodology shall also be used for any special reports. Dose projections shall be performed using simplified estimates. Fuel cycle dose calculations shall be performed annually or as required by special reports. Dose contributions may be calculated using the methodology in the appropriate generic information sections.

C4.2 DOSE MODELS FOR MAXIMUM EXPOSED INDIVIDUAL

C4.2.1 Liquid Effluents

For dose contributions from liquid radioactive releases, one of the two following cases will apply:

1. If the radionuclides Co-58 and/or Co-60 have been detected and Cs-134 and/or Cs-137 have not been detected (i.e., plants without any fuel failure) dose calculations indicate that the maximum exposed individual would be a child who consumed fish caught in the discharge canal and who drank water from the nearest "downstream" potable water intake. The dose from these two radionuclides has been calculated to be 27% of that individual's total body dose.

2. If the radionuclides Cs-134 and/or Cs-137 have been detected, dose calculations indicate that the maximum exposed individual would be an adult who consumed fish caught in the discharge canal and who drank water from the nearest "downstream" potable water intake. The dose from these two radionuclides has been calculated to be 90% of that individual's total body dose.

C4.2.2 Gaseous Effluents

C4.2.2.1 Noble Gases

For dose contributions from exposure to beta and gamma radiation from noble gases, it is assumed that the maximum exposed individual is an adult on the site boundary in each meteorological sectors.

C4.2.2.2 Radioiodines, Particulates, and Other Radionuclides T 1/2 $>$ 8 days

For dose contributions from radioiodines, particulates and other radionuclides; it is assumed that the maximum exposed individual is an infant who breathes the air and consumes milk from the nearest goat or cow in each meteorological sector.

C4.3 SIMPLIFIED DOSE ESTIMATE

C4.3.1 Liquid Effluents

For dose estimates, two simplified calculations using the assumptions presented in Section C4.2.1 and source terms presented in the FSAR are presented. Once operational source term data is available, this information shall be used to revise these calculations, if necessary.

Case 1 - No Cs-134 or Cs-137 present in effluent.

$$D_{WB} = 2.36E{+}03 \sum_{\ell=1}^{m} (F_\ell)(T_\ell)\ (C_{Co\text{-}60} + 0.35\ C_{Co\text{-}58})$$

where:

$2.36E{+}03 = 1.14E{+}05\ (U_{aw}/D_w + U_{af}\ BF_i)\ DF_{ait}\ (3.70)$

where:

$1.14E{+}05 = 10^6 pCi/\mu Ci \times 10^3 ml/kg \div 8760\ hr/yr$

U_{aw} = 510 kg/yr, child water consumption

D_w = 37.7, dilution factor from the near field area to the nearest potable water intake.

U_{af} = 6.9 kg/yr, child fish consumption

BF_i = 5.0E+01, bioaccumulation factor for Cobalt (Table 3.1-1)

DF_{ait} = 1.56E-05, child, total body, ingestion dose factor for Co-60 (Table 3.1-4)

3.70 = factor derived from assumption that 27% of dose is from Co-58 and Co-60 or 100% ÷ 27% = 3.70

And where:

$$F_\ell = \frac{f\sigma}{F + f}$$

where:

f = liquid radwaste flow, in gpm

σ = recirculation factor at equilibrium, 1.027

F = dilution flow, in gpm

And where:

T_ℓ = The length of time, in hours, over which $C_{Co\text{-}58}$, $C_{Co\text{-}60}$, and F_ℓ are averaged.

$C_{Co\text{-}58}$ = the average concentration of Co-58 in undiluted effluent, in μCi/ml, during the time period considered.

C_{Co-60} = the average concentration of Co-60 in undiluted effluent, in μCi/ml, during the time period considered.

0.35 = The ratio of the child total body ingestion dose factors for Co-58 and Co-60 or 5.51E-06 ÷ 1.56E-05 - Table 3.1-4.

Case 2 - Cs-134 and/or Cs-137 present in effluent.

$$D_{WB} = 6.38E{+}05 \sum_{\ell=1}^{m} (F_\ell)(T_\ell)\ (C_{Cs-134} + 0.59\ C_{Cs-137})$$

where:

$6.38E{+}05 = 1.14E{+}05\ (U_{aw}/D_w + U_{af}\ BF_i)\ DF_{ai\tau}\ (1.10)$

where:

$1.14E{+}05 = 10^6 pCi/\mu Ci \times 10^3 ml/kg \div 8760\ hr/yr$

U_{aw} = 730 kg/yr, adult water consumption

D_w = 37.7, dilution factor from the near field area to the nearest potable water intake.

U_{af} = 21 kg/yr, adult fish consumption

BF_i = 2.00E+03, bioaccumulation factor for Cesium (Table 3.1-1)

$DF_{ai\tau}$ = 1.21E-04, adult, total body, ingestion dose factor for Cs-134 (Table 3.1-2)

1.10 = factor derived from the assumption that 90% of dose is from Cs-134 and Cs-137 or 100% ÷ 90% = 1.10

And where:

$$F_\ell = \frac{f\sigma}{F + f}$$

where:

f = liquid radwaste flow, in gpm

σ = recirculation factor at equilibrium, 1.027

F = dilution flow, in gpm

And where:

T_ℓ = The length of time, in hours, over which C_{Cs-134}, C_{Cs-137}, and F_ℓ are averaged.

C_{Cs-134} = the average concentration of Cs-134 in undiluted effluent, in μCi/ml, during the time period considered.

C_{Cs-137} = the average concentration of Cs-137 in undiluted effluent, in μCi/ml, during the time period considered.

0.59 = The ratio of the adult total body ingestion dose factors for Cs-134 and Cs-137 or 7.14E-05 ÷ 1.21E-04 = 0.59

C4.3.2 Gaseous Effluents

Meteorological data is provided in Tables C4.0-1 and C4.0-2.

C4.3.2.1 Noble Gases

For dose estimates, simplified dose estimates using the assumptions in C4.2.2.1 and source terms in the FSAR are presented below. Once operational source term data is available, this information shall be used to revise these calculations, if necessary. These calculations further assume that the annual average dispersion parameter is used and that Xenon-133 contributes 45% of the dose.

$$D_\gamma = 2.91E\text{-}12\ [\tilde{Q}]_{Xe-133}\ (2.22)$$

$$D_\beta = 8.65E\text{-}12\ [\tilde{Q}]_{Xe-133}\ (2.22)$$

where:

2.91E-12 = (3.17E-8)(353) ($\overline{X/Q}$), derived from equation presented in Section 3.1.2.1.

8.65E-12 = (3.17E-08) (1050) ($\overline{X/Q}$), derived from equation presented in Section 3.1.2.1.

$\overline{X/Q}$ = 2.60E-07 sec/m^3, as defined in Section C2.2.2

$[\tilde{Q}]_{Xe-133}$ = the total Xenon-133 activity released in μCi

2.22 = factor derived from the assumption that 45% of the dose is contributed by Xe-133.

C4.3.2.2 Radioiodines, Particulates, and Other Radionuclides with T 1/2 > 8 days

For dose estimates, simplified dose estimates using the assumptions in C4.2.2.2 and source terms in the FSAR are presented below. Once operational source term data is available, this information shall be used to revise these calculations, if necessary. These calculations further assume that the annual average dispersion/deposition parameter is used and that 95% of the dose is from Iodine-131 concentrated in goat's milk. The simplified dose estimate to the thyroid of an infant is:

$$D = 2.00E\text{+}04\ w\ (\tilde{Q})_{I-131}\ (1.05)$$

where:

w = 2.40E-10 = $\overline{D/Q}$ for food and ground plane pathway, in m^{-2} from Table C4.0-2 for location of nearest real goat (NW sector at 2.5 miles).

$(\tilde{Q})_{I-131}$ = the total Iodine-131 activity released in μCi.

2.00E+04 = (3.17E-08)(R_i^C [$\overline{D/Q}$]) with the appropriate substitutions for goat's milk in the grass-cow-milk-pathway factor, R_i^C [$\overline{D/Q}$] for Iodine-131. See Section 3.1.2.2.

1.05 = factor derived from the assumption that 95% of the dose is contributed by I-131.

C4.3 FUEL CYCLE CALCULATIONS

These calculations shall be performed using models presented in Section 3.3.

TABLE C4.0-1

(1 of 1)

CATAWBA NUCLEAR STATION

DISPERSION PARAMETER ($\overline{X/Q}$) FOR LONG TERM RELEASES > 500 HR/YR OR > 125 HR/QTR

Sector	Distance to the control location, (miles) 0.5	1.0	1.5	2.0	2.5	3.0	3.5	4.0	4.5	5.0
N	9.8E-8	9.5E-8	9.2E-8	7.7E-8	6.5E-8	5.8E-8	4.8E-8	4.4E-8	3.9E-8	3.5E-8
NNE	2.5E-7	2.6E-7	2.5E-7	2.1E-7	1.7E-7	1.5E-7	1.3E-7	1.2E-7	9.8E-8	9.1E-8
NE	2.3E-7	1.9E-7	1.6E-7	1.3E-7	1.1E-7	9.3E-8	7.9E-8	7.0E-8	6.2E-8	5.7E-8
ENE	1.2E-7	1.1E-7	9.5E-8	7.8E-8	6.4E-8	5.8E-8	4.8E-8	4.4E-8	3.9E-8	3.5E-8
E	8.3E-8	6.6E-8	6.6E-8	5.6E-8	4.6E-8	4.2E-8	3.5E-8	3.3E-8	2.8E-8	2.6E-8
ESE	6.8E-8	6.0E-8	5.1E-8	4.3E-8	3.5E-8	3.3E-8	2.7E-8	2.4E-8	2.2E-8	2.1E-8
SE	6.2E-8	5.3E-8	4.7E-8	3.8E-8	3.2E-8	2.9E-8	2.5E-8	2.4E-8	2.0E-8	1.9E-8
SSE	8.2E-8	8.3E-8	7.5E-8	5.1E-8	5.1E-8	4.7E-8	3.8E-8	3.7E-8	3.0E-8	2.8E-8
S	1.0E-7	1.1E-7	1.1E-7	8.6E-8	7.2E-8	6.2E-8	5.3E-8	4.9E-8	4.2E-8	3.7E-8
SSW	1.5E-7	1.4E-7	1.3E-7	1.1E-7	8.7E-8	7.6E-8	6.3E-8	5.8E-8	5.0E-8	3.9E-8
SW	1.7E-7	1.7E-7	1.6E-7	1.3E-7	1.0E-7	8.5E-8	7.1E-8	6.3E-8	5.5E-8	4.9E-8
WSW	9.0E-8	9.2E-8	9.1E-8	7.6E-8	6.0E-8	5.1E-8	4.4E-8	3.9E-8	3.4E-8	3.1E-8
W	8.2E-8	8.4E-8	7.9E-8	6.7E-8	5.2E-8	4.4E-8	3.7E-8	3.3E-8	2.9E-8	2.5E-8
WNW	7.0E-8	6.6E-8	6.1E-8	5.0E-8	4.1E-8	3.4E-8	3.0E-8	2.6E-8	2.4E-8	2.1E-8
NW	8.0E-8	6.8E-8	6.0E-8	5.0E-8	4.1E-8	3.4E-8	3.1E-8	2.7E-8	2.5E-8	2.2E-8
NNW	1.1E-7	9.3E-8	8.3E-8	7.0E-8	5.9E-8	4.8E-8	4.5E-8	3.8E-8	3.6E-8	3.2E-8

TABLE C4.0-2

(1 of 1)

CATAWBA NUCLEAR STATION

DIPERSION PARAMETER ($\overline{D/Q}$) FOR LONG TERM RELEASES > 500 HR/YR OR > 125 HR/QTR

	Distance to the control location, (miles)									
Sector	0.5	1.0	1.5	2.0	2.5	3.0	3.5	4.0	4.5	5.0
N	1.4E-9	6.3E-10	4.8E-10	2.7E-10	1.9E-10	1.6E-10	1.4E-10	1.2E-10	9.5E-11	7.9E-11
NNE	5.1E-9	2.1E-9	1.5E-9	8.9E-10	7.2E-10	5.4E-10	4.8E-10	3.8E-10	3.5E-10	3.0E-10
NE	6.5E-9	2.6E-9	1.8E-9	1.1E-9	7.5E-10	6.0E-10	4.8E-10	4.1E-10	3.3E-10	3.0E-10
ENE	2.3E-9	8.5E-10	5.3E-10	3.5E-10	2.6E-10	2.1E-10	1.9E-10	1.5E-10	1.3E-10	1.1E-10
E	2.2E-9	8.1E-10	6.1E-10	3.0E-10	2.1E-10	1.7E-10	1.5E-10	1.1E-10	9.3E-11	8.0E-11
ESE	1.7E-9	5.4E-10	3.3E-10	2.0E-10	1.4E-10	1.1E-10	9.7E-11	7.3E-11	6.1E-11	5.2E-11
SE	1.6E-9	5.9E-10	3.6E-10	2.4E-10	1.6E-10	1.2E-10	1.0E-10	8.0E-11	6.9E-11	5.6E-11
SSE	1.4E-9	5.9E-10	3.9E-10	2.3E-10	1.7E-10	1.4E-10	1.2E-10	9.2E-11	8.0E-11	6.6E-11
S	1.7E-9	8.3E-10	6.6E-10	3.4E-10	2.7E-10	2.0E-10	1.9E-10	1.5E-10	1.4E-10	1.1E-10
SSW	2.5E-9	9.9E-10	7.7E-10	4.2E-10	3.5E-10	2.5E-10	2.4E-10	1.8E-10	1.7E-10	1.3E-10
SW	6.2E-9	2.4E-9	1.5E-9	8.8E-10	6.6E-10	5.0E-10	4.3E-10	3.5E-10	3.0E-10	2.4E-10
WSW	1.4E-9	5.6E-10	5.0E-10	2.5E-10	2.0E-10	1.6E-10	1.5E-10	1.2E-10	1.0E-10	8.9E-11
W	1.6E-9	6.8E-10	4.8E-10	2.8E-10	1.9E-10	1.8E-10	1.4E-10	1.2E-10	1.0E-10	9.1E-11
WNW	1.2E-9	4.9E-10	3.2E-10	2.0E-10	1.6E-10	1.2E-10	9.4E-11	7.8E-11	6.9E-11	5.8E-11
NW	1.6E-9	7.3E-10	4.8E-10	2.9E-10	2.4E-10	1.8E-10	1.5E-10	1.3E-10	1.1E-10	9.2E-11
NNW	2.0E-9	9.0E-10	6.1E-10	3.7E-10	2.5E-10	2.2E-10	1.7E-10	1.6E-10	1.1E-10	1.0E-10

TABLE C4.0-3 *

(1 of 3)

CATAWBA NUCLEAR STATION
ADULT A_{ait} DOSE PARAMETERS

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	0.0	4.58E-01	4.58E-01	4.58E-01	4.58E-01	4.58E-01	4.58E-01
NA 24	4.11E+02	4.11E+02	4.11E+02	4.11E+02	4.11E+02	4.11E+02	4.11E+02
CR 51	0.0	0.0	1.28E+00	7.65E-01	2.82E-01	1.70E+00	3.22E+02
MN 54	0.0	4.39E+03	8.37E+02	0.0	1.31E+03	0.0	1.34E+04
MN 56	0.0	1.10E+02	1.96E+01	0.0	1.40E+02	0.0	3.52E+03
FE 55	6.64E+02	4.59E+02	1.07E+02	0.0	0.0	2.56E+02	2.63E+02
FE 59	1.05E+03	2.46E+03	9.45E+02	0.0	0.0	6.89E+02	8.21E+03
CO 58	0.0	9.08E+01	2.04E+02	0.0	0.0	0.0	1.84E+03
CO 60	0.0	2.61E+02	5.75E+02	0.0	0.0	0.0	4.90E+03
NI 63	3.14E+04	2.18E+03	1.05E+03	0.0	0.0	0.0	4.54E+02
NI 65	1.28E+02	1.66E+01	7.56E+00	0.0	0.0	0.0	4.20E+02
CU 64	0.0	1.02E+01	4.77E+00	0.0	2.56E+01	0.0	8.66E+02
ZN 65	2.32E+04	7.38E+04	3.33E+04	0.0	4.93E+04	0.0	4.65E+04
ZN 69	4.93E+01	9.44E+01	6.56E+00	0.0	6.13E+01	0.0	1.42E+01
BR 83	0.0	0.0	4.05E+01	0.0	0.0	0.0	5.83E+01
BR 84	0.0	0.0	5.25E+01	0.0	0.0	0.0	4.12E-04
BR 85	0.0	0.0	2.16E+00	0.0	0.0	0.0	0.0
RB 86	0.0	1.01E+05	4.71E+04	0.0	0.0	0.0	1.99E+04
RB 88	0.0	2.90E+02	1.54E+02	0.0	0.0	0.0	4.00E-09
RB 89	0.0	1.92E+02	1.35E+02	0.0	0.0	0.0	1.12E-11
SR 89	2.28E+04	0.0	6.54E+02	0.0	0.0	0.0	3.66E+03
SR 90	2.84E+05	0.0	7.62E+04	0.0	0.0	0.0	1.62E+04
SR 91	4.20E+02	0.0	1.70E+01	0.0	0.0	0.0	2.00E+03
SR 92	1.59E+02	0.0	6.88E+00	0.0	0.0	0.0	3.15E+03
Y 90	5.97E-01	0.0	1.60E-02	0.0	0.0	0.0	6.33E+03
Y 91M	5.64E-03	0.0	2.18E-04	0.0	0.0	0.0	1.66E-02
Y 91	8.75E+00	0.0	2.34E-01	0.0	0.0	0.0	4.82E+03
Y 92	5.24E-02	0.0	1.53E-03	0.0	0.0	0.0	9.18E+02

* Methodology for table provided by: M. E. Wrangler, RAB:NRR:NRC on 3/17/83

TABLE C4.0-3

(2 of 3)

CATAWBA NUCLEAR STATION
ADULT $A_{ai\tau}$ DOSE PARAMETERS

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LII
Y 93	1.66E-01	0.0	4.59E-03	0.0	0.0	0.0	5.27E+03
ZR 95	3.07E-01	9.85E-02	6.67E-02	0.0	1.55E-01	0.0	3.12E+02
ZR 97	1.70E-02	3.43E-03	1.57E-03	0.0	5.18E-03	0.0	1.06E+03
NB 95	4.47E+02	2.49E+02	1.34E+02	0.0	2.46E+02	0.0	1.51E+06
MO 99	0.0	1.13E+02	2.14E+01	0.0	2.55E+02	0.0	2.61E+02
TC 99M	9.41E-03	2.66E-02	3.39E-01	0.0	4.04E-01	1.30E-02	1.57E+01
TC 101	9.68E-03	1.40E-02	1.37E-01	0.0	2.51E-01	7.13E-03	4.19E-14
RU 103	4.84E+00	0.0	2.08E+00	0.0	1.85E+01	0.0	5.65E+02
RU 105	4.03E-01	0.0	1.59E-01	0.0	5.20E+00	0.0	2.46E+02
RU 106	7.19E+01	0.0	9.10E+00	0.0	1.39E+02	0.0	4.65E+03
AG 110M	1.23E+00	1.14E+00	6.78E-01	0.0	2.24E+00	0.0	4.66E+02
TE 125M	2.57E+03	9.32E+02	3.45E+02	7.74E+02	1.05E+04	0.0	1.03E+04
TE 127M	6.50E+03	2.32E+03	7.92E+02	1.66E+03	2.64E+04	0.0	2.18E+04
TE 127	1.06E+02	3.79E+01	2.28E+01	7.82E+01	4.30E+02	0.0	8.33E+03
TE 129M	1.10E+04	4.12E+03	1.75E+03	3.79E+03	4.61E+04	0.0	5.56E+04
TE 129	3.01E+01	1.13E+01	7.34E+00	2.31E+01	1.27E+02	0.0	2.27E+01
TE 131M	1.66E+03	8.12E+02	6.77E+02	1.29E+03	8.23E+03	0.0	8.06E+04
TE 131	1.89E+01	7.90E+00	5.97E+00	1.55E+01	8.28E+01	0.0	2.68E+00
TE 132	2.42E+03	1.56E+03	1.47E+03	1.73E+03	1.51E+04	0.0	7.40E+04
I 130	2.88E+01	8.50E+01	3.35E+01	7.20E+03	1.33E+02	0.0	7.32E+01
I 131	1.59E+02	2.27E+02	1.30E+02	7.43E+04	3.89E+02	0.0	5.98E+01
I 132	7.74E+00	2.07E+01	7.24E+00	7.24E+02	3.30E+01	0.0	3.89E+00
I 133	5.41E+01	9.41E+01	2.87E+01	1.38E+04	1.64E+02	0.0	8.46E+01
I 134	4.04E+00	1.10E+01	3.93E+00	1.90E+02	1.75E+01	0.0	9.57E-03
I 135	1.69E+01	4.42E+01	1.63E+01	2.92E+03	7.09E+01	0.0	4.99E+01
CS 134	2.98E+05	7.09E+05	5.80E+05	0.0	2.29E+05	7.62E+04	1.24E+04
CS 136	3.12E+04	1.23E+05	8.86E+04	0.0	6.85E+04	9.39E+03	1.40E+04
CS 137	3.82E+05	5.22E+05	3.42E+05	0.0	1.77E+05	5.89E+04	1.01E+04
CS 138	2.64E+02	5.22E+02	2.59E+02	0.0	3.84E+02	3.79E+01	2.23E-03
BA 139	1.14E+00	8.14E-04	3.35E-02	0.0	7.61E-04	4.62E-04	2.03E+00

TABLE C4.0-3

(3 of 3)

CATAWBA NUCLEAR STATION
ADULT A_{ait} DOSE PARAMETERS

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA 140	2.39E+02	3.00E-01	1.57E+01	0.0	1.02E-01	1.72E-01	4.93E+02
BA 141	5.55E-01	4.19E-04	1.87E-02	0.0	3.90E-04	2.38E-04	2.62E-10
BA 142	2.51E-01	2.58E-04	1.58E-02	0.0	2.18E-04	1.46E-04	3.54E-19
LA 140	1.55E-01	7.82E-02	2.07E-02	0.0	0.0	0.0	5.74E+03
LA 142	7.94E-03	3.61E-03	9.00E-04	0.0	0.0	0.0	2.64E+01
CE 141	4.31E-02	2.91E-02	3.30E-03	0.0	1.35E-02	0.0	1.11E+02
CE 143	7.59E-03	5.61E+00	6.21E-04	0.0	2.47E-03	0.0	2.10E+02
CE 144	2.25E+00	9.39E-01	1.21E-01	0.0	5.57E-01	0.0	7.59E+02
PR 143	5.71E-01	2.29E-01	2.83E-02	0.0	1.32E-01	0.0	2.50E+03
PR 144	1.87E-03	7.76E-04	9.49E-05	0.0	4.38E-04	0.0	2.69E-10
ND 147	3.90E-01	4.51E-01	2.70E-02	0.0	2.64E-01	0.0	2.17E+03
W 187	2.96E+02	2.48E+02	8.65E+01	0.0	0.0	0.0	8.11E+04
NP 239	3.11E-02	3.06E-03	1.69E-03	0.0	9.54E-03	0.0	6.28E+02

C5.0 Radiological Environmental Monitoring

The Radiological Environmental Monitoring Program shall be conducted in accordance with Technical Specification, Section 3/4.12.

The monitoring program locations and analyses are given in Tables C5.0-1 through C5.0-3 and Figure C5.0-1.

Site specific characteristics make groundwater sampling, special low-level I-131 analyses on drinking water, and food product sampling unnecessary. Groundwater recharge is from precipitation and the groundwater gradient is toward the effluent discharge area; therefore, contamination of groundwater from liquid effluents is highly improbable. Special low-level I-131 analyses in drinking water will not be performed routinely since the expected I-131 dose from this pathway is less than 1 mrem/year. Food products will not be sampled since lakewater irrigation is not practiced in the vicinity.

The laboratory performing the radiological environmental analyses shall participate in an interlaboratory comparison program which has been approved by the NRC. This program is the Environmental Protection Agency's (EPA's) Environmental Radioactivity Laboratory Intercomparsion Studies (crosscheck) Program, our participation code is CP.

TABLE C5.0-1
(1 of 1)
CATAWBA RADIOLOGICAL MONITORING PROGRAM SAMPLING LOCATIONS

(TLD LOCATIONS)

SAMPLING LOCATION DESCRIPTION			SAMPLING LOCATION DESCRIPTION		
200	SITE BOUNDARY	(0.7M NNE)	232	4-5 MILE RADIUS	(4.1M NE)
201	SITE BOUNDARY	(0.5M NE)	233	4-5 MILE RADIUS	(4.0M ENE)
202	SITE BOUNDARY	(0.6M ENE)	234	4.5 MILE RADIUS	(4.5M E)
203	SITE BOUNDARY	(0.5M SE)	235	4.5 MILE RADIUS	(4.0M ESE)
204	SITE BOUNDARY	(0.5M SSW)	236	4-5 MILE RADIUS	(4.2M SE)
205	SITE BOUNDARY	(0.6M SW)	237	4-5 MILE RADIUS	(4.8M SSE)
206	SITE BOUNDARY	(0.7M WNW)	238	4-5 MILE RADIUS	(4.2M S)
207	SITE BOUNDARY	(0.8M NNW)	239	4-5 MILE RADIUS	(4.6M SSW)
212	SPECIAL INTEREST	(2.7M ESE)	240	4-5 MILE RADIUS	(4.1M SW)
217	CONTROL	(10.0M SSE)	241	4-5 MILE RADIUS	(4.7M WSW)
222	SITE BOUNDARY	(0.7M N)	242	4-5 MILE RADIUS	(4.6M W)
223	SITE BOUNDARY	(0.5M E)	243	4-5 MILE RADIUS	(4.6M WNW)
224	SITE BOUNDARY	(0.7M ESE)	244	4-5 MILE RADIUS	(4.1M NW)
225	SITE BOUNDARY	(0.5M SSE)	245	4-5 MILE RADIUS	(4.2M NNW)
226	SITE BOUNDARY	(0.5M S)	246	SPECIAL INTEREST	(8.1M ENE)
227	SITE BOUNDARY	(0.5M WSW)	247	CONTROL	(7.5M ESE)
228	SITE BOUNDARY	(0.6M W)	248	SPECIAL INTEREST	(8.2M SSE)
229	SITE BOUNDARY	(0.9M NW)	249	SPECIAL INTEREST	(8.1M S)
230	4-5 MILE RADIUS	(4.4M N)	250	SPECIAL INTEREST	(10.3M WSW)
231	4-5 MILE RADIUS	(4.2M NNE)	251	CONTROL	(9.8M WNW)

TABLE C5.0-2
(1 of 1)
CATAWBA RADIOLOGICAL MONITORING PROGRAM SAMPLING LOCATIONS
(OTHER SAMPLING LOCATIONS)

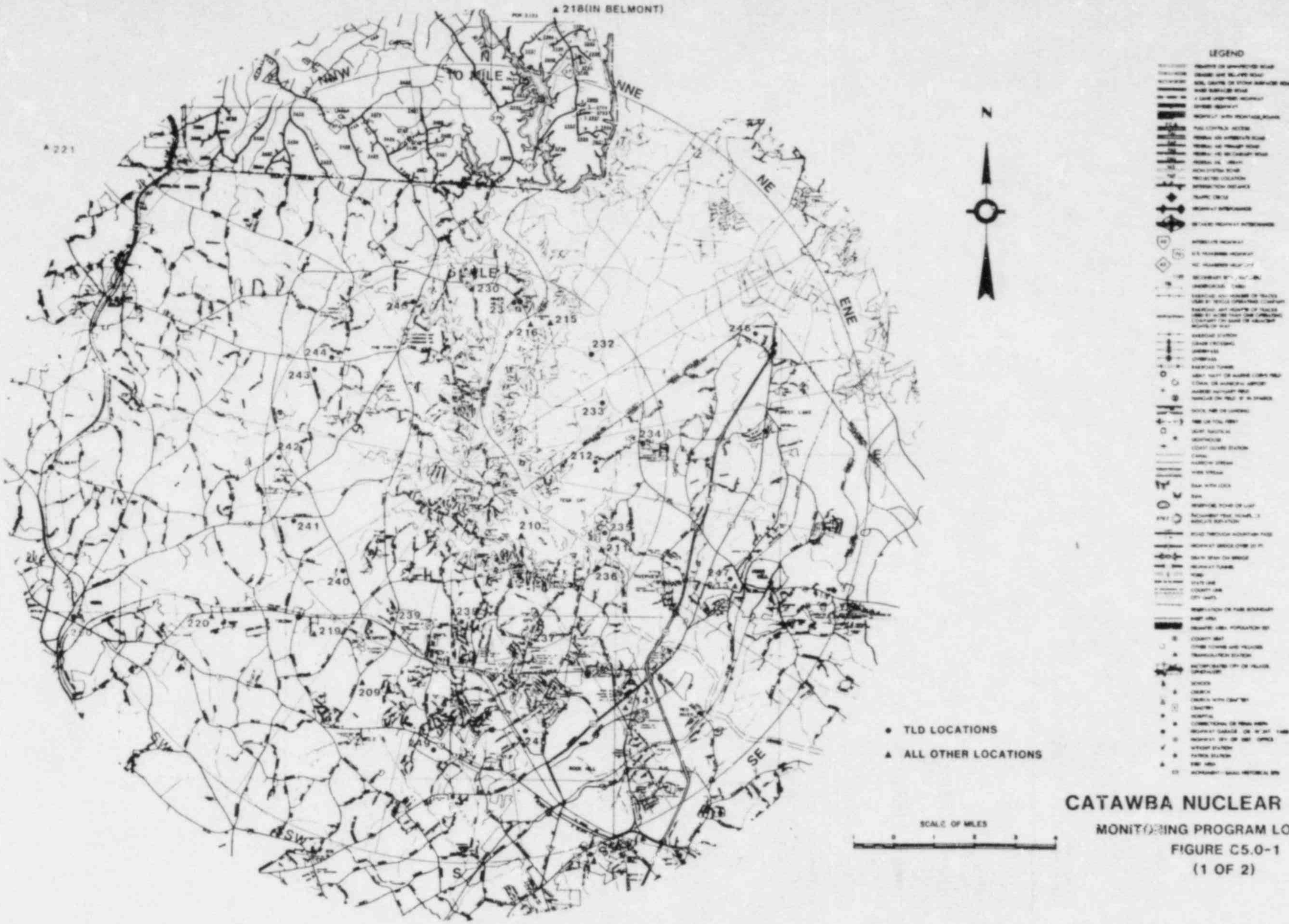
CODE:

W - Weekly
SM - Semimonthly
M - Monthly
Q - Quarterly
SA - Semiannual

	SAMPLING LOCATION DESCRIPTION							
200	Site Boundary (0.7m NNE)	W						
201	Site Boundary (0.5m NE)	W						M
205	Site Boundary (0.6m SW)	W						
208	Discharge Canal (0.5m S)		M		SA		SA	
209	Dairy (7.0m SSW)					SM		
210	Ebenezer Access (2.4m SE)				SA			
211	Wylie Dam (4.0m ESE)		M					
212	Tega Cay (2.7m ESE)	W						
213	Fort Mill Water Supply (7.5m ESE)			M				
214	Rock Hill Water Supply (7.3m SSE)			M				
215	Camp Steere-Hwy 49 (4.1m NNE) Control				SA			
216	Hwy 49 Bridge (4.0m NNE) Control		M				SA	
217	Rock Hill Substation (10.0m SSE) Control	W						M
218	Belmont Water Supply (13.5m N) Control			M				
219	Dairy (6.0m SW)					SM		
220	Dairy (8.0m WSW)					SM		
221	Dairy (13.0m NW) Control					SM		

TABLE C5.0-3
(1 of 1)
CATAWBA RADIOLOGICAL MONITORING PROGRAM ANALYSES

SAMPLE MEDIUM	ANALYSIS SCHEDULE	ANALYSES				
		GAMMA ISOTOPIC	TRITIUM	LOW LEVEL I-131	GROSS BETA	TLD
1. Air Radioiodine and Particulates	Weekly	X				
2. Direct Radiation	Quarterly					X
3. Surface Water	Monthly	X				
	Quarterly Composite		X			
4. Drinking Water	Monthly	X			X	
	Quarterly Composite		X			
5. Shoreline Sediment	Semiannually	X				
6. Milk	Semimonthly	X		X		
7. Fish	Semiannually	X				
8. Broadleaf Vegetation	Monthly	X				

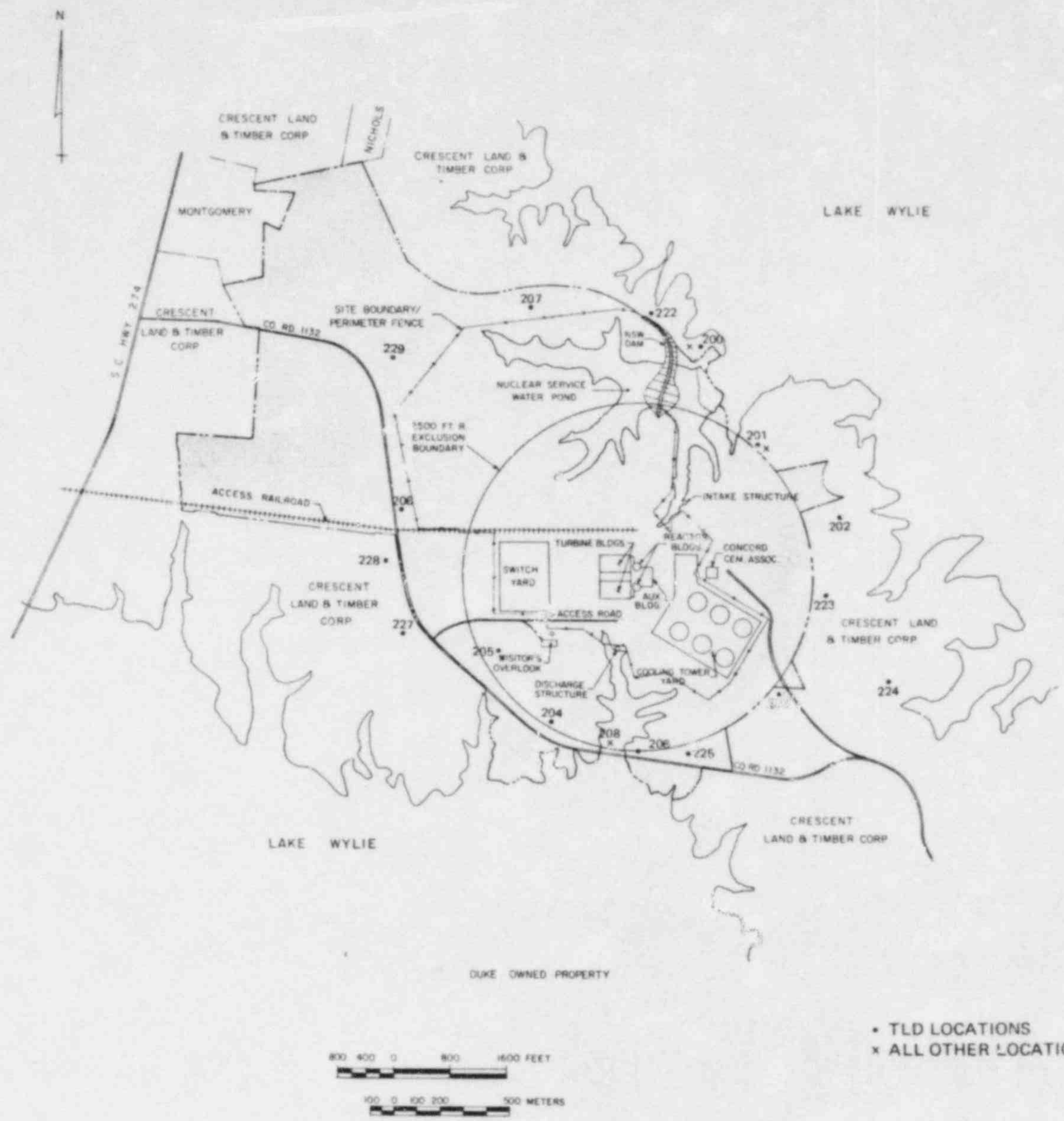


CATAWBA NUCLEAR STATION

MONITORING PROGRAM LOCATIONS

FIGURE C5.0-1

(1 OF 2)



• TLD LOCATIONS
× ALL OTHER LOCATIONS

MONITORING PROGRAM LOCATIONS
FIGURE C5.0-1
(2 OF 2)
CATAWBA NUCLEAR STATION