OFFSITE DOSE ASSESSMENT MANUAL

COOPER NUCLEAR STATION
OFFSITE DOSE ASSESSMENT MANUAL
-ODAM-
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COOPER NUCLEAR STATION
OFFSITE DOSE ASSESSMENT MANUAL
FOR GASEOUS AND LIQUID EFFLUENTS

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OFFSITE DOSE ASSESSMENT MANUAL
FOR GASEOUS AND LIQUID EFFLUENT

1.0 Introduction

This Manual describes acceptable methods of calculating radioactivity concentrations in the environment and the potentially resultant personal dose equivalent commitment beyond the site and exclusion area boundary that are associated with LWR liquid and gaseous effluents. The radioactivity concentrations and dose estimates are used to demonstrate compliance with the Appendix D Specifications required by 10 CFR 50.36.a. The methodology stated in this Manual is acceptable for use in demonstrating operational compliance with 10 CFR 20.1302, 10 CFR 50 Appendix I, and 40 CFR 190.10(a). Only the dose attributable to the Station is considered in demonstrating compliance with 40 CFR 190 since no other nuclear facility exists within 50 miles of the Station.

Calculations are made to assess the air dose from radioactive noble gases near ground level beyond the site and exclusion area boundary location that could be occupied by a person where the maximum air dose is expected. The maximum dose commitment to the person beyond the site and exclusion area boundary potentially experiencing the maximum exposure to all other radioactive material measured in gaseous and liquid effluents released from the Station is also calculated. Alternatively, the dose commitment from effluents other than radioactive noble gases may be calculated to correspond with residence at an occupiable location where airborne exposures are unlikely to underestimate those experienced by the maximally exposed person.

2.0 Liquid Effluent

2.1 Radioactivity In Liquid Waste

The concentration of radionuclides in liquid waste is determined by sampling and analysis in accord with Table D3.1.1-1, Radioactive Liquid Waste Sampling and Analysis. Alternatively, pre-release analysis of the radioactivity concentration in liquid waste required by DSR 3.1.1.1 may be done by gross β-γ counting provided an effluent concentration beyond the site and exclusion area boundary for unidentified emitters, 1×10^{-8} μCi/ml, is applied where the discharge canal meets the river. When a radionuclide concentration is below the LLD for the analysis, it is not reported as being present in the sample. A general diagram of the liquid effluent stream components is shown in Figure 3.1A.

2.2 Aqueous Concentration

Radioactive material in liquid effluent is diluted successively by water flowing in the discharge canal and in the river. The diluted concentration of radionuclide i in a receiving stream is estimated with the equation

$$C_{zi} = C_i \frac{F_1}{F_2}$$

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

    C_{zi} =   concentration of radionuclide i in the receiving stream (μCi/ml)

    F_1 =   release rate of liquid radwaste (ml/sec)*

    F_2 =   dilution flow of receiving stream of water (ml/sec)*

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

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For the purpose of calculating the radioactivity concentration in water beyond the site and exclusion area boundary (Section 2.4), the flow in the discharge canal, F_c, is assigned to F_2.

This method of estimating concentration of radionuclide i in a receiving stream is very conservative as it is based on not exceeding the concentration limits in 10CFR20 Appendix B, Table 2, Column 2 during the period of radioactive material discharge.

As an alternate to the above method, the concentration of radionuclide i in a receiving stream can be calculated using the monthly and quarterly composite samples. This method, discussed in the basis of DSR 3.1.1.2, is performed as follows:

For Sr-89, Sr-90 and Fe-55

$$Czi = \frac{Cqci\,Vrq}{Vdq}$$

where   Czi   =   concentration of radionuclide i in the receiving stream (μci/ml)

Cqci   =   concentration of radionuclide i in the quarterly composite sample (μci/ml)

Vrq   =   volume of liquid radwaste discharged during the quarter (ml)

Vdq   =   total volume of dilution flow corresponding to the time when Vrq was discharged for the quarter (ml)

For all other nuclides

$$Czi = \frac{Cmci\,Vrm}{Vdm}$$

where Czi   =   concentration of radionuclide i in the receiving stream (μci/ml)

    Cmci =   concentration of radionuclide i in the monthly composite sample (μci/ml)

    Vrm =   volume of liquid radwaste discharged during the month (ml)

    Vdm =   total volume of dilution flow corresponding to the time when Vrm was discharged for the month (ml)

2.3   Method of Establishing Alarm Setpoints

The liquid waste effluent monitor and the service water monitor are connected to alarms which provide automatic indication when 10 CFR Part 20, Appendix B, Table 2, Column 2 concentrations are expected to be exceeded beyond the site and exclusion area boundary. With prompt action to reduce radioactive releases following an alarm, the liquid release limit of 10 CFR Part 20.1302 and the limits provided by 10 CFR Part 50, Appendix I, Section IV are unlikely to be exceeded after the alarm.

The alarm setpoint for the liquid effluent radiation monitor is derived from the concentration limit provided in 10 CFR Part 20, Appendix B, Table 2, Column 2 applied where the discharge canal flows into the river. The alarm setpoint does not consider dilution, dispersion, or decay of radioactive material in the river. The radiation monitoring and isolation points are located in the liquid radwaste effluent line and the service water effluent line through which radioactive effluent is, or may be, eventually discharged into the discharge canal.

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The alarm setpoint calculation for each liquid effluent monitor is based upon measurement according to Table D3.1.1-1 of radioactivity in a batch of liquid to be released or in the continuous aqueous discharge. Alternatively, the alarm setpoint may be based upon gross β-γ activity analysis of the liquid waste provided the effluent concentration beyond the site and exclusion area boundary for unidentified emitters, 1×10^{-8} μCi/ml, is observed.

In any case, a monitor may be set to alarm or trip at a lower activity concentration than the calculated setpoint.

2.3.1 Setpoint for a Batch Release

A sample of each batch of liquid radwaste is analyzed for I-131 and principal gamma emitters, or for total activity concentration prior to release. The ratio, $FMPC_b$, of the activity concentration in the tank to the effluent concentration (10 CFR Part 20, Appendix B, Table 2, Column 2) beyond the site and exclusion area boundary is calculated with the equation

$$FMPC_{bp} = \sum_i \frac{C_{bpi}}{MPC_i} \quad identified$$

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where $FMPC_{bp}$ =   fraction of effluent concentration beyond the site and exclusion area boundary in batch derived from activity measured prior to release.

C_{bpi} =   concentration of radionuclide i (including I-131 and principal gamma emitters) in batch sample taken prior to release ($\mu Ci/ml$).

MPC_i =   effluent concentration beyond the site and exclusion area boundary of radionuclide i per 10 CFR Part 20, Appendix B, Table 2, Column 2 ($\mu Ci/ml$)

When $FMPC_{bp}$ is derived from analyses identifying iodine and principal gamma emitters only, the value $FMPC_{bp}$ may be adjusted to account for radionuclides measured in the monthly and quarterly composite sample, but not measured prior to release. This adjustment, derived from measurements during past calendar quarters, is calculated with the equation:

$$FMPC_b = FMPC_{bp} \div E_b$$

where E_b =   (Previous quarterly average of the fraction of the effluent concentration in the discharge canal due to I-131 and primary gamma emitters) ÷ (Previous quarterly average of the fraction of the effluent concentration in the discharge canal due to all radionuclides in batch releases.)

A reference value of E_b, derived from representative past measurements may be used routinely.

Whether radioiodine and primary gamma emitters are identified prior to a batch release or not, the liquid radwaste effluent line radiation monitor alarm and isolation valve closure setpoint is determined with the equation:

$$S = \frac{A}{FMPC_b} \cdot \frac{F_{S2}}{F_{S1}} \cdot g + Bkg$$

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where S = radiation monitor alarm setpoint (cpm or μCi/ml)

A = counting rate (cpm/ml) or activity concentration (μCi/ml) of sample from laboratory analysis*

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

F_{S1} = maximum flow in the batch release line (gal/min)**

F_{S2} = minimum flow in the discharge canal (gal/min)**

Bkg = monitoring instrument background (cpm or μCi/ml)

Note that A÷FMPC$_b$ represents the counting rate of a solution having the same radionuclide distribution as the sample and having the maximum permissible concentration of that mixture.

Gross β-γ analysis alone may be used to determine the radioactivity in a batch prior to release. In that event, the fraction of the effluent concentration beyond the site and exclusion area boundary in the batch is:

$$FMPC_{bp} = \frac{C_{bp}}{1 \times 10^{-8}}$$

where

C_{bp} = gross or total radioactivity concentration in batch sample taken prior to release (μCi/ml)

1×10^{-8} = effluent concentration beyond the site and exclusion area boundary of unidentified radionuclides (μCi/ml)

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*A equals $\sum\limits_i C_{bpi}$ if isotopic analysis was performed or C_{bp} if gross activity analysis was performed.

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

The value of $FMPC_{bp}$ computed with this expression is substituted in the preceding equation to calculate the setpoint.

2.3.2 Setpoint for a Continuous Release

Continuous aqueous radioactive discharges are sampled and analyzed according to the schedule in Table D3.1.1-1. The ratio $FMPC_{cw}$, of the activity concentration in each of the continuous release streams of the effluent concentration beyond the site and exclusion area boundary is calculated with the equations.

$$FMPC_{cw} = \sum_i \frac{C_{cwi}}{MPC_i} \quad identified$$

where

$FMPC_{cw}$ = fraction of effluent concentration beyond the site and exclusion area boundary in continuous release based upon activity measured in weekly composite sample(s).

C_{cwi} = concentration of radionuclide i (including I-131 and principal gamma emitters) in weekly composite sample(s) (μCi/ml)

When $FMPC_c$ is derived from analyses of I-131 and principal gamma emitters, it may be adjusted to account for radionuclides measured in the monthly and quarterly composite sample but not measured prior to release. Adjustment for radionuclides measured in monthly and quarterly composite samples but not in weekly composite samples is given by the equation

$$FMPC_c = FMPC_{cw} \div E_c$$

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where $E_c =$ (Quarterly average fraction of the effluent concentration in the discharge canal due to I-131 and primary gamma emitters measured in weekly composite sample of continuous releases during previous quarter) $\div$ (Quarterly average fraction of the effluent concentration in the discharge canal due to all radionuclides in samples of continuous releases during previous quarter.)

A reference value of E_c, derived from representative past measurements, may be used routinely, instead.

The alarm setpoint of the radiation monitor on the discharge line is determined with the equation

$$S = \frac{A}{FMPC_c} \cdot \frac{F_{S2}}{F_{S1}} \cdot g + Bkg$$

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

Terms g, F_{S1}, F_{S2}, and Bkg are defined the same as in the setpoint equation for a batch release.

Gross β-γ analysis alone may be used to determine the radioactivity in a liquid radioactive discharge. In that event, the fraction of the effluent concentration beyond the site and exclusion area boundary in a sample of the release is:

$$FMPC_c = \frac{C_c}{1 \times 10^{-8}}$$

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where C_c =   gross or total radioactivity concentration in continuous aqueous release (μCi/ml)

1×10^{-8} =   effluent concentration beyond the site and exclusion area boundary of unidentified radionuclides (μCi/ml)

The value of $FMPC_c$ computed with this expression is substituted in the preceding equation to calculate the setpoint.

In the event a long-term trend is evident in setpoints derived from the weekly sample and a setpoint value can be derived from the aggregate of the weekly samples which appears to have less variability and to better represent the effluent, then the setpoint based on the combined, long-term data may be used.

2.4   Radioactivity Concentration in Water Beyond the Site and Exclusion Area Boundary

DSR 3.1.1.2 requires that measured radioactivity concentrations in liquid releases be evaluated to verify that the activity concentration complied with Specification DLCO 3.1.1. Compliance with DLCO 3.1.1 is evaluated by calculating the average radioactivity concentration in water at the end of the discharge canal, expressed as a fraction of effluent concentration beyond the site and exclusion area boundary on the basis of measured release(s), per Table D3.1.1-1, of Fe-55, Sr-89, and Sr-90 averaged over no more than 92 days and other radionuclides averaged over no more than 31 days.

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The average concentration of radioactive noble gases in discharge canal water may be calculated separately as a fraction of the effluent concentration 2×10^{-4} uCi/ml, since the critical exposure pathway for it, immersion in water, differs from the critical exposure pathway for other radionuclides in water, which is via ingestion of the water.

The average concentration, expressed as a fraction of the effluent concentration, is calculated with the equation:

$$\overline{FMPC} = \frac{1}{3785(TE-TB)} \sum_k \frac{1}{F2_k} \sum_i \frac{\hat{Q}_{ki}}{MPC_i}$$

where $\overline{FMPC}$ = fraction of the effluent concentration beyond the site and exclusion area boundary of a mixture of radionuclides in water (unitless, and should be limited to ≤ 1)

3785 = conversion factor (ml/gal)

TE-TB = increment of time between beginning and ending period of interest during which the concentration is averaged (min)

$F2_k$ = flow of aqueous stream beyond the site and exclusion area boundary into which radioactive release represented by sample k is diluted, i.e., the discharge canal flow during the release represented by sample k (gal/min)

Q_{ki} = quantity of radionuclide i represented by sample k which is released as an effluent within the time boundaries TB and TE (μCi)

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MPC$_i$ =   maximum permissible concentration beyond the site and exclusion area boundary of radionuclide i per 10 CFR Part 20, Appendix B, Table 2, Column 2 (μCi/ml)

The data used to compute FMPC are measured by the radioactive liquid sampling and analysis program described in Table D3.1.1-1.

2.5   Accumulated Personal Maximum Dose

DSR 3.1.3.1 requires the dose or dose commitment to a member of the public due to radioactive material released in liquid effluent to be calculated on a <u>cumulative</u> basis at least once every 31 days. The requirement is satisfied by computing the accumulated dose commitment to the most exposed organ and to the total body of a hypothetical person exposed by eating fish taken from the river beyond the site and exclusion area boundary near the discharge canal and drinking water taken from the river three miles downstream.

The accumulated dose commitment is computed at least once every 31 days, but may be computed as analyses becomes available. The dose will be calculated in accordance with Regulatory Guide 1.109, Revision 1, utilizing the LADTAP II computer code.*

The LADTAP II program is routinely used for calculating radiological dose assessments for inclusion in the CNS Radioactive Effluents Release Report. |

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*With quality factor for Tritium reduced from 1.7 to 1.0 per ICRP.

Site specific parameters input to LADTAP II are listed below. These parameters are included in the program calculations and are only changed as conditions and/or situations warrant.

o   CNS effluent water flow in cfs, the average flow in the discharge canal during the time of interest.

o   Dilution factor for the effluent.
        Drinking water: ≤5 (for LADTAP variable Dilution factor)
        Fish*: ≤5 (for LADTAP variable Dilution factor)

*Fishing - Seasonal variation: Consumption of fish is evaluated from April through November.

Alternatively, the accumulated dose commitment may be calculated in the following way:

$$\Delta D_{ank} = 3.785 \times 10^{-3} \sum_{e} \sum_{i} A_{eani} \; C_{ik} \cdot \Delta t_k \left(\frac{F_1}{F_2} \right)_k$$

$$D_{an} = \sum_{k} \Delta D_{ank}$$

where ΔD_{ank} =   the dose commitment (mrem) to organ n of age group a due to the isotopes in a release represented by analysis k, where the analyses are those required by Table D3.1.1-1. Thus the contribution to the dose from gamma emitters become available on a batch basis for batch releases and on a weekly basis for

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continuous releases. Similarly the contributions from H-3 are available on a monthly basis and the contributions from Sr-89, Sr-90, and Fe-55 become available on a quarterly basis.

D_{an} = the dose commitment attributed to releases represented by all analyses k to organ n, including total body, of the maximally exposed person in age group a (mrem).

A_{eani} = transfer factor relating a unit release of radionuclide i (Ci) in a unit stream flow (gal/min) to dose commitment to organ n, or total body, of an exposed person in age group a via environmental pathway e

$$\frac{mrem}{Ci \cdot min/gal}$$

$3.785 \times 10^{-3} = 3785$ ml/gal $\times 10^{-6}$Ci/μCi

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

Δt_k = elapsed time of release represented by sample k during which radionuclide i is discharged at concentration C_{ik}, i.e., the duration of the release represented by sample k (minutes)

$(F/_1/F_2)$ = the quotient of the release flow, F_1, and the dilution flow, F_2, during the release represented by sample k

Pathway-to-dose transfer factors, A_{eani}, for use in calculating the dose commitment arising from radioactive material released in aqueous effluents, are calculated in accordance with equations and values recommended in Regulatory Guide 1.109, Revision O. Appropriate factors representing applicable environmental pathways of exposure and most exposed age group(s) are selected and used in calculating the dose commitment. The pathway(s) and thus age group(s) selected may vary by season. For instance, when fishing near the Station during the winter is nonexistent, evaluation of the fish pathway is not required.

The age group potentially most exposed via eating fish is expected to be the adult, and the age group potentially most exposed via drinking water from the Missouri River is expected to be the infant. Normally, only these need to be evaluated for compliance with DSR 3.1.3.1. For the purpose of calculating the dose to the Member of the Public who is potentially exposed most by eating fish taken from the river beyond the site and exclusion area boundary near the discharge canal, $F_2 = 5F_c$. As long as potable water is known not to be taken from the river within three miles downstream of Cooper Station, as verified by the annual land survey, the potential dose to a Member of the Public via drinking water will be assessed on the basis of water assumed to be taken from the river three miles downstream. At that location, F_2 is conservatively assumed to be $F_2 = 5F_c$. Variables F_1, F_2, and F_c are defined in Section 2.2.

2.6 Projected Personal Maximum Dose

DSR 3.1.3.2 requires the maximum total body and organ doses to a person beyond the site and exclusion area boundary due to radioactive material released in liquid effluent to be projected over a quarter at least one time during every 31 days if

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radioactive liquid radwaste is released and the radwaste system is not operated.

This requirement is satisfied by calculating the projected dose commitment to a hypothetical person exposed by eating fish taken from the river beyond the site and exclusion area boundary near the discharge canal and drinking water taken from the river three miles downstream. The potential dose commitments to organs and to the total body are computed separately.

The quarterly dose commitment to a maximally exposed hypothetical person is projected by computing the accumulated doses to the total body and most exposed organ during the most recent three months and assuming the result represents the projected doses during the current quarter. Doses will be calculated in accordance with Section 2.5.

As an alternative, the quarterly dose commitment to the total body and most exposed organ may be projected by using the equation

$$P_{an} = \frac{91}{X} D_{an}$$

where P_{an} = projected dose commitment (mrem) to organ n (including total body of age group a during the current quarter)

91 = number of days in a quarter

X = number of days to date in current quarter

D_{an} = dose commitment during the quarter-to-date (mrem) based upon results of aqueous effluent sampling and analyses available to date during the quarter

3.0 Gaseous Effluent

3.1 Introduction

The Station discharges gaseous effluent through a stack (Elevated Release Point) and discharges ventilation air from the radwaste, augmented radwaste, turbine, and reactor buildings through the respective building vents. These gaseous effluent streams, radioactivity monitoring points, and effluent discharge points are shown schematically in Figure 3-1. Gaseous release point locations and elevations at Cooper Station are described in Table 3-1. Gaseous discharges from the Elevated Release Point (ERP) are treated as an elevated release while discharges via building vents are assumed to be ground-level releases or split-wake releases.

Gaseous release point locations and elevations at the Station are described in Table 3-1.

3.2 Radioactivity in Gaseous Effluent

For the purpose of estimating radionuclide concentrations and radiation doses, beyond the site and exclusion area boundary measured radionuclide concentrations in gaseous effluent and in ventilation air exhausted from the Station are relied upon. Table D3.2.3-1 identifies the radioactive gaseous effluent measurements. When a radionuclide concentration is below the LLD for the analysis, it is not reported as being present in the sample.

<u>Noble Gases</u>. The distribution of noble gas radionuclides in a gaseous effluent is determined in one of the following ways.

1. Preferably, the radionuclide distribution is obtained by gamma spectrum analysis of effluent gas samples in accordance with DLCO 3.2.3, Table D3.2.3-1. Results of analyses of one or more samples may be averaged to obtain a representative spectrum.

2. In the event a representative radioactive noble gas distribution is unobtainable from samples taken during the period of interest, it may be derived from previous measurements or may be based upon a computed spectrum appearing in Table 3-2.

3. Alternatively, the total activity concentration of radioactive noble gases may be assumed to be krypton - 88.

The total quantity of radioactive noble gas discharged during an interval of time is determined by integrating the rate measurement of each effluent noble gas monitor. This may be done by the effluent monitoring system or the measured activity discharged via a gaseous effluent stream may be calculated with the equation

$$Q = 2.8 \times 10^4 \, \frac{N}{g} \cdot F$$

where Q   =   total radioactive noble gas release via a gaseous effluent stream during a given time interval (μCi)

N = net counts accumulated during the time interval

g = effluent noble gas monitor counting rate response

$$\left(\frac{cpm}{\mu Ci / cm^3} \right)$$

F = gaseous effluent stream discharge rate (cfm)

2.8×10^4 = conversion constant (cm³/ft³)

3.3 Main Condenser Air Ejector Noble Gas Monitor Alarm Setpoint

A noble gas activity monitor is provided to measure gross gamma activity in gases at the main condenser air ejector. The monitor includes an alarm that is set to report when the gamma radiation level in gas discharged by the main condenser air ejector indicates the gross radioactivity discharge rate exceeds 1 Ci/sec.

The alarm setpoint is determined with the relation

$$S = 2120 \ \frac{h \ P}{F} + Bkg$$

where S = the alarm setpoint (mr/hr)

h = monitor response to activity concentration of SJAE offgas being monitored (mr/hr per (μCi/cm³))

F = air ejector discharge rate (cfm)

Bkg = monitoring instrument background (mr/hr)

P = fraction of allowable limit representing a chosen margin of conservatism in the setpoint (unitless)

$$2120 = 1 \ \frac{Ci}{Sec.} \ x \ \frac{1.0E6 \mu Ci}{Ci} \ x \ \frac{1 \ ft^3}{28317 \ cm^3} \ x \ \frac{60 \ sec}{min.}$$

3.4 Effluent Noble Gas Monitor Alarm Setpoint

DSR 3.3.2.9 requires an alarm setpoint to be determined for each radioactive noble gas effluent monitor. Each setpoint is derived to cause the alarm to report when the dose equivalent rate beyond the site and exclusion area boundary due to radioactive noble gas in gaseous effluent exceeds a limit in DLCO 3.2.1.a. Each noble gas activity monitor included in Table D3.3.2-1 except the main condenser air ejector off gas monitor is set to initiate alarm at or below the derived setpoint.

For the purpose of deriving a setpoint, the distribution of noble gas radionuclides in an effluent stream is determined as described in Section 3.2.

3.4.1 Setpoint Based on Dose Rate

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

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$$S = 1.06 \frac{h \cdot P}{f} \frac{\displaystyle\sum_i C_i}{\displaystyle\sum_i \left(C_i \cdot DF_i^s\right)} + Bkg$$

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

$$S = 1.06 \frac{h \cdot P}{f \dfrac{X}{Q}} \frac{\displaystyle\sum_i C_i}{\displaystyle\sum_i \left(C_i \cdot DF_i^v\right)} + Bkg$$

where

S	=	the alarm setpoint (cpm, mr/hr, or $\mu Ci/cm^3$)
h	=	monitor response to activity concentration of effluent being monitored, (cpm per $\mu Ci/cm^3$, mr/hr per $\mu Ci/cm^3$, or $\mu Ci/cm^3$ per $\mu Ci/cm^3$) .
C_i	=	relative concentration of noble gas radionuclide i in effluent at the point of monitoring ($\mu Ci/cm^3$)
X/Q	=	atmospheric dispersion from point of ground-level or split-wake release to the location of potential exposure (sec/m^3)
DF_i^s	=	factor converting elevated release rate of radionuclide i to total body dose equivalent rate at the location of potential exposure (mrem)/(yr $\cdot \mu Ci/sec$)

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DF_i^v = factor converting ground-level or split-wake release of radionuclide i to the total body dose equivalent rate at the location of potential exposure

$$(\text{mrem})/(\text{yr} \cdot \mu\text{Ci/m}^3)$$

f = flow of gaseous effluent stream, i.e., flow past the monitor (ft^3/min)

Bkg = monitoring instrument background (cpm, mr/hr, or μCi/cm^3)

1.06 = 500 mrem/yr x 60 sec/min x 35.3 ft^3/m^3 x 1.0 m^3/1.0 x 10^6 cm^3

P = fraction of allowable limit representing a chosen margin of conservatism in the setpoint (unitless)

Each monitoring channel has a unique response, h, which is determined by the instrument calibration. In order to ensure the correct derivation of a setpoint, the monitor background (Bkg) and the monitor response factor (h) must be in consistent units.

The concentration of each noble gas radionuclide i in a gaseous effluent is determined as discussed in Section 3.2.

The atmospheric dispersion and the dose conversion factor, DF_i^s depends upon local
conditions. For the purpose of calculating radioactive noble gas effluent monitor alarm
setpoints appropriate for Cooper Station, the locations of maximum potential exposure
beyond the site and exclusion area boundary and the reference atmospheric dispersion
factors applicable to the derivation of setpoints are:

Discharge	Discharge	Receptor Location		Atm. Dispersion
Point	Height	Sector	Distance(m)	(sec/m3)
Vent	Ground-Level or Split-Wake	NNW	1,150	3.4×10^{-6}
ERP	Elevated	W	1,800	8.2×10^{-8}

The applicable dose conversion factors, DF_i^s and DF_i^v for deriving setpoints are
in Table 3-3.

3.5 Noble Gas Gamma Radiation Dose Accumulated in Air

DSR 3.2.2.1 requires the calculation on a cumulative basis of air dose due to gamma radiation from radioactive noble gas released in gaseous effluents. DLCO 3.2.2, Condition A requires reporting to the NRC when the air dose beyond the site and exclusion area boundary due to noble gas gamma radiation exceeds 5 mrad during any calendar quarter or 10 mrad during any calendar year.

The distribution of radioactive noble gases in gaseous releases and the quantity discharged during an interval of interest are determined as described in Section 3.2.

The gamma radiation dose to air beyond the site and exclusion area boundary as a consequence of noble gas released from the station will be calculated in accordance with Regulatory Guide 1.109, Revision 1, utilizing USNRC Computer Code GASPAR.*

The GASPAR program is routinely used for calculating radiological dose assessments for inclusion in the CNS Radioactive Effluents Release Report. |

Site specific parameters input to GASPAR are listed below. These parameters are included in the program calculations and are only changed as conditions and/or situations warrant.

o   Source terms for the intervals of interest

o   Fraction of year milch animals are on pasture

o   Atmospheric dispersion factors, for specific locations, obtained from annual meteorological data which are used to determine noble gas (gamma and beta) air doses for the following:

-       Maximum individual site boundary

-       Maximum individual nearest resident

-       Maximum individual nearest milch cow

-       Maximum individual nearest garden

o   Population distribution (0 to 50 miles)

o   Meat, milk, and vegetable distribution (0 to 50 miles)

o   Absolute humidity at Cooper Nuclear Station

o   Fraction of the year vegetables are grown

o   Fraction of daily feed intake derived from pasture for milch and meat animals

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*Quality factor for Tritium reduced from 1.7 to 1.0 per ICRP.

Alternatively, the gamma radiation dose to air beyond the site and exclusion area boundary as a consequence of noble gas released from the station may be calculated with the equation:

$$D = \sum_i \left(Q_{cs_i} \cdot A\gamma_{cs_i} \right) + \sum_i \left(Q_{cv_i} \cdot \left(\frac{X}{Q} \right)_{cv} \cdot A\gamma_{v_i} \right)$$

where D = noble gas gamma dose to air (mrad)

$Q_{cs_i} = \sum \Delta Q_{cs_i}$ = cumulative release of noble gas nuclide i from stack (µCi).

$A\gamma_{cs_i}$ = factor converting unit noble gas stack release to ground level air dose from overhead plume gamma radiation (mrad/µCi).

$A\gamma_{v_i}$ = factor converting time integrated, ground level concentration of noble gas to air dose from gamma radiation

$$\left(\frac{mrad}{\mu Ci \ \frac{sec}{m^3}} \right)$$

Q_{cvi} $= \underset{time}{\Sigma} \Delta Qcvi$ = cumulative release of noble gas nuclide i from building vents (μCi).

$\left(\dfrac{X}{Q}\right)_{cv}$ = long term average atmospheric dispersion factor for a ground level or split wake release (sec/m³).

DSR 3.2.2-1 is satisfied by calculating the noble gas gamma radiation dose to air beyond the site and exclusion area boundary at a point situated at the NNW site boundary, and on the basis of reference* atmospheric dispersion assuming continuous gaseous release. At that location, the reference atmospheric dispersion factor for a vent (ground-level) release is $X/Q = 3.4 \times 10^{-6}$ sec/m³. Appropriate values of $A\gamma_{cs_i}$ and $A\gamma_v$ for use in calculating air doses at that location are listed in Table 3-4.

3.6   Noble Gas Beta Radiation Dose Accumulated in Air

DLCO 3.2.2 requires that the air dose beyond the site and exclusion area boundary from beta radiation not exceed 10 mrad during any quarter and 20 mrad during any year. DSR 3.2.2-1 requires the air dose to be calculated on a cumulative basis.

The radioactive noble gas distribution and activity discharged are determined as described in § 3.4 herein.

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*Onsite meteorological data for the period July 1, 1976, to June 30, 1977, which was used in the Cooper Station Demonstrated of Compliance with 10 CFR 50, Appendix I, revision 1, January, 1978.

The beta radiation dose to air beyond the site and exclusion area boundary as a consequence of noble gas released from the station will be calculated in accordance with Regulatory Guide 1.109, Revision 1, utilizing USNRC Computer Code GASPAR.*

Alternatively, the beta radiation dose to air beyond the site and exclusion area boundary as a consequence of noble gas released from the station may be calculated with the equation

$$D = \sum_i \left(Q_{cs_i} \frac{X}{Q_{cs}} + Q_{cv_i} \frac{X}{Q_{cv}} \right) \cdot A\beta_i$$

Where

D = noble gas beta dose to air (mrad)

$(X/Q)_{cs}$ = long-term average atmospheric dispersion factor for stack releases (sec/m³)

$A\beta_i$ = factor converting time integrated ground level concentration of noble gas radionuclide i to air dose from beta radiation

$$\frac{mrad}{(\mu Ci\ sec)/m^3}$$

DSR 3.2.2.1 is satisfied by calculating the noble gas beta radiation dose to air beyond the site and exclusion area boundary at a point situated at the NNW site boundary, and on the basis of reference atmospheric dispersion assuming continuous gaseous discharge. At that location, the reference atmospheric dispersion factors are:

$(X/Q)_s$ = 1.2 x 10⁻⁸ sec/m³ at the NNW site boundary

$(X/Q)_v$ = 3.4 x 10⁻⁶ sec/m³

Beta radiation-to-air dose conversion factors, $A\beta_i$, for noble gas radionuclides are listed in Table 3-4.

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*Quality factor for Tritium reduced from 1.7 to 1.0 per ICRP.

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3.7 Dose Due to Iodine and Particulates in Gaseous Effluents*

DLCO 3.2.3 requires that radioiodine, and radioactive material in particulate form having half-lives greater than eight days in gaseous effluents released to the area beyond the site and exclusion area boundary cause no more than 7.5 mrem to any organ of a member of the public during any calendar quarter or 15 mrem to an organ of a member of the public during any calendar year. DSR 3.2.3.2 requires the dose to be calculated at least once every 31 days.

Radionuclides other than noble gases or tritium in gaseous effluents that are measured by the sampling and analysis program described in Table D3.2.3-1 are used as the release term in dose calculations. Airborne releases are discharged either via the stack (ERP) as an elevated release or via building vents and treated as a ground level or split-wake release. For each of these release combinations, samples are analyzed weekly, monthly, quarterly, or for a specific release according to Table D3.2.3-1.

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

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*The dose to any organ of a person arising from radioactive iodine-131, iodine-133, and radioactive material in particulate form having half-lives greater than eight days. Noble gases not considered.

$$\Delta Q_{ijk} = C_{ik} \, F_{aj} \, \Delta t_j$$

$$Q_{ik} = \sum_j \, C_{ik} \, F_{aj} \, \Delta t_j$$

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

C_{ik} = concentration of radionuclide i in gaseous effluent identified by analysis k (Ci/m^3) or (μCi/cm^3)

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

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

3.7.1 GASPAR Method

A person may be exposed directly to an airborne concentration of radioactive material discharged in effluent and indirectly via pathways involving deposition of radioactive material onto the ground. Dose estimates account for the separate exposure pathways. The dose commitment to a person beyond the site and exclusion area boundary associated with a gaseous release, Q_{ik}, of radioactive material other than noble gas will be calculated in accordance with Regulatory Guide 1.109, Revision 1, utilizing USNRC Computer Code GASPAR.*

The GASPAR program is routinely used for calculating radiological dose assessments. Site specific parameters input to GASPAR are listed in Section 3.5.

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*Quality factor for Tritium reduced from 1.7 to 1.0 per ICRP.

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3.7.2 Alternate Method

Alternatively, the dose commitment to a person beyond the site and exclusion area boundary associated with a gaseous release, Q_{ik}, of radioactive material other than noble gas may be calculated with one of the appropriate following equations

release via the stack:

$$D_{ansk} = Q_{iks}\left[\sum_i TA_{ani}\left(\frac{Xd}{Q}\right)_{cs} + \sum_e \sum_i TG_{eani}\left(\frac{D}{Q}\right)_{bse}\right]$$

release via a vent:

$$D_{anvk} = Q_{ikv}\left[\sum_i TA_{ani}\left(\frac{Xd}{Q}\right)_{bv} + \sum_e \sum_i TG_{eani}\left(\frac{D}{Q}\right)_{cve}\right]$$

where

D_{ansk} = the dose commitment (mrem) to organ n of a person in age group a due to radionuclides identified in analysis k of an elevated (ERP) release where the analysis is one required by Table D3.2.3-1.

D_{anvk} = the dose commitment from a vent release (mrem)

TA_{ani} = factor converting airborne concentration of radionuclide i to dose commitment to organ n of a person in age group a

$$\left(\frac{mrem}{(Ci\ sec)/m^3}\right)$$

TG_{eani} = factor converting ground deposition of radionuclide i to dose commitment to organ n of a person in age group a exposed via environmental pathway e (mrem/Ci/m²)

(D/Q) = relative deposition factor (m⁻²)

(Xd/Q) =   depleted atmospheric dispersion factor (mCi/m^3 per mCi/sec)

The analysis index k may represent either

p, analysis of a grab sample

w, a weekly composite analysis

m, a monthly composite analysis

q, a quarterly composite analysis

The dose commitment accumulated by a person beyond the site and exclusion area boundary is computed at least every 31 days, but may be calculated as analytical results of effluent measurements, performed according to Table D3.2.3-1, become available.

The dose is accumulated in the following way.

The dose accumulated as a result of stack discharge is

$$D_{ans} = \sum_{w} D_{answ} + \sum_{m} D_{ansm} + \sum_{q} D_{ansq}$$

and the dose accumulated as a result of a vent discharge is

$$D_{anv} = \sum_{w} D_{anvw} + \sum_{m} D_{anvm} + \sum_{q} D_{anvq}$$

Doses committed during the same time period due to discharges from the stack and vents are additive, thus:

$$D_{an} = D_{ans} + \sum_{v} D_{anv}$$

where D_{an} =   the dose commitment accumulated during the quarter to date as a result of all measured radioactive gaseous discharges except noble gases and tritium to any organ n, including total body, of a person offsite in age group a (mrem)

When the dose to a person from iodine and particulates discharged in gaseous effluent is calculated as required by DSR 3.2.3.2, appropriate environmental pathways of exposure will be evaluated. The pathway(s) and/or age group(s) selected may vary by season. Appropriate pathway-to-dose transfer factors, A_{eani}, are selected for use in calculating the dose.

The dose to a receptor 1.1 miles west of the Station is calculated on the basis of continuous gaseous release and reference meteorological conditions. The reference atmospheric dispersion and deposition factors at that location to be used for assessing compliance with DLCO 3.2.3 are:

$$\left(\frac{Xd}{Q}\right)_s = 8.1 \times 10^{-8} \; sec/m^3 \qquad \left(\frac{D}{Q}\right)_s = 4.6 \times 10^{-10} \; m^{-2}$$

$$\left(\frac{Xd}{Q}\right)_v = 4.4 \times 10^{-7} \; sec/m^3 \qquad \left(\frac{D}{Q}\right)_v = 9.5 \times 10^{-10} \; m^{-2}$$

The receptor is assumed to drink milk produced by the milch animal which experiences the maximum D/Q. Maximum values of the relative deposition factors for a milch animal located 3.7 miles northwest of the Station, are:

$$\left(\frac{D}{Q}\right)_s = 1.2 \times 10^{-10} \; m^{-2}$$

$$\left(\frac{D}{Q}\right)_v = 3.7 \times 10^{-10} \; m^{-2}$$

40 CFR Part 190. When the dose due to gaseous effluent is calculated for the purpose of evaluating compliance with 40 CFR Part 190 (reference Section 4.2), the dose contributed by tritium is included in the evaluation and is calculated in the following way.

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Since tritium in water vapor is absorbed directly by vegetation, the tritium concentration in growing vegetation is proportional to the airborne concentration rather than to relative deposition as in the case of particulates. Thus the dose commitment from airborne tritium via vegetation (fruit and vegetables), air-grass-cow-milk, or air-grass-cow-meat pathways is calculated with the appropriate one(s) of the equations:

for a stack release

$$D_{ansk} = \frac{X}{Q_s} \sum_i Q_{iks} \sum_p TA_{anip}$$

for a vent release

$$D_{ankv} = \frac{X}{Q_v} \sum_i Q_{ikv} \sum_p TA_{anip}$$

3.8   Dose to a Person from Noble Gases

DSR 3.4.1.1 requires the calculation of dose to a member of the public for the purpose of assessing compliance with provisions of 40 CFR Part 190.10(a). That assessment includes the calculation of the gamma dose to the total body and the beta plus gamma dose to the skin of the person due to radioactive noble gases in gaseous effluents.

3.8.1   Gamma Dose to Total Body - GASPAR Method

The gamma radiation dose to the whole body of a member of the public as a consequence of noble gas released from the station will be calculated in accordance with Regulatory Guide 1.109, Revision 1, utilizing USNRC Computer Code GASPAR.*

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*Quality factor for Tritium reduced from 1.7 to 1.0 per ICRP.

3.8.1.1 Alternate Method

Alternatively, the gamma radiation dose to the whole body of a member of the public as a consequence of noble gas released from the Station may be calculated with the equation:

$$D\gamma = \sum_i \left(Q_{csi} \cdot P\gamma_{cs_i} + Q_{cv_i} \left(\frac{X}{Q} \right)_{cv} \cdot P\gamma_{v_i} \right)$$

where

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

$P\gamma_{cs_i}$ = factor converting unit noble gas nuclide i in stack release to total body dose at ground level received from the overhead plume (mrem/μCi)

$P\gamma_{v_i}$ = factor converting time integrated, ground level concentration of noble gas nuclide i to air dose from gamma radiation

$$\left(\frac{.mrem}{\mu Ci \; \frac{sec}{m^3}} \right)$$

When the total body dose due to gamma radiation from noble gas is evaluated as required by DSR 3.4.1.1, the dose to the nearby resident exposed most by all applicable exposure pathways combined is computed. Alternatively, the nearby resident exposed to maximal ground-level noble gas concentrations (maximum X/Q) may be selected as the receptor. Values by $P\gamma cs_i$ and $P\gamma v_i$ for this receptor, 1.1 miles west of the station, appear in Table 3-5.

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3.8.2   Dose to Skin - GASPAR Method

The beta radiation dose to the skin of a member of the public due to beta radiation from noble gas released from the station will be calculated in accordance with Regulatory Guide 1.109, Revision 1, utilizing USNRC Computer Code GASPAR.*

3.8.2.1   Alternate Method

Alternatively, the beta radiation dose to the skin of a member of the public due to beta radiation from noble gas released from the Station may be calculated with the equation

$$D\beta = \sum_i \left(Q_{cs_i} \frac{X}{Q_{cs}} + Q_{cv_i} \cdot \frac{X}{Q_{cv}} \right) \cdot S\beta_i$$

where $D\beta$ =   noble gas beta dose to skin (mrem)

$S\beta_i$ =   factor converting time integrated ground level concentration of noble gas radionuclide i to skin dose from beta radiation

$$\frac{mrem}{(\mu Ci\ sec)/m^3}$$

Values of $S\beta_i$ for noble gases are included in Table 3-5.

When the skin dose due to noble gas beta radiation is evaluated as required by DSR 3.4.1.1, the receptor selected is the nearby resident exposed most via all applicable exposure pathways together. Alternatively, the nearby resident exposed to maximal ground-level concentrations (maximum X/Q) may be selected as the receptor. The location of the latter resident is 1.1 miles west of the station.

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*Quality factor Tritium reduced from 1.7 to 1.0 per ICRP.

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

3.9   Projected Organ Dose Due to Gaseous Effluent

DSR 3.2.5.1 and DSR 3.2.4.2 requires organ dose to a member of the public due to radioactive material in air effluent be projected during each month in which radioactive material is released in gaseous effluent without treatment. The purpose is to guide plant personnel in operating the EVTS and the Offgas Treatment System.

The organ dose is projected by calculating the dose to the most exposed organ accumulated during the month to date in accordance with Sections 3.7 and by projecting it for an entire 31 day time by employing the equation:

$$PD = \frac{31}{X} D$$

where:

PD   =   projected organ dose to a member of the public (mrem)

31   =   number of days over which dose is projected

X   =   number of days to date during the projection period

D   =   dose accumulated to the most exposed organ of a member of the public during the month to date (mrem).

3.10   Dose Rate Due to Tritium, Iodines, and Particulates in Gaseous Effluents

DLCO 3.2.1.b requires that the dose rate to any body organ created by the release of tritium, radioiodines, and radioactive material in particulate form having half-lives greater than eight days, shall not exceed 1500 mrem/yr. DSR 3.2.1.1 requires the dose rate to be calculated at least once every 31 days.

The dose equivalent rate from tritium, iodine, and radionuclides in particulate form in airborne effluent due to exposure by inhalation plus tritium absorption through the skin may be calculated for each discharge point by using the following equations. For effluent from an elevated release point, i.e., stack discharge above building wake, the equation is:

$$D_{ans} = \frac{10^{-6}}{TE-TB} \sum_k \sum_i \dot{Q}_{ski} \left(\frac{X}{Q}\right)_s TA_{eani}$$

For effluent from a ground-level release point, i.e., a building vent, the equation is:

$$D_{anv} = \frac{10^{-6}}{TE-TB} \sum_k \sum_i Q_{vki} \left(\frac{X}{Q}\right)_v TA_{eani}$$

Dose rates from separate release points may be combined to give

$$D_{an} = D_{ans} + \sum_v D_{anv}$$

where D_{an} =   dose equivalent commitment rate to organ n of a person in age group a due to radioactive particulates, iodine, and tritium in airborne effluent that are inhaled (mrem/hr)

D_{ans}, D_{anv} = dose equivalent commitment rate due to radioactive particulates, iodine, and tritium from an elevated release and a ground-level release respectively (mrem/hr)

Q_{ski}, Q_{vki} = quantity of radionuclide i released in a given effluent stream, either elevated or ground-level, based on analysis k (uCi) during discharge time increment TB to TE (hr)

TE = ending time of effluent discharge

TB = beginning time of effluent discharge

TE-TB = effluent averaging time (hr)

$(X/Q)_s, (X/Q)_v$ = atmospheric dispersion from an elevated or a ground-level release respectively to ground-level at the receptor (uCi/m³ per uCi/sec)

TA_{eani} = factor converting airborne concentration of radionuclide i to dose commitment to organ n of a person in age group a and where e represents exposure via inhalation

$$\left(\frac{mrem}{(Ci\ sec)/m^3} \right)$$

10^{-6} = conversion, $10^{-6} Ci/\mu Ci$

The analysis index, k, may represent either a grab sample, an integrated (continuous) sample, or a composite sample of an effluent. In turn, each sample represents certain radionuclides in the effluent during the time increment represented by the sample.

4.0 Dose Commitment From Releases Over Extended Time

4.1   Releases During A Quarter

An annual assessment of radiation doses arising from liquid and gaseous effluents from the Station during each calendar quarter is required. The assessment includes the following calculations of doses for

1.   total body and maximally exposed organ doses due to liquid effluent via drinking water and eating fish from the river as in § 2.6.

2.   total body and maximally exposed organ doses due to gaseous effluents* other than noble gases and tritium as in § 3.7.

3.   doses to air offsite due to noble gas γ as in § 3.5 and due to noble gas β as in § 3.6.

The dose calculations are based on liquid and gaseous effluents from the Station during each calendar quarter determined in accord with Tables D3.1.1-1 and D3.2.3-1.

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*radioactive iodine-131, iodine-133, and radioactive material in particulate form, having half-lives greater than eight days.

Aqueous concentration is estimated according to 2.2 on the basis of quarterly averaged stream flow or stream flow during discharge. If practical, quarterly averaged meteorological conditions concurrent with the quarterly gaseous release being evaluated are used to estimate atmospheric dispersion and deposition. Otherwise, the quarterly dose commitment due to gaseous effluent will be calculated using either reference meteorology or annual averaged meteorology during the year in which the release occurred.

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

Seasonal appropriateness of exposure pathways may be considered. Exposure by eating fresh vegetation or drinking milk from cows or goats fed fresh forage is an inappropriate assumption during the first or fourth calendar quarter; rather consumption of stored vegetation and stored forage is ordinarily assumed.

Similarly, the liquid effluent-river-fish-man pathway is not ordinarily assumed during the winter quarter.

Factors converting stack-released noble gas to gamma radiation dose from the overhead plume are calculated on the basis of reference meteorological data for the receptor location.

4.2   Releases During 12 Months

The regulation governing the maximum allowable dose or dose commitment to a member of the public from all uranium fuel cycle sources of radiation and radioactive material in the environment is stated in 40 CFR Part 190.10(a). It requires that the dose or dose commitment to a member of the public from all sources not exceed 75 mrem/yr to the thyroid or 25 mrem/yr to the total body or any other organ. DSR 3.4.1.1 requires calculation of the dose at least once per year to assess compliance with the regulation. If conditions warrant, according to provisions of DLCO 3.4.1, an assessment may be made for a portion of a calendar year.

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

The dose to a member of the public which is due to exposure to radioactive material in liquid and gaseous effluents from the station are ordinarily calculated while the dose attributable to irradiation is evaluated with environmental radiation dosimetry.

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

4.2.1   Calculated Doses

Doses to a member of the public are calculated on the basis of liquid and gaseous effluents from the station determined in accord with Tables D3.1.1-1 and D3.2.3-1.

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

1.   Total body and maximally exposed organ doses due to liquid effluent via drinking water and eating fish from the river as in § 2.6.

2.   Total body dose due to noble gas γ as in § 3.8.1.

3.   Skin dose due to noble gas β as in § 3.8.2.

4. total body and maximally exposed organ doses due to gaseous effluents* other than noble gases as in § 3.7.

Aqueous radioactive material concentrations are estimated according to § 2.2 on the basis of annual averaged stream flow.

Atmospheric dispersion, deposition, and if calculated, exposure by irradiation from airborne emitters are based on annual averaged meteorological conditions during the year evaluated or, alternatively, on reference meteorological conditions. In the event a portion of the year is examined, average meteorology for the period examined may be used in lieu of annual averaged or reference meteorology data.

Factors converting stack-released noble gas to gamma radiation dose from the overhead plume are calculated on the basis of annual averaged meteorological data for the receptor location.

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*radioactive iodine, tritium, and radioactive material in particulate form having half-lives greater than eight days.

4.2.2   Environmental Measurements

When assessing compliance with 40 CFR 190, Radiological Environmental Monitoring Program results may be used to indicate actual radioactivity levels in the environment attributable to CNS as an alternate to calculating the concentrations from radioactive effluent measurements. The measured environmental activity levels may thus be used to supplement the evaluation of doses to real persons for assessing compliance with 40 CFR 190.

The dose to a member of the public due to irradiation (external exposure to gamma radiation) from the station and station effluents will be estimated with the aid of environmental TLD, PIC, or similar environmental dosimetry. This will be done by examining the annual dosimetry data for a statistical difference between measurements near the station and background measurements. Alternatively, irradiation attributable to station effluents may be calculated by methods referenced earlier in this section.

The person most exposed to radiation and radioactive material in effluent from Cooper Station is expected to live within ten miles of the Station. Although the Station is in a rural area, the maximum personal exposure due to airborne effluent almost certainly occurs to a resident within three or four miles of it. Since the nearest public water intake downstream of Cooper Station in the Missouri River is about 85 miles, radioactive liquid effluent contamination of potable water is not foreseen to be significant. The other liquid effluent pathway of potential significance, via fish taken from the river, would be evaluated when assessing compliance with 40 CFR 190 only in the event that a significant increase in fishing downstream in the river near the Station occurs during the previous 12 months. Fishing within about ten miles

downstream of the Station is considered to be nonexistent during the first quarter and negligible during the remainder of the year. In the event the fish pathway is evaluated to assess compliance with 40 CFR 190, the fish would be taken from the river within ten miles downstream of the Station.

5.1   Environmental Sampling Program

DSR 4.1.1 requires a minimum radiological environmental monitoring program to be conducted as described in Table D4.1-1 of that document. APPENDIX C of the ODAM provides a numerical listing of the active sample stations along with a description of the sample types, locations, and maps showing their approximate location.

A radiological environmental monitoring program, approved by the Nuclear Regulatory Commission (NRC) was initiated at CNS before initial criticality was attained on February 21, 1974. The program monitors radiation levels in the air, terrestrial, and aquatic environments. Most samples are collected by Nebraska Public Power District (NPPD) personnel. However, all samples are shipped for analysis to a contractor's laboratory where there exists the special facilities required for measurements of extremely low levels of radioactivity.

A = particulate air filter

H = high efficiency particulate air filter

C = charcoal

⊘ = Instrument. Table D3.3.2-1 names instruments associated with alphanumeric

Note: Exhaust Ventilation Treatment Systems are identified by "EVTS."
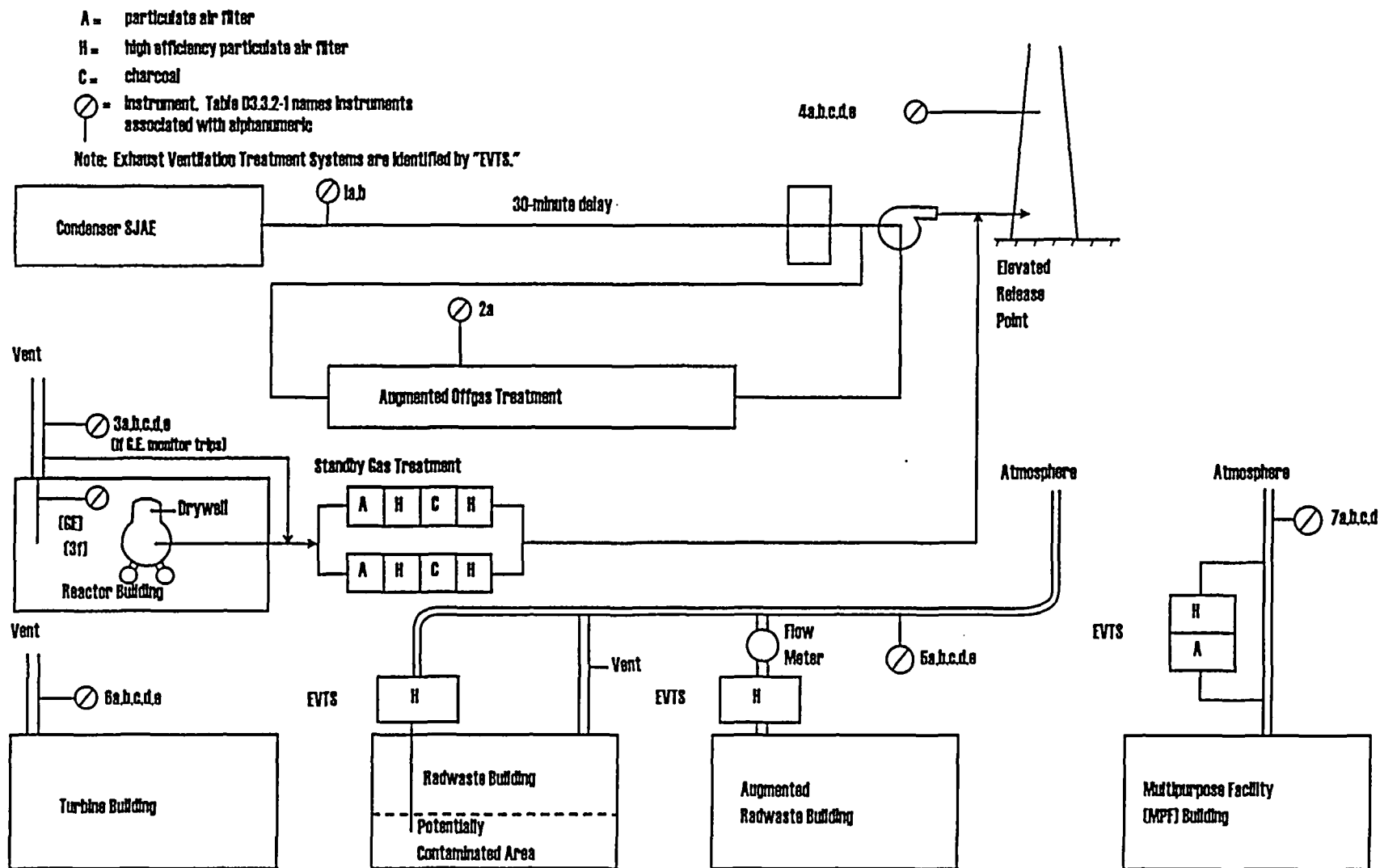
Figure 3-1 Gaseous Effluent Streams, Treatment and Monitoring Equipment, and Discharge Points.

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FIGURE 3.14 – LIQUID EFFLUENT STREAM, TREATMENT AND
MONITORING, AND DISCHARGE POINT.

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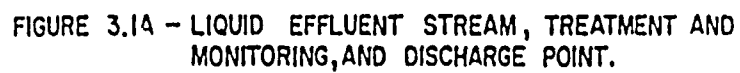
Table 3-1
Atmospheric Gaseous Release Points at the
Cooper Nuclear Generating Station

Structure	Reactor Building	Turbine Building	Combined Radwaste-Augmented Radwaste Building	Elevated Release Point
Number of Ducts	1	4	1	1
Duct Size (inches)	96" x 48"	48" x 96"	72" I.D.	14" I.D.
Height of Vent (feet above roof)	15	1.3	Horizontal discharge at rooftop	325 (above grade)
Flow Rate (cfm)	73405	50710(a)	67070	3000
Flow Velocity (fps)	3.82	26.4	39.5	46.7
Exhaust - Winter	70	70	70	60
Temp. ($^\circ$F) - Summer	90	90	90	90
Release Mode	Partial Elevated	Ground Level	Ground Level	Elevated

(a)   Data given is for one operating fan. Multiply data by total number of fans in operation.

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Table 3-2

Computed Release of Radioactive Noble Gases

In Gaseous Effluent From Cooper Nuclear Station

	Stack Release		Plant Vents Release	
Nuclide	(Ci/yr)	Fraction	(Ci/yr)	Fraction
Kr-83m	3.60E+01	8.38E-03	0	0
Kr-85m	6.50E+01	1.51E-02	7.10E+01	1.14E-02
Kr-85	2.00E+02	4.66E-02	0	0
Kr-87	2.13E+02	4.96E-02	1.33E+02	2.13E-02
Kr-88	2.13E+02	4.96E-02	2.33E+02	3.74E-02
Kr-89	1.00E+03	2.33E-01	0	0
Xe-133m	3.00E 00	6.99E-04	0	0
Xe-133	1.51E+02	3.52E-02	2.63E+03	4.22E-01
Xe-135m	7.20E+01	1.68E-02	6.96E+02	1.12E-01
Xe-135	2.64E+02	6.15E-02	1.06E+03	1.70E-01
Xe-137	1.20E+03	2.79E-01	0	0
Xe-138	8.77E+02	2.04E-01	1.41E+03	2.26E-01
Total	4294.	1.0	6233.	1.0

Releases computed by BWR-GALE for Cooper Station Base Case gaseous radwaste treatment.

The release rate (Ci/yr) is included only to show the basis of the radionuclide distribution. To estimate the concentrations of radionuclides in a sample in which only the total radioactivity has been measured, multiply the total activity concentration by the fraction of respective radionuclides listed above.

Table 3-3

Dose Conversion Factors for Deriving Radioactive

Noble Gas Effluent Monitor Setpoints

Radionuclide	Factor DF^s_i	for Stack Release[a]	Factor DF^γ_i for Ground-Level or Split-Wake Release
	(mrem)/(yr $\cdot \mu$Ci/sec)[a]	mrem/μCi[a]	(mrem)/(yr $\cdot \mu$Ci/m^3)
Kr-83m	3.5E-9	1.1E-16	7.56 E-2
Kr-85m	1.2E-4	3.8E-12	1.17 E3
Kr-85	1.7E-6	5.5E-14	1.61 E1
Kr-87	5.1E-4	1.6E-11	5.92 E3
Kr-88	1.4E-3	4.4E-11	1.47 E4
Kr-89	6.6E-4	2.1E-11	1.66 E4
Kr-90	--	--	1.56 E4
Xe-131m	3.1E-5	9.7E-13	9.15 E1
Xe-133m	2.3E-5	7.3E-13	2.51 E2
Xe-133	2.5E-5	8.0E-13	2.94 E2
Xe-135m	2.5E-4	7.8E-12	3.12 E3
Xe-135	1.9E-4	6.0E-12	1.81 E3
Xe-137	5.4E-5	1.7E-12	1.42 E3
Xe-138	8.0E-4	2.5E-11	8.83 E3
Xe-139	1.6E-5	5.2E-13	5.02 E3
Ar-41	9.7E-4	3.1E-11	8.84 E3

[a]Based on reference meteorology; applicable at the site boundary, 1,250 meters NNW of the ERP.

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Table 3-4

Transfer Factors for Maximum Dose To A

Person Beyond The Site And Exclusion Area Boundary Due To
Radioactive Noble Gases

Radionuclide	Dose Transfer Factors		
	$A\gamma_{cs_i}$ [a] $\dfrac{mrad}{\mu Ci}$	$A\gamma_{v_i}$ $\dfrac{mrad}{\mu Ci\ sec/m^3}$	$A\beta_i$ $\dfrac{mrad}{\mu Ci\ sec/m^3}$
Kr-83m	2.6E-14	6.1E-7	9.13E-6
Kr-85m	4.0E-12	3.9E-5	6.24E-5
Kr-85	5.8E-14	5.4E-7	6.18E-5
Kr-87	1.7E-11	2.0E-4	3.26E-4
Kr-88	4.6E-11	4.8E-4	9.28E-5
Kr-89	2.2E-11	5.5E-4	3.36E-4
Kr-90	--	5.2E-4	2.48E-4
Xe-131m	1.1E-11	4.9E-6	3.52E-5
Xe-133m	8.7E-13	1.0E-5	4.69E-5
Xe-133	9.0E-13	1.1E-5	3.33E-5
Xe-135m	8.3E-12	1.1E-4	2.34E-5
Xe-135	6.3E-12	6.1E-5	7.79E-5
Xe-137	1.8E-12	4.8E-5	4.02E-4
Xe-138	2.7E-11	2.9E-4	1.51E-4
Ar-41	3.2E-11	2.9E-4	1.04E-4

[a]Dose at NNW site boundary

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Table 3-5

Transfer Factors for Maximum Dose To A

Person Beyond Site and Exclusion Area Boundary Due To
Radioactive Noble Gases

Radionuclide	Dose Transfer Factors		
	$P\gamma_{cs_i}$ [a,b]	$P\gamma_{v_i}$	$S\beta_i$
	$\frac{mrem}{\mu Ci}$	$\frac{mrem}{\mu Ci\ sec/m^3}$	$\frac{mrem}{\mu Ci\ sec/m^3}$
Kr-83m	1.6E-16	2.4E-9	--
Kr-85m	2.4E-12	3.7E-5	4.6E-5
Kr-85	3.0E-14	5.1E-7	4.2E-5
Kr-87	7.9E-12	1.9E-4	3.1E-4
Kr-88	2.3E-11	4.7E-4	7.5E-5
Kr-89	6.7E-12	5.3E-4	3.2E-4
Kr-90	--	4.9E-4	2.3E-4
Xe-131m	7.7E-13	2.9E-6	1.5E-5
Xe-133m	5.9E-13	8.0E-6	3.1E-5
Xe-133	6.9E-13	9.3E-6	9.7E-6
Xe-135m	3.3E-12	9.9E-5	2.3E-5
Xe-135	.3.7E-12	5.7E-5	5.9E-5
Xe-137	5.1E-13	4.5E-5	3.9E-4
Xe-138	1.2E-11	2.8E-4	1.3E-4
Ar-41	1.5E-11	2.8E-4	8.5E-5

[a]Receptor located at 1.1 miles west of Station

[b]Based on reference meteorology at Cooper Station

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APPENDIX A

(DELETED)

1/27/89

APPENDIX B

REFERENCE METEOROLOGICAL DATA

Reference meteorological measurements were at Cooper Station during the period from July 1, 1976, through June 30, 1977. The summary data and the computer code, PUFF, were used to generate tables of reference values of X/Q, depleted X/Q, and D/Q herein.

UNDEPLETED MEAN RELATIVE CONCENTRATION (sec/m³)
ELEVATED RELEASE POINT - STANDARD DISTANCES
COOPER NUCLEAR STATION
NEBRASKA PUBLIC POWER DISTRICT

DISTANCE (miles)

SECTOR	.5	1.5	2.5	3.5	4.5	7.5	15.	25.	35.	45.
NNE	6.7E-09	2.3E-08	2.2E-08	1.8E-08	1.5E-08	1.9E-08	5.8E-09	4.7E-09	3.0E-09	1.8E-09
NE	6.1E-09	1.4E-08	1.4E-08	1.3E-08	1.1E-08	1.5E-08	6.9E-09	2.7E-09	2.4E-09	1.8E-09
ENE	7.0E-09	1.4E-08	1.4E-08	1.2E-08	9.3E-09	1.3E-08	2.9E-09	3.7E-09	1.5E-09	9.4E-10
E	6.5E-09	1.4E-08	1.3E-08	1.2E-08	9.5E-09	1.5E-08	4.0E-09	2.3E-09	1.3E-09	3.0E-10
ESE	5.2E-09	1.2E-08	1.0E-08	9.8E-09	7.9E-09	7.3E-09	4.1E-09	1.8E-09	1.2E-09	6.3E-10
SE	8.2E-09	1.9E-08	1.6E-08	1.4E-08	1.2E-08	1.0E-08	3.7E-09	1.6E-09	1.3E-09	6.5E-10
SSE	1.1E-08	3.2E-08	2.3E-08	2.0E-08	3.4E-08	2.6E-08	6.1E-09	2.2E-09	2.3E-09	1.2E-09
S	1.9E-08	3.4E-08	3.3E-08	2.6E-08	2.5E-08	1.6E-08	4.8E-09	2.4E-09	1.4E-09	1.1E-09
SSW	1.0E-08	4.3E-08	1.7E-08	1.7E-08	1.4E-08	9.5E-09	2.5E-09	1.2E-09	9.9E-10	5.1E-10
SW	4.4E-09	5.0E-08	1.7E-08	1.1E-08	1.1E-08	9.3E-09	3.1E-09	1.5E-09	9.4E-10	7.3E-10
WSW	4.1E-09	6.6E-08	3.2E-08	2.8E-08	1.2E-08	6.6E-09	4.1E-09	1.6E-09	1.1E-09	5.0E-10
W	5.6E-09	6.8E-08	3.8E-08	2.2E-08	1.8E-08	6.4E-09	4.1E-09	1.3E-09	8.2E-10	4.9E-10
WNW	6.1E-09	8.0E-08	5.2E-08	3.4E-08	2.1E-08	9.5E-09	3.2E-09	1.6E-09	1.0E-09	6.6E-10
NW	4.8E-09	8.8E-08	7.4E-08	5.2E-08	3.3E-08	1.4E-08	7.2E-09	3.4E-09	1.9E-09	1.3E-09
NNW	8.4E-09	2.7E-08	7.9E-08	6.9E-08	2.2E-08	2.1E-08	5.5E-09	3.1E-09	2.2E-09	1.6E-09
N	7.5E-09	3.5E-08	3.3E-08	2.5E-08	2.0E-08	1.6E-08	6.8E-09	5.2E-09	3.4E-09	1.1E-09

UNDEPLETED MEAN RELATIVE CONCENTRATION (sec/m³)
GROUND LEVEL RELEASE POINT - STANDARD DISTANCES
COOPER NUCLEAR STATION
NEBRASKA PUBLIC POWER DISTRICT

DISTANCE (miles)

SECTOR	.5	1.5	2.5	3.5	4.5	7.5	15.	25.	35.	45.
NNE	3.2E-06	5.5E-07	2.2E-07	1.5E-07	8.0E-08	4.4E-08	1.2E-08	4.9E-09	3.2E-09	2.4E-09
NE	2.0E-06	3.3E-07	1.8E-07	1.2E-07	6.1E-08	3.1E-08	9.2E-09	4.1E-09	2.6E-09	1.4E-09
ENE	2.2E-06	2.9E-07	1.5E-07	8.1E-08	5.4E-08	2.0E-08	7.4E-09	3.1E-09	1.6E-09	8.0E-10
E	2.2E-06	3.1E-07	1.5E-07	7.2E-08	5.5E-08	2.3E-08	6.3E-09	3.1E-09	1.8E-09	9.6E-10
ESE	2.4E-06	3.9E-07	1.5E-07	7.8E-08	5.7E-08	2.7E-08	7.4E-09	2.6E-09	1.3E-09	8.1E-10
SE	2.4E-06	3.9E-07	1.6E-07	1.2E-07	6.1E-08	2.5E-08	6.5E-09	1.8E-09	1.0E-09	7.8E-10
SSE	3.8E-06	6.0E-07	2.6E-07	1.5E-07	9.6E-08	4.2E-08	8.7E-09	2.8E-09	1.7E-09	1.2E-09
S	4.6E-06	8.1E-07	3.7E-07	2.0E-07	1.4E-07	6.6E-08	1.8E-08	6.4E-09	3.6E-09	2.1E-09
SSW	2.6E-06	5.0E-07	2.1E-07	1.1E-07	8.4E-08	5.5E-08	5.6E-09	1.5E-09	8.2E-10	4.8E-10
SW	1.9E-06	2.6E-07	1.8E-07	8.1E-08	6.2E-08	2.0E-08	5.2E-09	1.0E-09	3.9E-10	2.5E-10
WSW	2.0E-06	2.8E-07	1.7E-07	9.0E-08	6.4E-08	1.7E-08	3.6E-09	1.3E-09	7.4E-10	5.1E-10
W	1.6E-06	3.7E-07	1.4E-07	1.0E-07	6.5E-08	1.9E-08	6.1E-09	2.4E-09	1.1E-09	6.0E-10
WNW	3.1E-06	4.9E-07	2.2E-07	1.2E-07	1.0E-07	3.7E-08	1.0E-08	4.1E-09	2.1E-09	1.2E-09
NW	4.9E-06	7.8E-07	3.4E-07	2.2E-07	1.3E-07	6.5E-08	1.9E-08	5.0E-09	2.8E-09	2.0E-09
NNW	6.1E-06	9.7E-07	4.1E-07	2.5E-07	1.7E-07	9.5E-08	2.9E-08	1.2E-08	5.8E-09	1.6E-09
N	5.2E-06	8.9E-07	3.9E-07	2.2E-07	1.6E-07	7.4E-08	2.4E-08	1.1E-08	6.1E-09	3.5E-09

DEPLETED MEAN RELATIVE CONCENTRATION (sec/m³)
ELEVATED RELEASE POINT - STANDARD DISTANCES
COOPER NUCLEAR STATION
NEBRASKA PUBLIC POWER DISTRICT

DISTANCE (miles)

SECTOR	.5	1.5	2.5	3.5	4.5	7.5	15.	25.	35.	45.
NNE	6.6E-09	2.2E-08	2.1E-08	1.7E-08	1.5E-08	1.8E-08	5.4E-09	4.5E-09	2.8E-09	1.6E-09
NE	6.0E-09	1.4E-08	1.4E-08	1.3E-08	1.1E-08	1.5E-08	6.5E-09	2.5E-09	2.2E-09	1.7E-09
ENE	6.9E-09	1.3E-08	1.4E-08	1.1E-08	8.8E-09	1.3E-08	2.7E-09	3.5E-09	1.4E-09	8.6E-10
E	6.4E-09	1.3E-08	1.3E-08	1.1E-08	9.0E-09	1.5E-08	3.9E-09	2.2E-09	1.2E-09	2.6E-10
ESE	5.1E-09	1.1E-08	1.0E-08	9.5E-09	7.6E-09	6.9E-09	3.9E-09	1.6E-09	1.1E-09	5.6E-10
SE	8.1E-09	1.9E-08	1.6E-08	1.3E-08	1.1E-08	9.6E-08	3.4E-09	1.4E-09	1.1E-09	5.5E-10
SSE	1.1E-08	3.1E-08	2.3E-08	2.0E-08	3.3E-08	2.5E-08	5.6E-09	1.9E-09	2.0E-09	9.8E-10
S	1.9E-08	3.3E-08	3.2E-08	2.5E-08	2.4E-08	1.6E-08	4.4E-09	2.0E-09	1.1E-09	8.3E-10
SSW	1.0E-08	4.3E-08	1.7E-08	1.6E-08	1.4E-08	9.0E-09	2.3E-09	1.0E-09	8.6E-10	4.2E-10
SW	4.3E-09	4.9E-08	1.6E-08	1.1E-08	1.0E-08	9.0E-09	2.9E-09	1.4E-09	8.4E-10	6.4E-10
WSW	4.0E-09	6.6E-08	3.2E-08	1.7E-08	1.1E-08	6.3E-09	3.9E-09	1.5E-09	9.5E-10	4.2E-10
W	5.5E-09	6.8E-08	3.7E-08	2.1E-08	1.7E-08	6.0E-09	3.8E-09	1.1E-09	6.8E-10	4.0E-10
WNW	6.0E-09	7.9E-08	5.1E-08	3.3E-08	2.1E-08	9.0E-09	3.0E-09	1.4E-09	8.8E-10	5.5E-10
NW	4.7E-09	8.7E-08	7.3E-08	5.1E-08	3.2E-08	1.3E-08	6.9E-09	3.1E-09	1.7E-09	1.2E-09
NNW	8.3E-09	2.6E-08	7.8E-08	6.8E-08	2.1E-08	2.1E-08	5.1E-09	2.8E-09	2.0E-09	1.5E-09
N	7.3E-09	3.5E-08	3.2E-08	2.4E-08	1.9E-08	1.5E-08	6.3E-09	4.8E-09	3.1E-09	9.4E-10

DEPLETED MEAN RELATIVE CONCENTRATION (sec/m³)
GROUND LEVEL RELEASE POINT - STANDARD DISTANCES
COOPER NUCLEAR STATION
NEBRASKA PUBLIC POWER DISTRICT

DISTANCE (miles)

SECTOR	.5	1.5	2.5	3.5	4.5	7.5	15.	25.	35.	45.
NNE	2.8E-06	4.5E-07	1.7E-07	1.1E-07	6.1E-08	3.2E-08	7.8E-09	2.7E-09	1.6E-09	1.1E-09
NE	1.7E-06	2.8E-07	1.4E-07	9.1E-08	4.6E-08	2.2E-08	5.7E-09	2.2E-09	1.2E-09	5.6E-10
ENE	1.9E-06	2.4E-07	1.2E-07	6.2E-08	4.0E-08	1.4E-08	4.7E-09	1.7E-09	7.7E-10	3.3E-10
E	1.9E-06	2.5E-07	1.2E-07	5.5E-08	4.1E-08	1.6E-08	3.9E-09	1.5E-09	8.3E-10	3.9E-10
ESE	2.1E-06	3.2E-07	1.2E-07	6.0E-08	4.3E-08	1.9E-08	4.6E-09	1.5E-09	6.3E-10	3.9E-10
SE	2.1E-06	3.2E-07	1.3E-07	9.0E-08	4.6E-08	1.7E-08	3.9E-09	9.5E-10	5.0E-10	3.6E-10
SSE	3.3E-06	5.0E-07	2.1E-07	1.2E-07	7.3E-08	3.0E-08	5.4E-09	1.6E-09	8.5E-10	5.2E-10
S	4.0E-06	6.7E-07	3.0E-07	1.6E-07	1.1E-07	4.8E-08	1.2E-08	3.7E-09	1.9E-09	9.4E-10
SSW	2.3E-06	4.2E-07	1.7E-07	8.1E-08	6.3E-08	3.9E-09	3.4E-09	8.4E-10	4.2E-10	2.1E-10
SW	1.7E-06	2.2E-07	1.4E-07	6.1E-08	4.5E-08	1.4E-08	3.1E-09	5.8E-10	1.8E-10	1.1E-10
WSW	1.7E-06	2.3E-07	1.4E-07	6.8E-08	4.7E-08	1.2E-08	2.1E-09	7.0E-10	3.8E-10	2.5E-10
W	1.4E-06	3.0E-07	1.1E-07	7.7E-08	4.8E-08	1.3E-08	3.7E-09	1.2E-09	5.0E-10	2.7E-10
WNW	2.7E-06	4.0E-07	1.7E-07	9.2E-08	7.6E-08	2.7E-08	6.3E-09	2.3E-09	1.0E-09	5.8E-10
NW	4.1E-06	6.5E-07	2.7E-07	1.7E-07	1.0E-07	4.7E-08	1.2E-08	2.9E-09	1.5E-09	9.3E-10
NNW	5.4E-06	8.1E-07	3.3E-07	1.9E-07	1.3E-07	6.9E-08	1.9E-08	6.5E-09	3.0E-09	7.6E-10
N	4.6E-06	7.5E-07	3.1E-07	1.7E-07	1.3E-07	5.4E-08	1.5E-08	5.9E-09	3.0E-09	1.6E-09

MEAN RELATIVE DEPOSITION (m⁻²)
ELEVATED RELEASE POINT - STANDARD DISTANCES
COOPER NUCLEAR STATION
NEBRASKA PUBLIC POWER DISTRICT

DISTANCE (miles)

SECTOR	.5	1.5	2.5	3.5	4.5	7.5	15.	25.	35.	45.
NNE	2.6E-10	3.0E-10	1.8E-10	1.3E-10	9.2E-11	5.7E-11	2.3E-11	1.3E-11	8.1E-12	5.8E-12
NE	1.9E-10	2.0E-10	1.2E-10	8.2E-11	6.1E-11	4.0E-11	1.6E-11	8.3E-12	6.0E-12	3.8E-12
ENE	1.4E-10	1.4E-10	8.7E-11	6.2E-11	4.5E-11	2.9E-11	1.1E-11	5.7E-12	3.8E-12	2.6E-12
E	9.6E-11	9.5E-11	6.4E-11	4.6E-11	3.6E-11	2.3E-11	7.6E-12	3.7E-12	2.5E-12	8.5E-13
ESE	7.7E-11	1.0E-10	6.6E-11	4.8E-11	3.8E-11	2.3E-11	1.2E-11	5.2E-12	3.5E-12	2.0E-12
SE	2.3E-10	2.3E-10	1.4E-10	1.0E-10	7.5E-11	4.0E-11	1.7E-11	7.5E-12	4.8E-12	3.3E-12
SSE	4.2E-10	4.5E-10	2.6E-10	1.7E-10	1.6E-10	7.7E-11	3.3E-11	1.6E-11	1.1E-11	7.6E-12
S	6.4E-10	5.1E-10	3.0E-10	2.0E-10	1.5E-10	7.2E-11	2.9E-11	1.6E-11	1.1E-11	6.7E-12
SSW	3.0E-10	3.4E-10	1.4E-10	9.7E-11	7.2E-11	3.5E-11	1.3E-11	6.5E-12	4.9E-12	2.6E-12
SW	7.9E-11	2.1E-10	8.4E-11	5.0E-11	4.0E-11	2.1E-11	7.4E-12	3.8E-12	2.4E-12	1.8E-12
WSW	5.7E-11	2.3E-10	1.0E-10	6.2E-11	4.3E-11	2.3E-11	8.5E-12	4.3E-12	2.7E-12	1.8E-12
W	1.0E-10	3.4E-10	1.6E-10	9.8E-11	6.9E-11	2.9E-11	1.3E-11	6.2E-12	3.4E-12	2.0E-12
WNW	1.2E-10	4.1E-10	2.1E-10	1.3E-10	8.3E-11	3.9E-11	1.4E-11	7.0E-12	4.1E-12	2.6E-12
NW	1.2E-10	3.8E-10	2.1E-10	1.3E-10	8.2E-11	4.1E-11	1.7E-11	1.0E-11	6.3E-12	3.9E-12
NNW	2.3E-10	2.6E-10	3.0E-10	2.0E-10	1.1E-10	6.0E-11	2.1E-11	1.1E-11	6.1E-12	3.9E-12
N	2.5E-10	3.7E-10	2.3E-10	1.5E-10	1.2E-10	7.1E-11	2.9E-11	1.7E-11	1.3E-11	5.2E-12

MEAN RELATIVE DEPOSITION (m^{-2})
GROUND LEVEL RELEASE POINT - STANDARD DISTANCES
COOPER NUCLEAR STATION
NEBRASKA PUBLIC POWER DISTRICT

DISTANCE (miles)

SECTOR	.5	1.5	2.5	3.5	4.5	7.5	15.	25.	35.	45.
NNE	8.0E-09	1.2E-09	5.2E-10	3.1E-10	2.0E-10	9.9E-11	3.3E-11	1.6E-11	9.6E-12	6.0E-12
NE	5.1E-09	7.6E-10	3.4E-10	2.0E-10	1.3E-10	6.9E-11	2.4E-11	1.1E-11	6.7E-12	4.1E-12
ENE	4.0E-09	6.1E-10	2.7E-10	1.6E-10	1.1E-10	4.8E-11	2.0E-11	7.6E-12	3.9E-12	2.5E-12
E	4.0E-09	6.1E-10	2.8E-10	1.6E-10	1.1E-10	5.0E-11	1.8E-11	8.0E-12	4.2E-12	2.3E-12
ESE	5.3E-09	8.2E-10	3.5E-10	2.0E-10	1.4E-10	6.7E-11	2.1E-11	9.6E-12	5.6E-12	3.8E-12
SE	6.4E-09	9.6E-10	3.9E-10	2.4E-10	1.6E-10	7.1E-11	2.5E-11	1.1E-11	6.8E-12	4.1E-12
SSE	1.0E-08	1.5E-09	6.1E-10	3.5E-10	2.3E-10	1.1E-10	3.9E-11	1.8E-11	1.1E-11	6.5E-12
S	8.7E-09	1.4E-09	5.8E-10	3.3E-10	2.3E-10	1.1E-10	4.0E-11	1.8E-11	1.0E-11	6.3E-12
SSW	3.7E-09	6.0E-10	2.6E-10	1.5E-10	1.0E-10	5.9E-11	1.5E-11	5.0E-12	3.0E-12	1.8E-12
SW	2.9E-09	4.4E-10	2.2E-10	1.2E-10	8.3E-11	3.5E-11	1.1E-11	3.0E-12	1.4E-12	8.7E-13
WSW	2.8E-09	4.6E-10	2.2E-10	1.3E-10	9.0E-11	3.7E-11	1.1E-11	4.2E-12	2.2E-12	1.4E-12
W	3.6E-09	5.9E-10	2.6E-10	1.5E-10	1.0E-10	4.6E-11	1.7E-11	6.9E-12	3.8E-12	2.2E-12
WNW	5.6E-09	8.7E-10	3.8E-10	2.3E-10	1.6E-10	7.3E-11	2.5E-11	1.0E-11	6.2E-12	3.8E-12
NW	1.0E-08	1.6E-09	6.8E-10	4.1E-10	2.7E-10	1.3E-10	4.5E-11	1.8E-11	1.1E-11	6.8E-12
NNW	1.1E-08	1.6E-09	6.9E-10	4.1E-10	2.8E-10	1.4E-10	5.2E-11	2.3E-11	1.3E-11	5.2E-12
N	1.2E-08	1.9E-09	8.1E-10	4.6E-10	3.2E-10	1.5E-10	5.8E-11	2.7E-11	1.7E-11	1.0E-11

APPENDIX C

ENVIRONMENTAL RADIATION MONITORING PROGRAM

Appendix C contains the active environmental sampling stations for the Environmental Radiation Monitoring Program at Cooper Nuclear Station. Included in this appendix is a description of each sample and sample station along with maps showing the approximate location of each sampling station.

REMP SAMPLE STATION DESCRIPTION
SAMPLE TYPES AND SAMPLE LOCATIONS
(See Sample Station Locations Map - Figures C-1 and C-2)

Sample Station[a]	Sample Description - Type and Location

No. 1

Type: (1) Air Particulate and Charcoal Filters
(2) Environmental Thermoluminescent Dosimetry

Location: Outside the northwest edge of fence, east of the gate to the LLRW storage pad on the CNS site, NW¼, S32, T5N, R16E, Nemaha County, Nebraska.

No. 2

Type: (1) Air Particulate and Charcoal Filters
(2) Environmental Thermoluminescent Dosimetry

Location: On north side of county road access to the south portion of the CNS site, SW¼, S32, T5N, R16E, Nemaha County, Nebraska.

No. 3

Type: (1) Air Particulate and Charcoal Filters
(2) Environmental Thermoluminescent Dosimetry

Location: Located on the north side of the Brownville State Recreation Park access road near water gauging station, SE¼, S18, T5N, R16E, Nemaha County, Nebraska.

No. 4

Type: (1) Air Particulate and Charcoal Filters
(2) Environmental Thermoluminescent Dosimetry

Location: Located ½ mile south of Phelps City, Missouri, on west side of Highway "U," NE¼, S2, T64N, R42W, Atchison County, Missouri.

No. 5

Type: (1) Air Particulate and Charcoal Filters
(2) Environmental Thermoluminescent Dosimetry

Location: One-fourth mile south and ¼ mile east of Langdon, Missouri, on north side of road, west of railroad tracks, SW¼, S18, T64N, R41W, Atchison County, Missouri.

No. 6

Type: (1) Air Particulate and Charcoal Filters
(2) Environmental Thermoluminescent Dosimetry

Location: One mile west of the end of Missouri State Highway "U," SW corner of the intersection, NW¼, S34, T64N, R42W, Atchison County, Missouri.

Sample Station	Sample Description - Type and Location	
No. 7	Type: (1)	Air Particulate and Charcoal Filters
	(2)	Environmental Thermoluminescent Dosimetry
	Location:	One-quarter mile east of Highway 67 at Nemaha on north side of road, SW¼, S6, T4N, R16E, Nemaha County, Nebraska.
No. 8	Type: (1)	Air Particulate and Charcoal Filters
	(2)	Environmental Thermoluminescent Dosimetry
	Location:	One-half mile north, ¾ mile west and ¾ mile north of Nemaha on west side of road adjacent to the transmission line, NE¼, S35, T5N, R15E, Nemaha County, Nebraska.
No. 9	Type: (1)	Air Particulate and Charcoal Filters
	(2)	Environmental Thermoluminescent Dosimetry
	Location:	Four miles north of Highway No. 136 on Highway No. 67. One mile east of Highway No. 67 and ½ mile north on west side of road, SW¼, S26, T6N, R15E, Nemaha County, Nebraska.
No. 10	Type: (1)	Air Particulate and Charcoal Filters
	(2)	Environmental Thermoluminescent Dosimetry
	Location:	One mile north of Barada, Nebraska, in SW corner of intersection, NE¼, S14, T3N, R16E, Richardson County, Nebraska.
No. 11	Type: (1)	Water - Ground
	Location:	Plant well water supply header at well pits, NW¼, S32, T5N, R16E, Nemaha County, Nebraska.
No. 12	Type: (1)	Water - River
	Location:	Sample (1) will be taken from the Missouri River immediately upstream from the Plant Intake Structure (River Mile 532.5). During periods when conditions warrant, Station 35 may be used as an alternate to Station 12 (upstream collection site) for sample type (1).
No. 20	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	On NNW boundary of NPPD property, east side of county road, SE¼, S30, T5N, R16E, Nemaha County, Nebraska.

Sample Station	Sample Description - Type and Location

Sample Station | **Sample Description - Type and Location**

No. 28 Type: (1) Water - River
(2) Fish
(3) Sediment from Shoreline

Location: Samples (1) and (3) are taken from the Missouri River or its shore, downstream, near River Mile 530. Sample (2) is taken from the | Missouri river ½ to 3 miles downstream from the plant site. |

No. 35 Type: (1) Fish
(2) Water - River (Alternate Site)
(3) Food Products - Broadleaf Vegetation

Location: Sample (1) is taken from the Missouri River about one to three miles | above CNS intake structure. During periods when conditions warrant, | Station 35 may be used as an alternate to Station 12 (upstream collection site) for sample type (2). Samples (2) and (3) are taken | about ¼ mile south of the Brownville State Recreation Area in Sector | A.

No. 44 Type: (1) Environmental Thermoluminescent Dosimetry

Location: One-quarter mile south of Auburn Country Club on Highway No. 75, | ½ mile east of Highway No. 75 at fence line north of county road, | SE¼, S27, T5N, R14E, Nemaha County, Nebraska.

No. 47 Type: (1) Water - Ground

Location: At Falls City Municipal Water Supply Wells approximately 2 miles | south of Rulo, Nebraska (out of Main Header Flow Meter), east side | of road, SW¼, S20, T1N, R18E, Richardson County, Nebraska. |

No. 56 Type: (1) Environmental Thermoluminescent Dosimetry

Location: One and one-fourth mile SW of Langdon, Missouri, on Highway "U", | on the right side of the highway, NW¼, S23, T64N, R42W, Atchison | County, Missouri.

No. 58 Type: (1) Environmental Thermoluminescent Dosimetry

Location: Three miles south of Brownville, Nebraska, on county road, at the SE | corner of the intersection, with the farm road leading to Sample Station No. 2, SE¼, S31, T5N, R16E, Nemaha County, Nebraska.

No. 59 Type: (1) Environmental Thermoluminescent Dosimetry

Location: One mile SSE of the CNS Elevated Release Point, in the vicinity of | the levee at the south boundary of NPPD property, SE¼, S32, T5N, R16E, Nemaha County, Nebraska.
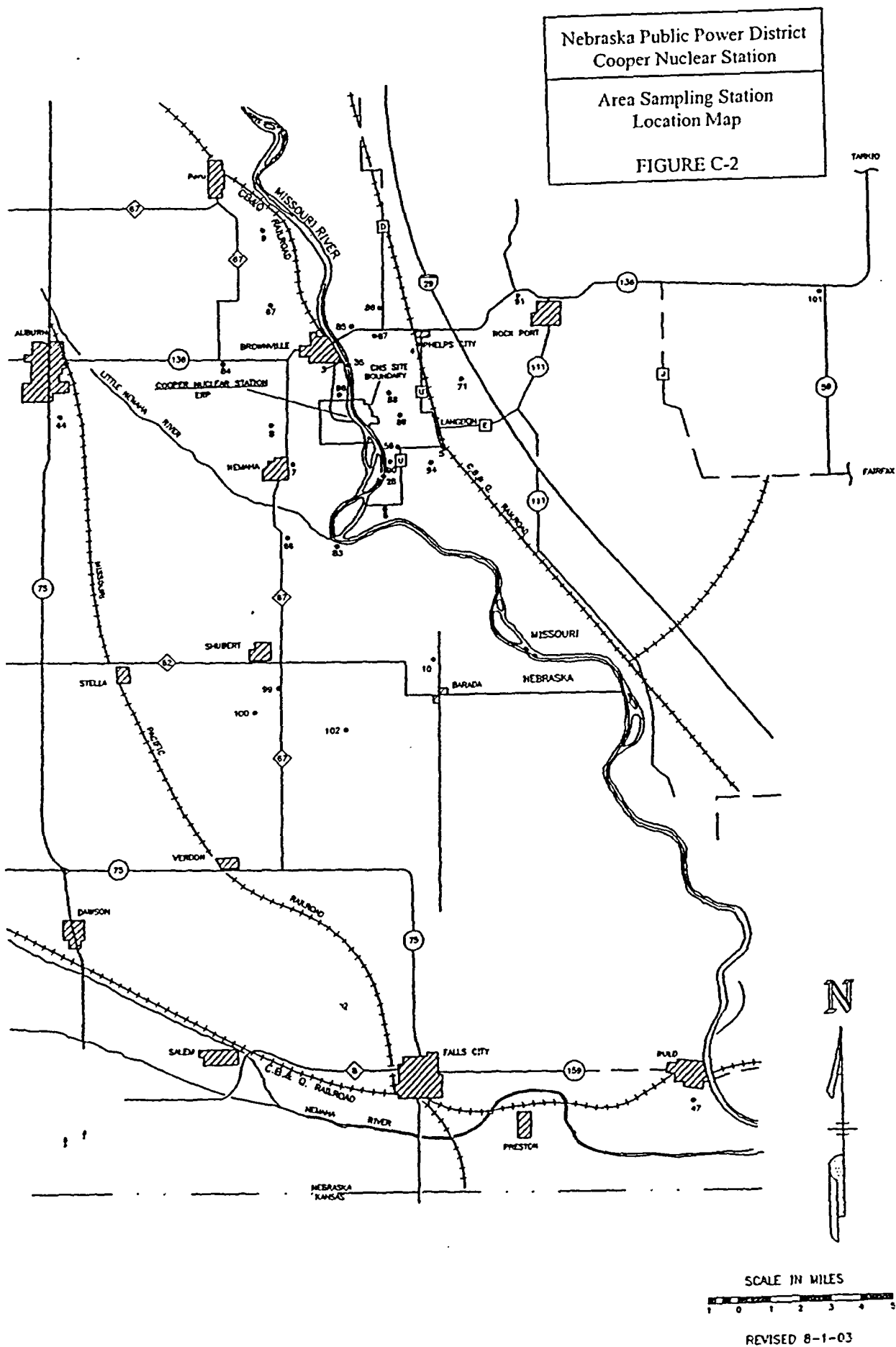
Sample Station	Sample Description - Type and Location	
No. 66	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	Two miles south of Nemaha, Nebraska, on Highway No. 67 - east side of highway, NW¼, S19, T4N, R16E, Nemaha County, Nebraska.
No. 67	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	Two miles west of Brownville, Nebraska, on Highway No. 136, then north 1½ miles on county road, and east ½ mile, on north side of road, NE¼, S11, T5N, R15E, Nemaha County, Nebraska.
No. 71	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	Two miles east of Phelps City, Missouri, on Highway No. 136, then south 1½ miles on county road, and west ¼ mile, SE¼, S6, T64N, R41W, Atchison County, Missouri.
No. 79	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	One and ⅞ miles south of Brownville, Nebraska, on the east side of the paved road, NPPD property, SE¼, S30, T5N, R16E, Nemaha County, Nebraska.
No. 80	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	Two and ⅛ miles south of Brownville, Nebraska, on the east side of the paved road, NPPD property, NE¼, S31, T5N, R16E, Nemaha County, Nebraska.
No. 81	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	Two and ⅜ miles south of Brownville, Nebraska, in the NE corner of the intersection of the paved county road and the CNS access road, NPPD property, NE¼, S31, T5N, R16E, Nemaha County, Nebraska.
No. 82	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	Seven eighths mile south of Cooper Nuclear Station in field, about ½ mile east of farm buildings on NPPD property, SW¼, S32, T5N, R16E, Nemaha County, Nebraska.
No. 83	Type: (1)	Environmental Thermoluminescent Dosimetry
	Location:	Two and ¼ miles south of Nemaha, Nebraska, on Highway No. 67, then east one mile to the junction of the driveway and county road on the east side of the driveway, NE¼, S19, T4N, R16E, Nemaha County, Nebraska.

Sample Station	Sample Description - Type and Location		
No. 84	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	Two and ½ miles west of Brownville, Nebraska, on the south side of Highway No. 136, west of school, NW¼, S22, T5N, R15E, Nemaha County, Nebraska.	|
No. 85	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	One mile east of Brownville, Nebraska, on Highway No. 136, then north ¼ mile on the east side of the county road, NE¼, S33, T65N, R42W, Atchison County, Missouri.	|
No. 86	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	One mile west of Phelps City, Missouri, on Highway No. 136, then north 1½ miles on Highway "D" - on the west side of road, SE¼, S22, T65N, R42W, Atchison County, Missouri.	|
No. 87	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	One mile west of Phelps City, Missouri, on Highway No. 136, then south ½ mile on county road and ¾ mile west on county road to the end of the road, NW¼, S3, T64N, R42W, Atchison County, Missouri.	|
No. 88	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	One mile west of Phelps City, Missouri, on Highway No. 136, then south 2 miles, west side of road, NW¼, S11, T64N, R42W, Atchison County, Missouri.	|
No. 89	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	Two and ½ miles south of Phelps City, Missouri, on Highway "U", then ½ mile west in the SE corner of the county road intersection, NE¼, S14, T64N, R42W, Atchison County, Missouri.	|
No. 90	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	One and ½ miles west and ¾ mile south of Langdon, Missouri, on Highway "U", then ¼ mile west, SW¼, S23, T64N, R42W, Atchison County, Missouri.	|

Sample Station	Sample Description - Type and Location		
No. 91	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	One half mile west of Rock Port, Missouri, on the south side of the intersection of Highway No. 136 and Highway No. 275 at the south side of the water tower, NW¼, S28, T65N, R41W, Atchison County, Missouri.	
No. 94	Type: (1)	Environmental Thermoluminescent Dosimetry	
	Location:	One quarter mile south of Langdon, Missouri, on the west side of the road, NE¼, S24, T64N, R42W, Atchison County, Missouri.	
No. 96	Type: (1)	Food Products - Broadleaf Vegetation	
	Location:	Approximately 1 mile south of Brownville, Nebraska, along paved road in the road ditch in Sector R, SW¼, S19, T5N, R16E, Nemaha County, Nebraska.	
No. 99	Type: (1)	Milk - Nearest Producer	|
	Location:	One and ¼ miles south of Shubert, Nebraska, on the west side of Highway No. 67, NE¼, S24, T3N, R15E, Richardson County, Nebraska.	
No. 100	Type: (1)	Milk (Other Producer)	
	Location:	Two miles south of Shubert, Nebraska, SW¼, S23, T3N, R15E, Richardson County, Nebraska.	|
No. 101	Type: (1)	Food Products - Broadleaf Vegetation	
	Location:	Five and ½ miles east and ½ mile north of Rock Port, Missouri, near the junction of Highway No. 136 and Highway No. 59, in Sector D. Encompasses portions of several sections, T65N, R40W, Atchison County, Missouri.	
No. 102	Type: (1)	Mile (Other Producer)	|
	Location:	Two miles south and 3 miles east, then ½ mile south of Shubert, Nebraska, NW¼, S28, T3N, R16E, Richardson County, Nebraska.	|

NOTES:

(a)   Sample Station numbers missing from the sequence are for inactive or discontinued Sample Stations.

NEBRASKA   MISSOURI

N

NEBRASKA PUBLIC POWER DISTRICT
COOPER NUCLEAR STATION

Site Sampling Station
Location Map

FIGURE C-1

Nebraska Public Power District
Cooper Nuclear Station

Area Sampling Station
Location Map

FIGURE C-2

SCALE IN MILES

REVISED 8-1-03

APPENDIX D

ODAM SPECIFICATIONS

OFFSITE DOSE ASSESSMENT MANUAL
APPENDIX D

TABLE OF CONTENTS

D 1.0 USE AND APPLICATION

The Offsite Dose Assessment Manual (ODAM) Specifications are contained in Section 3.0 of this appendix. They contain operational requirements. Surveillance Requirements. and reporting requirements. Additionally. the Required Actions and associated Completion Times for degraded Conditions are specified. The format is consistent with the Technical Specifications (Appendix A to the CNS Operating License).

The definitions contained in Technical Specifications Section 1.1. "Definitions." apply to the ODAM Specifications. Defined terms are shown in all capital letters. consistent with the Technical Specifications.

The rules of usage for the ODAM Specifications are the same as those for the Technical Specifications. These rules are found in Technical Specifications Sections 1.2. "Logical Connectors;" 1.3. "Completion Times;" and 1.4: "Frequency."
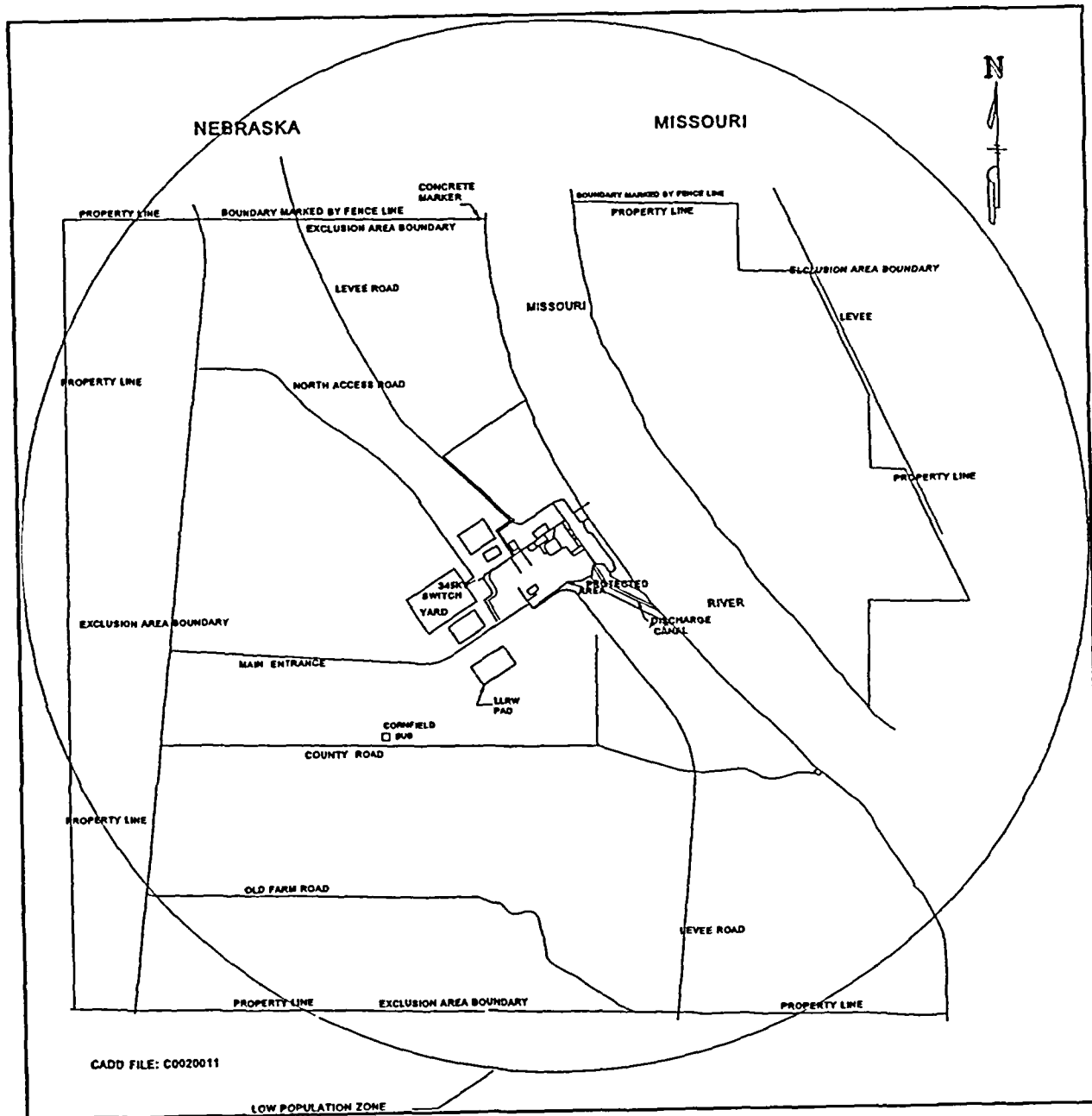
D 2.0 DEFINITIONS

Figure D2.a-1 (Page 1 of 1)
Site and Exclusion Area Boundaries

D 3.0 ODAM Specifications Applicability

DLCO 3.0.1   DLCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in DLCO 3.0.2.

DLCO 3.0.2   Upon discovery of a failure to meet a DLCO, the Required Actions of the associated Conditions shall be met, except as provided in DLCO 3.0.5.

If the DLCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

DLCO 3.0.3   When a DLCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, action shall be initiated to:

a.   Restore compliance with the DLCO or associated ACTIONS, and

b.   Enter the circumstances into the Corrective Action Program.

------------------------------------NOTE----------------------------------
DLCO 3.0.3.b shall be completed if DLCO 3.0.3 is entered.

Exceptions to this Specification are stated in the individual Specifications.

DLCO 3.0.4   Not Applicable to ODAM Specifications.

DLCO 3.0.5   Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to DLCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

D 3.0 ODAM Specifications Applicability

DSR 3.0.1   DSRs shall be met during the MODES or other specified conditions in the
Applicability for individual DLCOs, unless otherwise stated in the DSR. Failure to
meet a Surveillance, whether such failure is experienced during the performance
of the Surveillance or between performances of the Surveillance, shall be failure
to meet the DLCO. Failure to perform a Surveillance within the specified
Frequency shall be failure to meet the DLCO except as provided in DSR 3.0.3.
Surveillances do not have to be performed on inoperable equipment or variables
outside specified limits.

DSR 3.0.2   The specified Frequency for each DSR is met if the Surveillance is performed
within 1.25 times the interval specified in the Frequency, as measured from the
previous performance or as measured from the time a specified condition of the
Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis,
the above Frequency extension applies to each performance after the initial
performance.

Exceptions to this Specification are stated in the individual Specifications.

DSR 3.0.3   If it is discovered that a Surveillance was not performed within its specified
Frequency, then compliance with the requirement to declare the DLCO not met
may be delayed, from the time of discovery, up to 24 hours or up to the limit of
the specified Frequency, whichever is less. This delay period is permitted to
allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the DLCO must
immediately be declared not met, and the applicable Condition(s) must be
entered.

When the Surveillance is performed within the delay period and the Surveillance
is not met, the DLCO must immediately be declared not met, and the applicable
Condition(s) must be entered.

D 3.1 LIQUID EFFLUENTS

D 3.1.1 Liquid Effluents Concentration

DLCO 3.1.1   The concentration of radioactive material in water beyond the Site and Exclusion Area Boundary (Figure D2.a-1) due to radioactive liquid effluent shall not exceed:

a.   The concentration specified in 10 CFR Part 20.1302 for radionuclides other than dissolved or entrained noble gases; and

b. 2×10^{-4} μCi/ml total activity concentration for dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	Concentration of radioactive material beyond the Site and Exclusion Area Boundary due to radioactive liquid effluent exceeds limits.	A.1	Initiate action to restore concentration to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.1.1.1	Perform radioactive liquid waste sampling and activity analysis.	In accordance with Table D3.1.1-1
DSR 3.1.1.2	The analytical results shall be used with methods in the ODAM to verify that the average concentration beyond the Site and Exclusion Area Boundary does not exceed DLCO 3.1.1 when SR-89, SR-90, and Fe-55 concentrations are averaged over no more than 3 months and other radionuclide concentrations are averaged over no more than 31 days.	In accordance with ODAM Section 2.4

Table D3.1.1-1 (Page 1 of 2)
Radioactive Liquid Waste Sampling and Analysis

LIQUID RELEASE TYPE	SAMPLE TYPE	SAMPLE FREQUENCY	ANALYSIS FREQUENCY	SAMPLE ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD)(h)
Batch Waste Release Tanks (c)	Grab sample	Each batch (a)	Each batch (a)	Principal Gamma Emmitters (j)(k)	5×10^{-7} µCi/ml (i)
				I-131	1×10^{-6} µCi/ml
	Grab sample	One batch/ 31 days (a)	31 days (b)	Dissolved and Entrained Gases (gamma emitters)	1×10^{-5} µCi/ml
	Proportional Composite of grab samples (f)	Each batch (a)	31 days (b)	H-3	1×10^{-5} µCi/ml
				Gross Alpha	1×10^{-7} µCi/ml
			92 days (b)	Sr-89	5×10^{-8} µCi/ml
				Sr-90	5×10^{-8} µCi/ml
				Fe-55	1×10^{-6} µCi/ml
Plant Service Water Effluent (d)	Grab Sample	7 days	7 days (b)	Principal Gamma Emitters (j)(k)	5×10^{-7} µCi/ml (i)
Plant Continuous Discharge (e)	Proportional Composite of Grab Samples (g)	24 hours	7 days (b)	Principal Gamma Emitters (j) (k)	5×10^{-7} µCi/ml (i)
				I-131	1×10^{-6} µCi/ml
	Grab Sample	31 days	31 days (b)	Dissolved and Entrained Gases (gamma emitters)	1×10^{-5} µCi/ml
	Proportional Composite of Grab Samples (g)	24 hours	31 days (b)	H-3	1×10^{-5} µCi/ml
				Gross Alpha	1×10^{-7} µCi/ml
			92 days (b)	Sr-89	5×10^{-8} µCi/ml
				Sr-90	5×10^{-8} µCi/ml
				Fe-55	1×10^{-6} µCi/ml

(a) Complete prior to each release.

(b) Analysis may be performed after release.

(c) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed.

(d) A grab sample of plant service water effluent shall be analyzed at least once each week in accordance with Table D3.1.1-1, Plant Service Water Effluent. In the event the radioactivity concentration in a sample exceeds 3×10^{-6} µCi/ml, or in the event the plant service water effluent monitor indicates the presence of an activity concentration greater than 3×10^{-6} µCi/ml, sampling and analysis according to Table D3.1.1-1, Plant Continuous Discharge, shall commence and shall be performed as long as the condition persists.

(e) A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

(f) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

(g) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, daily grab samples shall be collected in proportion to the rate of flow of the effluent stream. Prior to analysis, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

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

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

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

Where:

LLD is the "a priori" lower limit of detection as described above (as picocurie 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),

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

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt shall be used in the calculation.

(i) For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased inversely proportionally to the magnitude of the gamma yield (i.e., $5 \times 10^{-7}/I$, where I is the photon abundance expressed as a decimal fraction), but in no case shall the LLD, as calculated in this manner for a specific radionuclide, be greater than 10% of the value specified in 10 CFR 20, Appendix B, Table 2, Column 2.

(j) The principal gamma emmitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. 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 LLD's higher than required, the reasons shall be documented in the Radioactive Effluent Release Report.

(k) If an isotopic analysis is unavailable, batch releases may be made for up to 14 days provided the gross beta/gamma concentration to the unrestricted area is $\leq 1 \times 10^{-6}$ µc/ml and the sample is analyzed when the instrumentation is once again available.

D 3.1 LIQUID EFFLUENTS

D 3.1.2 Liquid Waste Concentration

DLCO 3.1.2   The concentration of radioactive materials in liquid wastes from pre-release analysis shall be $\leq$ 0.01 µCi/ml, excluding tritium and noble gases.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of radioactive materials in liquid wastes from pre-release analysis > 0.01 µCi/ml, excluding tritium and noble gases.	A.1 Appropriate parts of the liquid radwaste treatment system shall be used to reduce the concentration.	Prior to liquid waste discharge
B. Required Action and associated Completion Time not met. AND Radioactive liquid waste being discharged without treatment in excess of 0.01 µCi/ml, excluding tritium and noble gases.	B.1 Prepare and submit a Special Report to the NRC pursuant to Specification D 5.4 that identifies equipment or subsystems not OPERABLE and the reason for the inoperability, action(s) taken to restore the inoperable equipment to OPERABLE status and a summary description of the action(s) taken to prevent a recurrence.	31 days following the end of the quarter in which the limit was exceeded

D 3.1 LIQUID EFFLUENTS

D 3.1.3 Liquid Effluents Dose

DLCO 3.1.3   The dose to a Member of the Public due to radioactive material in liquid effluents beyond the Site and Exclusion Area Boundary (Figure D2.a-1) shall be limited to:

    a.   ≤ 1.5 mrem to the total body or ≤ 5.0 mrem to any body organ during any calender quarter; and

    b.   ≤ 3.0 mrem to the total body or ≤ 10.0 mrem to any body organ during any calender year.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated dose due to radioactive material in liquid effluents beyond the Site and Exclusion Area Boundary exceeds the limit.	A.1 Prepare and submit a Special Report, in lieu of any other report, pursuant to Specification D 5.4 to the NRC which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken.	31 days following the end of the quarter in which the limit was exceeded

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Calculated dose due to radioactive material in liquid effluents beyond the Site and Exclusion Area Boundary exceeds two times the limit.	B.1 Prepare and submit a Special Report, in lieu of any other report, pursuant to Specification D 5.4 to the NRC which: 1) defines actions to be taken to reduce releases and prevent recurrence, and 2) results of an exposure analysis including effluent pathways and direct radiation to determine whether the dose or dose commitment to a Member of the Public due to radiation and radioactive releases from Cooper Station during the calender year through the period covered by the calculation was $\leq$ 75 mrem to the thyroid and $\leq$ 25 mrem to the total body and all other body organs.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.1.3.1	Perform an assessment of compliance with DLCO 3.1.3.	31 days
DSR 3.1.3.2	Project a prospect of compliance with DLCO 3.1.3 for radioactive liquid releases without radwaste system in operation.	In any quarter in which Radioactive liquid releases are made and the radwaste system is not operated

D 3.1 LIQUID EFFLUENTS

D 3.1.4 Outside Temporary Storage of Radioactive Liquid

DLCO 3.1.4   Radioactive liquid contained in unprotected outdoor temporary liquid storage tanks shall conform to the requirements of Technical Specification (TS) 5.5.8.b.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Level of radioactivity exceeds the limits of TS 5.5.8.b.	A.1 Suspend addition of radioactive material. AND	Immediately
	A.2 Begin measures to reduce content to within the limits of TS 5.5.8.b. AND	Immediately
	A.3 Describe the events leading to the condition in the Radioactive Effluent Release Report.	In accordance with the Radioactive Effluent Release Report frequency

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.1.4.1	Sample and analyze radioactive liquid located in unprotected outdoor temporary liquid storage tanks for level of radioactivity.	7 days during addition of radioactive liquid to the tanks

D 3.2 GASEOUS EFFLUENTS

D 3.2.1 Gaseous Effluents Concentration

DLCO 3.2.1   The dose rate beyond the Site and Exclusion Area Boundary (Figure
                    D2.a-1) due to radioactive gaseous effluents shall be limited to the
                    following:

a.   For noble gases, $\le$ 500 mrem per year to the total body and
      $\le$ 3000 mrem per year to the skin; and

b.   For H-3, I-131, I-133, and radioactive material in particulate form with
      half lives $\ge$ 8 days, $\le$ 1500 mrem per year to any organ when;

1.   The dose rate due to H-3, Sr-89, Sr-90, and alpha emitting
      radionuclides is averaged over $\le$ 3 months and;

2.   The dose rate due to other radionuclides is averaged over $\le$ 31
      days.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Dose rates beyond the Site and Exclusion Area Boundary due to radioactive gaseous effluents exceeds limits.	A.1 Decrease release rate to comply with the limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
DSR 3.2.1.1 Perform an assessment of compliance for DLCO 3.2.1(b).	31 days

D 3.2 GASEOUS EFFLUENTS

D 3.2.2 Noble Gases Dose

DLCO 3.2.2   The air dose beyond the Site and Exclusion Area Boundary (Figure D2.a-1) due to noble gases released in gaseous effluents shall be limited to the following:

a.   For gamma radiation, $\leq$ 5 mrad during any calender quarter and $\leq$ 10 mrad during any calender year; and

b.   For beta radiation, $\leq$ 10 mrad during any calender quarter and $\leq$ 20 mrad during any calender year.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated air dose due to radioactive noble gases beyond the Site and Exclusion Area Boundary exceeds the limit.	A.1 Prepare and submit a Special Report pursuant to Specification D 5.4 to the NRC in lieu of any other report which identifies the cause(s) and defines the corrective actions taken.	31 days following the end of the quarter in which the limit was exceeded

ACTIONS   (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
B.	Calculated air dose due to radioactive noble gases beyond the Site and Exclusion Area Boundary exceeds two times the limit.	B.1	Prepare and submit a Special Report, in lieu of any other report, pursuant to Specification D 5.4 to the NRC which 1) defines actions to be taken to reduce releases and prevent recurrence and 2) results of an exposure analysis including effluent pathways and direct radiation to determine whether the dose or dose commitment to a Member of the Public due to radiation and radioactive releases from Cooper Station during the calender year through the period covered by the calculation was ≤ 75 mrem to the thyroid and ≤ 25 mrem to the total body or any other body organ.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.2.1	Perform an assessment of compliance for DLCO 3.2.2.	31 days

D 3.2 GASEOUS EFFLUENTS

D 3.2.3 Iodine and Particulates

DLCO 3.2.3   The dose to a Member of the Public due to I-131, I-133 and radioactive material in particulate form having a half-life > 8 days in gaseous effluents beyond the Site and Exclusion Area Boundary (Figure D2.a-1) shall be limited to:

a.   ≤ 7.5 mrem to any organ during any calender quarter; and

b.   ≤ 15 mrem to any organ during any calender year.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated dose due to I-131, I-133 and radioactive material in particulate form having a half-life > 8 days beyond the Site and Exclusion Area Boundary exceeds the limit.	A.1 Prepare and submit a Special Report, in lieu of any other report, pursuant to Specification D 5.4 to the NRC which identifies the cause(s) for exceeding the limit(s) and describes the corrective action taken.	31 days following the end of the quarter in which the limit was exceeded.

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Calculated dose due to I-131, I-133 and radioactive material in particulate form having a half-life > 8 days beyond the Site and Exclusion Area Boundary exceeds two times the limit.	B.1 Prepare and submit a Special Report, in lieu of any other report, pursuant to Specification D 5.4 to the NRC which 1) defines actions to be taken to reduce releases and prevent recurrence and 2) results of an exposure analysis including effluent pathways and direct radiation to determine whether the dose or dose commitment to a Member of the Public due to radiation and radioactive releases from Cooper Station was ≤ 75 mrem to the thyroid and ≤ 25 mrem to the total body or any other body organ.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.3.1	Perform radioactive gaseous waste sampling and activity analysis on effluents other than noble gases.	In accordance with Table D3.2.3-1
DSR 3.2.3.2	Perform a dose assessment to determine compliance with DLCO 3.2.3.	31 days

Table D3.2.3-1 (Page 1 of 3)
Radioactive Gaseous Waste Sampling and Analysis

GASEOUS RELEASE TYPE	SAMPLE TYPE	SAMPLE FREQUENCY	ANALYSIS FREQUENCY	SAMPLE ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION(LLD) (h)
1. Elevated Release Point (ERP)	Grab Sample	31 days (c)	31 days (c)	Principal Gamma Emitters (f)	1×10^{-4} µCi/ml (i)
	Grab Sample	92 days (a)	92 days	H-3	1×10^{-6} µCi/ml
	Charcoal Sample	Continuous (b)	7 days (d)	I-131	1×10^{-12} µCi/ml
				I-133	1×10^{-10} µCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1×10^{-11} µCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1×10^{-11} µCi/ml
				Sr-90	1×10^{-11} µCi/ml
				Gross Alpha	1×10^{-11} µCi/ml
	Noble Gas Monitor	Continuous (b)	Continuous	Gross Noble Gases (g) (Beta, Gamma)	1×10^{-6} µCi/ml
2. Reactor Building Vent	Grab Sample	31 days (c)	31 days (c)	Principal Gamma Emitters (f)	1×10^{-4} µCi/ml (i)
	Grab Sample	92 days (a)	92 days	H-3	1×10^{-6} µCi/ml
	Charcoal Sample	Continuous (b)	7 days (d)	I-131	1×10^{-12} µCi/ml
				I-133	1×10^{-10} µCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1×10^{-11} µCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1×10^{-11} µCi/ml
				Sr-90	1×10^{-11} µCi/ml
				Gross Alpha	1×10^{-11} µCi/ml
	Noble Gas Monitor	Continuous (b)	Continuous	Gross Noble Gases (g) (Beta, Gamma)	1×10^{-6} µCi/ml
3. Augmented Radwaste Building Vent	Grab Sample	31 days (c)	31 days (c)	Principal Gamma Emitters (f)	1×10^{-4} µCi/ml (i)
	Grab Sample	92 days (a)	92 days	H-3	1×10^{-6} µCi/ml
	Charcoal Sample	Continuous (b)	7 days (d)	I-131	1×10^{-12} µCi/ml

(continued)

Table D3.2.3-1 (Page 2 of 3)
Radioactive Gaseous Waste Sampling and Analysis

GASEOUS RELEASE TYPE	SAMPLE TYPE	SAMPLE FREQUENCY	ANALYSIS FREQUENCY	SAMPLE ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD)(h)
3. (continued)	Charcoal Sample	Continuous (b)	7 days (d)	I-133	1×10^{-10} µCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1×10^{-11} µCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1×10^{-11} µCi/ml
				Sr-90	1×10^{-11} µCi/ml
				Gross Alpha	1×10^{-11} µCi/ml
	Noble Gas Monitor	Continuous (b)	Continuous	Gross Noble Gases (g) (Beta, Gamma)	1×10^{-6} µCi/ml
4. Turbine Building Vent (Gaseous)	Grab Sample	31 days (c)	31 days (c)	Principal Gamma Emitters (f)	1×10^{-4} µCi/ml (i)
	Grab Sample	92 days (a)	92 days	H-3	1×10^{-6} µCi/ml
	Charcoal Sample	Continuous (b)	7 days (d)	I-131	1×10^{-12} µCi/ml
				I-133	1×10^{-10} µCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1×10^{-11} µCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1×10^{-11} µCi/ml
				Sr-90	1×10^{-11} µCi/ml
				Gross Alpha	1×10^{-11} µCi/ml
	Noble Gas Monitor	Continuous (b)	Continuous	Gross Noble Gases (g) (Beta, Gamma)	1×10^{-6} µCi/ml
5. Multi Purpose Facility (MPF) Building Vent (Gaseous)	Charcoal Sample	Continuous (b)	7 days (d)	I-131	1×10^{-12} µCi/ml
				I-133	1×10^{-10} µCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1×10^{-11} µCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1×10^{-11} µCi/ml
				Sr-90	1×10^{-11} µCi/ml
				Gross Alpha	1×10^{-11} µCi/ml

(a) A H-3 grab sample will also be taken when the reactor vessel head is removed. This sample will be taken at the ERP or Reactor Building Vent whichever will be representative dependent upon the head removal vacuum procedure.

(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 Specifications D 3.2.1, D 3.2.2 and D 3.2.3.

(continued)

Table D3.2.3-1 (Page 3 of 3)
Radioactive Gaseous Waste Sampling and Analysis

(c) Analyses shall also be performed following an increase as indicated by the gaseous release monitor of greater than 50% in the steady state release, after factoring out increases due to power changes or other operational occurrences, which could alter the mixture of radionuclides.

(d) Analysis shall also be performed following an increase as indicated by the gaseous release monitor of greater than 50% in the steady state release, after factoring out increases due to power changes or other operational occurrences, which could alter the mixture of radionuclides. When samples collected for 24 hours or less are analyzed, the corresponding LLD's may be increased by a factor of 10.

(e) A quarterly composite particulate sample shall include a portion of each week's particulate samples collected during the quarter.

(f) 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 the 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 measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances cause LLD's higher than required for more than 31 days, the reasons shall be documented in the Radioactive Effluent Release Report.

(g) The noble gas continuous monitor shall be calibrated using laboratory analysis of the grab samples from Table D3.2.3-1 or using reference sources.

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

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

$$LLD = \frac{(4.66)(s_b)}{(E)(V)(2.22)\,(Y)e^{-\lambda t}}$$

Where:

LLD is the "a priori" lower limit of detection as described above (as picocurie 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),

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

t is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and t shall be used in the calculation.

(i) For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased inversely proportional to the magnitude of the gamma yield (i.e., 1 x 10⁻⁴/I, where I is the photon abundance expressed as a decimal fraction), but in no case shall the LLD, as calculated in this manner for a specific radionuclide, be greater than 10% of the values specified in 10 CFR 20, Appendix B, Table 2, Column 1.

D 3.2 GASEOUS EFFLUENTS

D 3.2.4 Offgas Treatment System

DLCO 3.2.4   Gaseous releases discharged through the Offgas Treatment System shall have at least one train of charcoal adsorbers in service.

APPLICABILITY: Main condenser air ejector in service, except during startup or shutdown with reactor < 10% rated power or when system cannot function due to low offgas flow.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gaseous releases discharged without either train of charcoal adsorbers in service.	A.1 Restore release of gaseous discharge via charcoal adsorbers.	7 days
B. Required Action and associated Completion Time not met.	B.1 Prepare and submit a Special Report pursuant to Specification D 5.4 to the NRC which identifies the inoperable equipment and describes the corrective action taken.	31 days following the end of the quarter in which the release occurred

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
DSR 3.2.4.1	Verify operation of the Offgas Treatment System charcoal adsorbers by using the gaseous effluent monitoring program in D 3.3.2. Gaseous Effluent Monitoring.	In accordance with the DSR frequencies of D 3.3.2.
DSR 3.2.4.2	Project the prospect of compliance with DLCO 3.2.5. Condition B.	Every 31 days when radioactive material in gaseous effluent is released without treatment.

D 3.2 GASEOUS EFFLUENTS

D 3.2.5 Exhaust Ventilation Treatment Systems (EVTS)

DLCO 3.2.5   The Exhaust Ventilation Treatment Systems (EVTS) shall be operated to treat radioactive materials in effluent air.

APPLICABILITY: When radioactive material in gaseous effluent is being released via the associated pathway.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radioactive material in gaseous effluent released without treatment.	A.1 Ensure DSR 3.2.5.1 is met.	31 days
B. Air is discharged without treatment for > 31 days. AND The projected dose to a Member of the Public due to activity in air effluent via that pathway exceeds 0.3 mrem to any body organ.	B.1 Prepare and submit a Special Report pursuant to Specification D 5.4 to the NRC which identifies the inoperable equipment and describes the corrective action taken.	31 days following the end of the quarter in which the release occurred

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
DSR 3.2.5.1 Project the prospect of compliance with DLCO 3.2.5.	Every 31 days when radioactive material in gaseous effluent is released without treatment.

D 3.2 GASEOUS EFFLUENTS

D 3.2.6 Hydrogen Concentration

DLCO 3.2.6   The concentration of hydrogen in the augmented offgas treatment system downstream of the recombiners shall be limited to $\leq$ 2% by volume.

APPLICABILITY: During augmented offgas treatment system operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of hydrogen exceeds limits.	A.1 Restore concentration to within limits.	48 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
DSR 3.2.6.1 Verify Hydrogen concentration in the augmented offgas treatment system downstream of the recombiners is within limits.	24 hours

D 3.2 GASEOUS RELEASES

D 3.2.7 Primary Containment Venting and Purging

DLCO 3.2.7   Venting and purging of the primary containment shall be through the Standby Gas Treatment System.

---------------------------------------NOTES---------------------------------------
1.   This specification does not apply to Normal Ventilation.

2.   This specification does not apply during startup while performing primary containment inerting in accordance with Technical Specification 3.6.3.1, Primary Containment Oxygen Concentration, following a shutdown of > 24 hours.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirement not met.	A.1 Suspend all venting and purging of the primary containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
DSR 3.2.7.1 The primary containment shall be determined to be aligned for venting or purging through the Standby Gas Treatment System.	Once within 4 hours prior to venting or purging of the primary containment AND 12 hours thereafter during venting or purging of the primary containment.

D 3.3 INSTRUMENTATION

D 3.3.1 Liquid Effluent Monitoring

DLCO 3.3.1   The liquid effluent radiation monitoring instrumentation channels shown on Table D3.3.1-1 shall be OPERABLE with:

a.   The minimum OPERABLE channel(s)in service.

b.   The alarm and trip setpoints set to ensure that the limits of DLCO 3.1.1 are not exceeded.

APPLICABILITY: According to Table D3.3.1-1.

ACTIONS

--NOTE--
Separate condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Liquid effluent radiation monitoring instrumentation channel alarm and trip setpoint less conservative than required.	A.1 Suspend liquid effluent radiation release monitored by the inoperable channel. OR	Immediately
	A.2 Declare channel inoperable.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more channels inoperable.	B.1 Enter the Condition referenced in Table D3.3.1-1 for the channel.	Immediately
	AND	
	B.2.1 Restore inoperable channel(s) to OPERABLE status.	31 days
	OR	
	B.2.2 In lieu of any other report, explain in the Radioactive Effluent Release Report why the instrument was not repaired in a timely manner.	In accordance with the Radioactive Effluent Release Report frequency.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action B.1 and referenced in Table D3.3.1-1.	C.1 Analyze a minimum of 2 independent samples in accordance with Table D3.1.1-1. AND C.2 ------------NOTE---------- Determination Action and Verification Action will be performed by two separate technically qualified members of the Facility Staff. ------------------------------ Determine and independently verify the release rate calculations and discharge valving.	Prior to initiating a release Prior to initiating a release
D. As required by Required Action B.1 and referenced in Table D3.3.1-1.	D.1 Collect and analyze a grab sample for gross beta radioactivity or gross gamma radioactivity (as applicable) at a lower limit of detection $\leq 10^{-6}$ µCi/ml.	24 hours AND Once per 24 hours thereafter
E. As required by Required Action B.1 and referenced in Table D3.3.1-1.	E.1 Estimate flow rate during actual release.	4 hours AND Once per 4 hours thereafter

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time for Condition C or E not met.	F.1 Suspend liquid effluent releases monitored by the inoperable channel(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.3.1.1	Perform CHANNEL CHECK.	24 hours
DSR 3.3.1.2	Perform CHANNEL CHECK for each channel to demonstrate OPERABILITY by verifying indication of flow during periods of release.	24 hours on any day on which continuous, periodic, or batch releases are made
DSR 3.3.1.3	Perform SOURCE CHECK.	Completed prior to each release
DSR 3.3.1.4	Perform SOURCE CHECK.	31 days
DSR 3.3.1.5	Perform CHANNEL CALIBRATION	18 months
DSR 3.3.1.6	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate automatic isolation of the pathway for instrument indication levels measured above the alarm/trip setpoint and circuit failure; and control room alarm annunciation for instrument indication levels measured above the alarm/trip setpoint, circuit failure and instrument indicating downscale failure.	92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.3.1.7	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate control room alarm annunciation for instrument indication levels measured above the alarm/trip setpoint, circuit failure, instrument indicating downscale failure, and instrument controls not set in operate mode.	92 days
DSR 3.3.1.8	Perform CHANNEL FUNCTIONAL TEST.	184 days
DSR 3.3.1.9	Perform LOGIC SYSTEM FUNCTIONAL TEST	184 days

Table D3.3.1-1
Radioactive Liquid Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIAL CONDITIONS	MINIMUM CHANNELS OPERABLE	CONDITION REFERENCED FROM ACTION B.1	SURVEILLANCE REQUIREMENTS
1. Gross Beta or Gamma Radioactivity Monitors Providing Automatic Isolation				
a. Liquid Radwaste Effluent Line	(a)	$1_{(b)}$	C	DSR 3.3.1.1 DSR 3.3.1.3 DSR 3.3.1.5 DSR 3.3.1.6 DSR 3.3.1.9
2. Gross Beta or Gamma Radioactivity Monitors Providing Alarm but not Providing Automatic Isolation				
a. Service Water System Effluent Line	(a)	1	D	DSR 3.3.1.1 DSR 3.3.1.4 DSR 3.3.1.5 DSR 3.3.1.7
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	(a)	1	E	DSR 3.3.1.2 DSR 3.3.1.5 DSR 3.3.1.8

(a) During releases via this pathway.

(b) Set to alarm and automatically close the waste discharge valve prior to exceeding the limits of DLCO 3.1.1.

D 3.3 INSTRUMENTATION

D 3.3.2 Gaseous Effluent Monitoring

DLCO 3.3.2   The gaseous effluent radiation monitoring instrumentation channel(s)
                    shown in Table D3.3.2-1 shall be OPERABLE with:

                    a.   The minimum OPERABLE channel(s) in service.

                    b.   The alarm and trip setpoints set to ensure that the limits of
                          DLCO 3.2.1 are not exceeded.

APPLICABILITY: According to Table D3.3.2-1.

ACTIONS

---NOTE--- |
Separate Condition entry is allowed for each channel. |

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gaseous effluent radiation monitoring instrumentation channel alarm and trip setpoint less conservative than required.	A.1 Suspend gaseous effluent radiation release monitored by inoperable channel. OR A.2 Declare channel inoperable.	Immediately Immediately

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
B.	One or more channels inoperable.	B.1	Enter the Condition referenced in Table D3.3.2-1 for the channel.	Immediately
			AND	
		B.2.1	Restore inoperable channel(s) to OPERABLE status.	31 days
			OR	
		B.2.2	In lieu of any other report, explain in the Radioactive Effluent Release Report why the instrument was not repaired in a timely manner.	In accordance with the Radioactive Effluent Release Report frequency
C.	As required by Required Action B.1 and referenced in Table D3.3.2-1.	C.1	Ensure the offgas delay system is not bypassed.	Immediately
			AND	
		C.2	Ensure the Elevated Release Point Monitoring noble gas activity monitor is OPERABLE.	Immediately
			AND	
		C.3	Restore inoperable channels to OPERABLE status.	72 hours
D.	Required Action and associated Completion Time for Condition C not met.	D.1	Be in MODE 2.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action B.1 and referenced in Table D3.3.2-1.	E.1 Estimate flowrate.	24 hours AND Once per 24 hours thereafter
F. As required by Required Action B.1 and referenced in Table D3.3.2-1.	F.1 Take grab samples. AND F.2 Analyze for gross activity.	24 hours AND Once per 24 hours thereafter 24 hours from time of sampling completion

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action B.1 and referenced in Table D3.3.2-1.	G.1.1 Verify one Function 2.a monitor OPERABLE. AND	Immediately
	G.1.2 Monitor recombiner exhaust temperature. OR	Immediately
	G.2.1.1 Collect gas sample.	24 hours AND Once per 24 hours thereafter
	AND	
	G.2.1.2 Analyze gas sample.	4 hours from time of sampling completion
	AND	
	G.2.2.1 Verify one Function 2.a monitor OPERABLE. OR	Immediately
	G.2.2.2 Monitor recombiner exhaust temperature.	Immediately
H. Required Action and associated Completion Time for Condition G not met.	H.1 Discontinue operation of the augmented offgas treatment system.	Immediately

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
I.	As required by Required Action B.1 and referenced in Table D3.3.2-1.	I.1	--------NOTE------- When the primary monitoring system is inoperable and the backup system is in service, sampling may be discontinued for up to 30 minutes only for changing particulate filters and iodine cartridges. ------------------- Continuously collect samples with auxiliary sampling equipment as required in Table D3.2.3-1.	4 Hours
		OR		
		I.2.1	If auxiliary sampling equipment cannot be established within the specified completion time, enter the problem into the Corrective Action Program to evaluate particulate and iodine effluent releases. AND	Immediately
		I.2.2	Report this event in the Radioactive Effluent Release Report.	In accordance with the Radioactive Effluent Release Report Frequency
J.	Required Action and associated Completion Time for Condition E, F, or I not met.	J.1	Discontinue effluent releases via this pathway.	Immediately

ACTIONS

	CONDITION		REQUIRED ACTION	COMPLETION TIME
K.	Function 1.a trip capability not maintained. AND Radiation level exceeds 1.0 ci/sec (prior to 30 min. delay line) for > 15 consecutive minutes.	K.1 AND K.2 AND K.3	Close the offgas isolation valve. Initiate reactor shutdown. Be in MODE 4.	Immediately Immediately 24 hours

SURVEILLANCE REQUIREMENTS

--NOTES--

1.   Refer to Table D3.3.2-1 to determine which DSRs apply for each instrument.

2.   When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function is maintained.

SURVEILLANCE		FREQUENCY
DSR 3.3.2.1	Perform CHANNEL CHECK.	24 hours
DSR 3.3.2.2	Perform CHANNEL CHECK.	7 days
DSR 3.3.2.3	Perform SOURCE CHECK.	31 days
DSR 3.3.2.4	Perform CHANNEL FUNCTIONAL TEST.	31 days
DSR 3.3.2.5	Perform SOURCE CHECK.	92 days
DSR 3.3.2.6	Perform CHANNEL CALIBRATION. The CHANNEL CALIBRATION shall include the use of a standard gas sample containing a percentage of hydrogen to verify accuracy of the monitoring channel in its operating range.	92 days
DSR 3.3.2.7	Perform CHANNEL FUNCTIONAL TEST.	92 days
DSR 3.3.2.8	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if; the instrument indicates measured levels above the alarm/trip setpoint, circuit failure, instrument indicates a downscale failure, or instrument controls not set in operate mode.	92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.3.2.9	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if; the instrument indicates measured levels above the alarm/trip setpoint or circuit failure.	92 days
DSR 3.3.2.10	Perform CHANNEL CALIBRATION. For Function 1.a, the time delay setting for closure of the steam jet air ejector isolation valves shall be ≤ 15 minutes and trip settings shall correspond to Technical Specification 3.7.5.	18 months
DSR 3.3.2.11	Perform CHANNEL FUNCTIONAL TEST. 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, circuit failure, instrument indicates a downscale failure, or instrument controls not set in operate mode.	18 months
DSR 3.3.2.12	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

Table D3.3.2-1 (Page 1 of 3)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIAL CONDITIONS	MINIMUM CHANNELS OPERABLE	CONDITION REFERENCED FROM ACTION B.1	SURVEILLANCE REQUIREMENTS
1. Steam Jet Air Ejector				
a. Noble Gas Activity Monitor	(a)	1(e)	C	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.8 DSR 3.3.2.10 DSR 3.3.2.11 DSR 3.3.2.12
b. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
2. Augmented Offgas Treatment System Explosive Gas Monitoring System				
a. Hydrogen Monitor (2% monitor)	(c)	2	G	DSR 3.3.2.1 DSR 3.3.2.4 DSR 3.3.2.6
3. Reactor Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
f. Isolation Monitor	(d)	(d)	(d)	DSR 3.3.2.5 DSR 3.3.2.11

(continued)

(a) During operation of the steam jet air ejector

(b) During releases via this pathway

(c) During augmented offgas treatment system operation

(d) See Technical Specification 3.3.6.2

(e) Second channel must either be OPERABLE or be in the tripped condition.

(f) A channel may be removed from service for up to 30 minutes for changing particulate filters or iodine cartridges or for low flow alarm check without entering Conditions or Required Actions.

Table D3.3.2-1 (Page 2 of 3)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIAL CONDITION	MINIMUM CHANNELS OPERABLE	CONDITION REFERENCED FROM ACTION B.1	SURVEILLANCE REQUIREMENTS
4. Elevated Release Point Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
5. Radwaste Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
6. Turbine Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2

(continued)

(b) During releases via this pathway

(f) A channel may be removed from service for up to 30 minutes for changing particulate filters or iodine cartridges or for low flow alarm check without entering Conditions or Required Actions.

Table D3.3.2-1 (Page 3 of 3)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIAL CONDITION	MINIMUM CHANNELS OPERABLE	CONDITION REFERENCED FROM ACTION B.1	SURVEILLANCE REQUIREMENTS	
6. (continued)					
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2	|
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10	
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10	
7. Multi Purpose Facility (MPF) Building Ventilation Monitoring System					
a. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2	|
b. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2	|
c. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10	
d. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10	

(b) During releases via this pathway

(f) A channel may be removed from service for up to 30 minutes for changing particulate filters or iodine cartridges or for low flow alarm check without entering Conditions or Required Actions.

D 3.4 LIQUID/GASEOUS DOSE

D 3.4.1 Liquid/Gaseous Effluents Dose

DLCO 3.4.1   The dose or dose commitment to an actual Member of the Public due to radiation and radioactive releases from Cooper Station shall be limited to ≤ 75 mrem to the thyroid and ≤ 25 mrem to the total body or any other body organ during a calender year.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Estimated dose or dose commitment due to radiation and radioactive releases exceeds limits.	A.1 Verify the condition resulting in doses exceeding these limits is corrected.	Immediately
B. Required Action and associated Completion Time not met.	B.1 ----------------NOTE-------------- This is the Special Report required by D 3.1.2, D 3.1.3, or D 3.2.3 supplemented with the following. ------------------------------------ Submit a Special Report pursuant to Specification D 5.4, including information specified in 40 CFR Part 190.11(b). This submission shall be deemed a timely request for variance in accord with provisions of 40 CFR Part 190. The variance is granted until NRC staff action on the item is complete.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
DSR 3.4.1.1 Perform a cumulative dose calculation due to radioactive material in gaseous and liquid effluents to determine compliance with DLCO 3.4.1.	12 months

D 3.5 SOLID RADIOACTIVE WASTE

D 3.5.1 Solid Radioactive Waste

DLCO 3.5.1   The appropriate equipment of the solid radwaste system shall be OPERABLE to process radioactive waste containing liquid and liquid destined for disposal subject to 10 CFR Part 61 to a form that meets applicable requirements of 10 CFR Part 61.56 before the waste is shipped from the site.

APPLICABILITY: During solid radwaste processing.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Container of waste does not comply with 10 CFR Part 61.56.	A.1 Suspend delivery to a carrier for transport.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.5.1.1	Sample and analyze the dewatered radioactive waste for pH.	Prior to solidification of every 10th batch of dewatered waste.
DSR 3.5.1.2	Inspect solidified or dewatered radioactive waste to insure that there is no free standing liquid on top of the solid waste.	Prior to capping each drum or High Integrity Container (HIC)
DSR 3.5.1.3	Record the following information for radioactive solid waste shipped offsite during the report period for the Radioactive Effluent Release Report per the Reporting Requirements in Technical Specification 5.6.3; a) Container burial volume, b) Total curie quantity (determined by measurement or estimate), c) Principal gamma radionuclides (determined by measurement or estimate), d) Type of waste, and e) Solidification agent.	In accordance with 10 CFR 50.36a

D 4.0 MONITORING PROGRAM

D 4.1 Monitoring Program Compliance

DLCO 4.1   The radiological environmental monitoring program shall be conducted as specified in Table D4.1-1, using analytical techniques such that the detection capabilities in Table D4.1-2 are achieved.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radiological environmental monitoring program not conducted as specified in Table D4.1-1.	A.1 Prepare and submit to the NRC in the Annual Radiological Environmental Report the reasons for not conducting the program in accordance with Table D4.1-1 and the plans for preventing recurrence.	May 15th following the end of the year
B. Environmental sampling medium is not available from a sampling location as specified in Table D4.1-1.	B.1 Report in the Annual Radiological Environmental Report the cause and location where replacement samples were obtained.	May 15th following the end of the year

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
DSR 4.1.1	Perform radiological environmental sampling and analysis.	In accordance with Table D4.1-1
DSR 4.1.2	Conduct a land use census to identify the location of the nearest garden that is greater than 500 square feet in area and that yields edible leafy vegetables, the location of the nearest milk animal, and the location of the nearest resident in each of the 16 meteorological sectors within 3 miles of the Station.	12 months
DSR 4.1.3	Summarize results of radiological environmental analysis in the Annual Radiological Environmental Report.	May 15th following the end of the year
DSR 4.1.4	Submit results of the land use census in the Annual Radiological Environmental Report.	May 15th following the end of the year

Table D4.1-1 (Page 1 of 2)
Radiological Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLE STATIONS	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Airborne			
a. Radioiodine and Particulate	At least 5 locations	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.	Radioiodine canister: Analyze at least once per 7 days for I-131. Particulate sample: Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic[a] analysis on each sample in which gross beta activity is >10 times the yearly mean of control samples. Perform gamma isotopic[a] analysis on composite (by location) sample at least once per 92 days.
2. Direct Radiation	At least 32 locations	Thermoluminescent Dosimeters (TLD)[b] exchange and read-out at least once per 92 days.	Gamma dose: At least once per 92 days.
3. Waterborne			
a. River Water	At least 2 locations	Collect a one (1) gallon grab sample at least once per 31 days.	Gamma isotopic[a] analysis of each sample. Composite grab sample for tritium analysis at least once per 92 days.
b. Ground Water	At least 2 locations	Collect a one (1) gallon grab sample at least once per 92 days.	Gamma isotopic[a] and tritium analysis of each sample.
c. Sediment from Shoreline	At least 1 location	Two (2) times a year, once in the spring and once in the fall.	Gamma isotopic[a] analysis of each sample.
4. Ingestion			
a. Milk (nearest producer)	At least 1 location	At least once per 15 days during Peak Pasture Period[c]; at least once per 31 days at other times.	Gamma isotopic[a] and I-131 analysis of each sample.
b. Milk (other producers)	At least 2 locations	At least once per 92 days.	Gamma isotopic[a] and I-131 analysis of each sample.
c. Fish	At least 2 locations	Two times per year (once in the summer and once in the fall). Attempt to include the following: 1. Bottom feeding species 2. Middle-Top feeding species	Gamma isotopic[a] analysis on edible portions.

(continued)

Table D4.1-1 (Page 2 of 2)
Radiological Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLE STATIONS	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. (continued)			
d. Food Products	Samples of three different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available.	Gamma isotopic[a] and I-131 analysis
	One sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic[a] and I-131 analysis

(a) Gamma isotopic analysis refers to high resolution gamma spectrum analysis as follows: the sample is scanned for gamma-ray activity. If no activity is found for a selected nuclide, the detection sensitivity for that nuclide will be calculated using the counting time, detector efficiency, gamma energy, geometry, and detector background appropriate to the particular sample in question. The following nineteen (19) nuclides shall be analyzed routinely:

BaLa-140	Cs-137	Ra-226
Be-7	Fe-59	Ru-103
Ce-141	I-131	Ru-106
Ce-144	K-40	Th-228
Co-58	Nb-95	Zn-65
Co-60	Mn-54	Zr-95
Cs-134		

Any radionuclide detected, i.e., having a measured concentration greater than the LLD, whether or not it is one of the 19 nuclides listed above, shall be regarded as present in the sample.

(b) Thermoluminescent Dosimeters (TLD) is a single phosphore. Two or more phosphores in one package are considered to be two or more dosimeters.

(c) Peak Pasture Period is June 1 through September 30 of each year.

Table D4.1-2 (Page 1 of 2)
Detection Capabilities for Environmental Sample Analysis

	LOWER LIMIT OF DETECTION[a] (LLD) [b]					
ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GAS (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
gross beta	4	1×10^{-2}				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58	15		130			
Co-60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	1 [c]	7×10^{-2}		1	60	
Cs-134	15	5×10^{-2}	130	15	60	150
Cs-137	18	6×10^{-2}	150	18	80	180
Ba-140	60			60		
La-140	15			15		

(a) This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

Table D4.1-2 (Page 2 of 2)
Detection Capabilities for Environmental Sample Analysis

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

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

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

Where:

LLD is the "a priori" lower limit of detection as described above (as picocurie 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)

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

Δt is the elapsed time between sample collection (or midpoint of the sample collection period) and time of counting.

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples).

Analysis shall be performed in such a manner that the stated LLD's will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLD's unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Report.

(c) LLD for drinking water.

D 4.0 MONITORING PROGRAM

D 4.2 Monitoring Program Concentration

DLCO 4.2   Radioactivity concentrations in sampled medium from the radiological
                  environmental monitoring program shall not exceed values specified in
                  Table D5.4-1 when averaged over a calender quarter.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radioactivity concentrations of sampled medium from the radiological environmental monitoring program exceeds values specified in Table D5.4-1, averaged over a calender quarter which is attributable to release(s) from the Station.	A.1 Prepare and submit to the NRC a Special Report in accordance with Specification D 5.4 which includes an evaluation of any release conditions, environmental factors or other conditions which caused the value(s) to be exceeded.	31 days following the end of the quarter
B. Radioactivity concentrations of sampled medium from the radiological environmental monitoring program exceeds values specified in Table D5.4-1, averaged over a calender quarter which is not attributable to release(s) from the Station.	B.1 Report and explain in the Annual Radiological Environmental Report the results of the sample(s).	May 15th following the end of the year

D 4.0 MONITORING PROGRAM

D 4.3 Monitoring Program Dose

DLCO 4.3   The calculated personal dose associated with sampled exposure pathway(s) shall not exceed 120% of the calculated dose at the maximum dose location associated with like pathways at a location where sampling is conducted as specified in Table D4.1-1.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Location(s) identified at which the calculated dose associated with the exposure pathway(s) exceeds 120% of the calculated dose at the maximum dose location associated with like pathways at a location where sampling is conducted as specified in Table D4.1-1.	A.1 ------------NOTE----------- Only applicable if samples are reasonably attainable at the new location. ------------------------------- Add new sampling location(s) identified having maximum exposure potential to the radiological environmental monitoring program and Table D4.1-1. AND	92 days
	A.2 Describe change made to Table D4.1-1 in the Annual Radiological Environmental Report.	May 15th following the end of the year

D 5.0 MISCELLANEOUS PROGRAMS/REPORTS

D 5.1 Interlaboratory Comparison Program

DLCO 5.1   Analyses shall be performed on radioactive materials supplied as part of the Interlaboratory Comparison Program.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Analyses not performed.	A.1 Report to the NRC in the Annual Radiological Environmental Report the corrective actions taken to prevent recurrence.	May 15th following the end of the year

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
DSR 5.1.1 Submit a brief summary of the results obtained from the Interlaboratory Comparison Program in the Annual Radiological Environmental Report pursuant to Technical Specification 5.6.2.	May 15th following the end of the year

D 5.0 MISCELLANEOUS PROGRAMS/REPORTS

D 5.2 Annual Radiological Environmental Report

The Annual Radiological Environmental Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Assessment Manual (ODAM), and in 10 CFR50, Appendix I, Section IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODAM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in Regulatory Guide 4.8, December 1975. 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 in a supplementary report as soon as possible.

Details of the report shall include the following:

a. A summary of doses to a Member of the Public beyond the Site and Exclusion Area Boundary due to Cooper Nuclear Station aqueous and airborne radioactive effluents, calculated in accordance with methods compatible with the ODAM.

b. A summary of the results of the land use census required in DSR 4.1.2.

c. Summarized and tabulated results in the format of Table D5.2-1 of analysis of samples required by the radiological environmental monitoring program, and taken during the report period.

d. A summary description of the radiological environmental monitoring program including any changes; a map of all sampling locations keyed to a table giving distances and directions from the reactor; and, the results of participation in the Interlaboratory Comparison Program, required by D 5.1.

e. Summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period.

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility ____Cooper Nuclear Station____ Docket No. ____50-298____

Location of Facility ____Nemaha, Nebraska____ Reporting Period _____
(County, State)

Medium of Pathways Sampled (Unit of Measurement)	Type & Total No. of Analyses Performed	Lower Limit of Detection(1) (LLD)	All Indicator Locations Mean[] (2) Range (2)	Location with Highest Annual Mean		Control Locations Mean[] (2) Range (2)	No. of Reportable Occurrences
				Name Distance & Direction	Mean[] (2) Range (2)		

Table Notes:

(1) Nominal Lower Limit of Detection (LLD).
(2) Mean and Range based upon detectable measurements only. Fraction of detectable measurements at specified location
     indicated in brackets [].

D 5.2-2

D 5.0 MISCELLANEOUS PROGRAMS/REPORTS

D 5.3 Radioactive Effluent Release Report

The Radioactive Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODAM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

The Radioactive Effluent Release Report shall be submitted to the NRC by May 1 of each year and shall include the following:

a. A summary by calendar quarter of the quantities of radioactive liquid and gaseous effluents released from the Station, reported in the format recommended in Regulatory Guide 1.21, Appendix B, Tables 1 and 2.

b. A summary of radioactive solid waste shipped from CNS, including information provided in DSR 3.5.1.3.

c. A summary of meteorological data collected during the year.

d. A list and brief description of each unplanned release of gaseous or liquid radioactive effluent that causes a limit in DLCO 3.1.1, DLCO 3.1.3, DLCO 3.2.1, DLCO 3.2.2 or DLCO 3.2.3 to be exceeded.

e. Calculated offsite dose to humans resulting from the release of effluents and their subsequent dispersion on the atmosphere reported in accordance with Regulatory Guide 1.21.

f. A summary of changes made to the CNS Process Control Program. The summary shall contain sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information; a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and documentation of the fact that the change has been reviewed and found acceptable by the SORC.

D 5.3 Radioactive Effluent Release Report (continued)

g. Changes made to the CNS Offsite Dose Assessment Manual shall be submitted
to the NRC in the form of a complete, legible copy of the entire ODAM as part of,
or concurrent with the Radioactive Effluent Release Report for the period of the
report in which any change in the ODAM was made. Each change shall be
identified by markings in the margin of the affected pages, clearly indicating the
area of the page that was changed, and shall indicate the date (i.e., month and
year) the change was implemented.

D 5.0 MISCELLANEOUS PROGRAMS/REPORTS

D 5.4 Special Reports

Special reports shall be submitted to the Director, Nuclear Reactor Regulation, USNRC, Washington, D.C. 20555 and to the NRC Regional Administrator within the time period specified for each report.

Special reports (in lieu of Licensee Event Reports) may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Offsite Dose Assessment Manual.

A special report is required if measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table D5.4-1 when averaged over any calendar quarter sampling period. When more than one of the radionuclides in Table D5.4-1 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Limit Level (1)}} + \frac{\text{Concentration (2)}}{\text{Limit Level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table D5.4-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of DLCO 3.1.3 and 3.2.3. This 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 Report.

Table D5.4-1
Reporting Levels for Radioactivity Concentrations in Environmental Samples

Reporting Levels

ANALYSIS	WATER (pCi/1)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/Kg, Wet)	MILK (pCi/1)	FOOD PRODUCTS (pCi/Kg, Wet)
H-3	2E + 4(a) 3E + 4(c)				
Mn-54	1E + 3		3E + 4		
Fe-59	4E + 2		1E + 4		
Co-58	1E + 3		3E + 4		
Co-60	3E + 2		1E + 4		
Zn-65	3E + 2		2E + 4		
Zr-Nb-95	4E + 2(b)				
I-131	2	0.9		3	1E + 2
Cs-134	30	10	1E + 3	60	1E + 3
Cs-137	50	20	2E + 3	70	2E + 3
Ba-La-140	2E + 2(b)			3E + 2(b)	

(a) For drinking water samples. This is the 40 CFR 141 value.

(b) Concentration of parent or daughter.

(c) For samples of water not used as a source of drinking water.

D 5.0 MISCELLANEOUS PROGRAMS/REPORTS

D 5.5 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous, and Solid)

The radioactive waste treatment systems (liquid, gaseous, and solid) are those systems described in the facility Safety Analysis Report and amendments thereto, which are used to maintain that control over radioactive materials in gaseous and liquid effluents and in solid waste packaged for offsite shipment required to meet the DLCO's set forth in Specifications D 3.1.1, D 3.1.2, D 3.1.3, D 3.1.4, D 3.2.1, D 3.2.2, D 3.2.3, D 3.2.4, D 3.2.5, D 3.2.6, D 3.2.7, D 3.3.2, D 3.4.1, and D 3.5.1. The NRC is notified of major changes to these systems under the provisions of 10 CFR Part 50.59 and Part 50.71 (USAR revisions).

OFFSITE DOSE ASSESSMENT MANUAL
APPENDIX D

TABLE OF CONTENTS - BASES

B 3.0 ODAM Specification Applicability

BASES

DLCOs 3.0.1, 3.0.2, and 3.0.5, and DSRs 3.0.1 and 3.0.2 reflect parallel requirements in the Technical Specifications. Refer to the Technical Specification Bases for appropriate discussion.

DSR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been complete within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with DSR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, DSR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

DSR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for DSRs is expected to be an infrequent occurrence. Use of the delay period established by DSR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable DLCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable DLCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with DSR 3.0.1.

(continued)

BASES
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(continued)

ODAM Specification DLCO 3.0.3, in lieu of imposing a plant shutdown as paralleled in Technical Specification 3.0.3, requires: (a) an Action to initiate efforts to restore compliance with the ODAM or associated Actions; and (b) an Action that requires entering the circumstances into the Corrective Action Program (CAP). These actions ensure that the appropriate actions continue to be focused on and that the circumstances concerning failure to comply with the ODAM Actions would be reviewed. This review will be conducted in accordance with the procedural guidance for CAP Notifications.

There are no ODAM 3.0 Specifications that parallel Technical Specification LCO 3.0.4 or SR 3.0.4. Restrictions in entering MODES or other specified conditions in the Applicability have historically not been applied to ODAM Specifications. There are also no ODAM 3.0 Specifications that parallel Technical Specification LCO 3.0.6 and LCO 3.0.7, which allow for exceptions and revisions of other Technical Specifications. They are not applicable to the ODAM since it is not permitted to allow the ODAM to revise a Technical Specification.

(*Note that there currently are no identified ODAM DLCOs that support Technical Specification systems; however, this discussion is presented to address the philosophy that would be applied.*) An allowance similar to Technical Specification LCO 3.0.6 does not apply to the ODAM. When a Technical Specification supported system LCO is discovered to be not met solely due to a ODAM support system DLCO not met, appropriate Technical Specification ACTIONS are required to be entered immediately. This applies even in instances where the ODAM contains a delay prior to declaring a Technical Specification supported system inoperable. In this case, certain ODAM inoperabilities may not directly impact the OPERABILITY of the Technical Specification supported system and delayed declaration of inoperability of the supported system is acceptable. In other cases, discovered support system inoperabilities that directly result in supported system inability to perform the safety function, should result in immediate declaration of inoperability of the supported system.

Technical Specification LCO 3.0.7 has no parallel in the ODAM since it provides for explicit changes to specified Technical Specifications by the Section 3.10 Specifications. However, in the event that LCO 3.0.7 provides for changes to the Technical Specification MODE definitions by the Section 3.10 Specifications, the revised MODE definitions apply to all plant references, including ODAM references.

B 3.1 LIQUID EFFLUENTS

B 3.1.1 Liquid Effluents Concentration

BASES

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20.1302. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section IV.A guides on technical specifications in Appendix I. 10 CFR Part 50. for an individual and (2) the limits of 10 CFR Part 20.1301 and 20.1302(b)(2)(i) to the population. 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.

Since Service Water is not a normal or expected source of significant radioactive release. routine sampling and monitoring for radioactivity is precautionary. An activity concentration of 3×10^{-6} μCi/ml in Service Water effluent is diluted in the discharge canal to about 1.5% of the 10 CFR 20 Appendix B Table 2 Column 2 concentration with only one circulating water pump operating. During normal Station operation the dilution would be even greater. By monitoring Service Water effluent continuously for radioactivity and by confirmatory sampling weekly. reasonable assurance that its activity concentration can be kept to a small fraction of the 10 CFR Part 20.1302 limit and within the Specification D 3.1.3 limit is provided.

By monitoring Service Water continuously and liquid radwaste continuously during discharge with the monitor set to alarm or trip before the limit specified in 10 CFR 20.1302 is exceeded. reasonable assurance of compliance with Specification D 3.1.1 is provided. Verification that radioactivity in liquid effluent averaged only a small fraction of the concentration limit is provided by calculations demonstrating compliance with Specification D 3.1.3.

Compliance with 10 CFR Part 20.1302(b)(2)(i) implies that the concentration limit represented by 10 CFR Part 20. Appendix B. Table 2 will be met within a suitable and reasonable averaging time for assessing compliance. That averaging time is dependent upon the resolving time of the measurements or estimates which are used to evaluate compliance. Assessment of compliance is done by sampling and analysis according to DSR 3.1.1.2. by estimating or measuring the maximum release flow and the minimum dilution flow coincident during the period of release represented by the sample. and by computing the concentration as a fraction of the limit beyond the site and exclusion area boundary periodically on the basis of these data.

Reporting by Special Reports and other reports required by the ODAM and Section 5.6 of Technical Specifications is used in lieu of reporting per 10 CFR 50.73.

B 3.1 LIQUID EFFLUENTS

B 3.1.2 Liquid Waste Concentration

BASES

Specification D 3.1.2 implements the requirements of 10 CFR Part 50.36a(a)(1) that operating procedures be established and followed and that equipment be maintained and used to keep releases to the environment as low as is reasonably achievable. The OPERABILITY of the liquid radwaste treatment system ensures that the appropriate portions will be available for use whenever liquid effluents require treatment prior to release to the environment. The specification that the portions of the system which were used to establish compliance with the design objectives in 10 CFR Part 50. Appendix I. Section II be used when specified provides reasonable assurance that releases of radioactive material in liquid effluent will be kept as low as is reasonably achievable. The activity concentration, 0.01 μCi/ml. below which liquid radwaste treatment would not be cost beneficial. and therefore not required. is demonstrated below:

> The quantity of radioactive material in liquid effluent released annually from Cooper Station has been calculated to be[1]

total iodines	3.65 curies
total others (less H^3)	0.7
	total 4.35 curies

The population dose commitment resulting from the radioactive material in liquid effluent released annually has been calculated to be

thyroid	1.95 manrem
total body	0.56
	total 2.5 manrem

Therefore. population doses are about 0.5 manrem per curie of iodine released and about 0.8 manrem per curie of other radionuclides (less H^3) released in liquids. It would be conservative to assume one manrem committed per curie released in liquid effluent.

The volume of liquid waste processed and intended for discharge is estimated to be:

Low Purity Waste	5700 gal/day	1.8×10^6 gal/yr
Chem Waste + Demin Regenerant Waste	4000 gal/day	1.2×10^6 gal/yr

<div align="right">(continued)</div>

BASES

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The annual costs to operate the radwaste processing equipment, neglecting credit for capital recovery, are estimated according to Regulatory Guide 1.110 to be:

> Dirty Waste Ionex $ 88,000/yr
> Evaporator $114,000/yr

Unit volume operating costs are about:

Cost to ion exchanger = $\frac{\$ 88,000}{1.8E+6 \text{ gal}}$ = $0.05/gal

Cost to evaporate = $\frac{\$114,000}{1.2E+6 \text{ gal}}$ = $0.10/gal

Assuming the cost-benefit balance is $1,000 expenditure per manrem reduction and assuming treatment removes all radioactivity from the liquid, then

(1) the activity concentration in a batch below which treatment is not cost-beneficial is

$$C = \frac{\$88,000}{1.8E+6 \text{ gal} \times 3785 \frac{\text{ml}}{\text{gal}}} \times \frac{1 \text{ curie}}{\text{manrem}} \times \frac{10^6 \ \mu Ci}{\text{curie}} \times \frac{1 \text{ manrem}}{\$1,000}$$

C = 0.013 $\mu Ci/ml$

(2) the activity concentration below which evaporation is not costbeneficial is

$$C = \frac{\$114,000}{1.2E+6 \text{ gal} \times 3785 \frac{\text{ml}}{\text{gal}}} \times \frac{1 \text{ curie}}{\text{manrem}} \times \frac{10^6 \ \mu Ci}{\text{curie}} \times \frac{1 \text{ manrem}}{\$1,000}$$

C = 0.025 $\mu Ci/ml$

Therefore, to one significant digit, radwaste treatment of liquids containing less than 0.01 $\mu Ci/ml$ is not justified.

BASES

(continued)

[1]Demonstration of Compliance with 10 CFR 50 Appendix I, Revision 1 and Supplement 2. Nebraska Public Power District, Cooper Nuclear Station, January 9, 1978.

B 3.1 LIQUID EFFLUENTS

B 3.1.3 Liquid Effluents Dose

BASES

Note: The Bases discussion refers to "technical specifications" and quotes the Staff's use of "technical specifications." The statements and opinions pre-date Generic Letter 89-01. Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the relocation of procedural details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program.

Generic Letter 89-01 provides the guidance and justification for relocation of these "technical specifications" to the Offsite Dose Assessment (ODAM) Manual and the Process Control Program (PCP). Therefore, "technical specifications" as used in this Bases refers to ODAM Specifications.

Specifications D 3.1.3, D 3.2.2 and D 3.2.3 implement the requirements of 10 CFR Part 50.36a and of 10 CFR Part 50, Appendix I, Section IV. These specifications state ODAM LIMITING CONDITIONS FOR OPERATION (DLCO) to keep levels of radioactive materials in LWR effluents as low as is reasonably achievable. Compliance with these specifications will also keep average releases of radioactive material in effluents at small percentages of the limits specified in 10 CFR Part 20.1301. Surveillance Requirements provide for the measurement of releases and calculation of doses to verify compliance with the Specifications. Action statements in these Specifications implement the requirements of 10 CFR Part 50.36(c)(2) and 10 CFR Part 50, Appendix I, Section IV.A in the event an LCO is not met. Annual dose limitations stated in Specifications D 3.1.3, D 3.2.2 and D 3.2.3 are not strict limits as used elsewhere in the Technical Specifications (are not an immediate safety concern) but do obligate NPPD to take the applicable Required Action in Specifications D 3.1.3, D 3.2.2 and D 3.2.3.

(continued)

BASES

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10 CFR Part 50 contains two distinctly separate statements of requirements pertaining to effluents from nuclear power reactors. The first concerns a description of equipment to maintain control over radioactive materials in effluents, determination of design objectives, and means to be employed to keep radioactivity in effluents ALARA. This requirement is stated in Part 50, Section 34a and Appendix I, Section II. Appendix I, Section III stipulates that conformance with the guidance on design objectives be demonstrated by calculations (since demonstration is expected to be prospective). The other is a requirement for developing limiting conditions for operation in technical specifications. It is stated in 10 CFR Part 50, Section 36a and Appendix I, Section IV. Both the intent of the Commission and the requirement are clearly stated in the Opinion of the Commission; [1] relevant paragraphs from that document follow:

> Section 50.36a(b) of 10 CFR Part 50 provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications which take into account the need for operating flexibility and at the same time ensure that the licensee will exert his best efforts to keep levels of radioactive materials in effluents as low as practicable. The Appendix I that we adopt provides more specific guidance to licensees in this respect.

A. The Rule

> Section IV of Appendix I specifies action levels for the licensee. If, for any individual light water cooled nuclear power reactor, the quantity of radioactive material actually released in effluents to unrestricted areas during any calendar quarter is such as to cause radiation exposure, calculated on the same basis as the design objective exposure, which would exceed one-half the annual design objective exposure, the licensee shall make an investigation to identify the causes of these high release rates, define and initiate a program of action to correct the situation, and report these actions to the Commission within 30 days of the end of the calendar quarter.

The conclusion of the NRC Staff in the Appendix I Rulemaking Hearing [2] agrees with that of the Commission. The Staff recommended, "...that the limiting conditions for operation described in Appendix I, Section IV be applicable upon publication to technical specifications included in any license authorizing operation of a light water cooled nuclear power reactor..." (p. 73).

BASES

(continued)

The action to be taken by a licensee in the event a limiting condition is exceeded, is stated in Appendix I, Section IV.A and in the Opinion of the Commission. [3] ODAM Specifications D 3.1.3, D 3.2.2, D 3.2.3 and Surveillances DSR 3.1.3.1, 3.1.3.2, 3.2.2.1, 3.2.3.1 and 3.2.3.2 for Cooper Station conform to this requirement.

Guidance for developing limiting conditions for operation for surveillance and monitoring is included in Appendix I, Section IV.B.

Although "it is expected that the annual releases of radioactive material in effluents from light water cooled nuclear power reactors can generally be maintained within the levels set forth as numerical guides for design objectives in Section II" (Appendix I, Section IV), no recommendation was made by either the Staff in its Concluding Statement [4] or by the Commission in its Opinion [5] that design objective values should appear as technical specification limits. The Opinion of the Commission and the statement of Appendix I are clear. Limiting conditions of operation (LCO) related to the quantity of radioactive material in effluents released to an unrestricted area stated in technical specifications shall conform to Appendix I, Section IV.A. Licensee action in the event an LCO is exceeded should be in accord with Section IV.A. Finally, surveillance and monitoring of effluents and the environment should conform to Section IV.B.

With the implementation of Specification D 3.1.3 and Surveillances DSR 3.1.3.1 and 3.1.3.2, there is reasonable assurance that Station operation will not cause a radionuclide concentration in public drinking water taken from the River that exceeds the standard for anthropogenic radioactivity in community drinking water.

[1] NRC Commissioners, "Opinion of the Commission," in the Appendix I Rulemaking hearing, Docket Rm 502, p. 101-102, April 30, 1975.

[2] NRC Staff, "Concluding Statement of the Regulatory Staff," in the Appendix I Rule- making Hearing, Docket RM 502, pp. 17, 69, 73, 115, February, 1974.

[3] NRC Commissioners, p. 101

[4] NRC Staff, op. cit.

[5] NRC Commissioners, op. cit.

(continued)

BASES

(continued)

[6]Generic Letter 89-01, Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the relocation of procedural details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program.

B 3.1 LIQUID EFFLUENTS

B 3.1.4 Outside Temporary Storage of Radioactive Liquid

BASES

Custom Technical Specifications Bases did not exist.

B 3.2 GASEOUS EFFLUENTS

B 3.2.1 Gaseous Effluents Concentration

BASES

DLCO 3.2.1(a) is included to assure that a measure of control is provided over the concentration of radionuclides in air leaving the exclusion area. Radioactive noble gases are monitored by instruments that provide a measure of release rate and cause automatic alarm when the noble gas concentration beyond the Site and Exclusion Area Boundary is expected to exceed the dose rate specified in DLCO 3.2.1(a). With prompt action to reduce the radioactive noble gas concentration in effluent following alarm initiation, it can be maintained at a small fraction of the annual limit. The specified release rate limits restrict the corresponding gamma and beta dose rates above background to an individual at or beyond the exclusion area boundary to $\leq$ 500 mrem/year to the total body or to $\leq$ 3000 mrem/year to the skin.

Radioiodines and radionuclides in particulate form are sampled with integrating samplers that permit assessment of the average release rate during each sample collection period. By complying with DLCO 3.2.2 and 3.2.3 the average concentration beyond the Site and Exclusion Area Boundary will be maintained at a small fraction of the 10 CFR Part 20.1302(b)(2)(i) concentration limit.

B 3.2 GASEOUS EFFLUENTS

B 3.2.2 Noble Gases Dose

BASES

Assessments of dose required by Surveillances DSR 3.2.2.1 and DSR 3.2.3.2 to verify compliance with Appendix I, Section IV is based on measured radioactivity in gaseous effluent and on calculational methods stated in the ODAM. Pathways of exposure and location of individuals are selected such that the dose to a nearby resident is unlikely to be underestimated. Dose assessment methodology described in the ODAM for gaseous effluent will be consistent with the methodology in Regulatory Guides 1.109 and 1.111. Cumulative and projected assessments of dose made during a quarter are based on historical average, or reference (the same period of record used in the design objective Appendix I evaluation) atmospheric conditions. Assessments made for the Annual Radiological Environmental Report | will be based on quarterly and annual averages of atmospheric conditions during the period of release.

The bases for Specification D 3.2.2 and Surveillance DSR 3.2.2.1 are also discussed in the bases for Specification D 3.1.3 and Surveillances DSR 3.1.3.1 and 3.1.3.2.

B 3.2 GASEOUS EFFLUENTS

B 3.2.3 Iodine and Particulates

BASES

This bases for Specification D 3.2.3 and Surveillances DSR 3.2.3.1 and 3.2.3.2 are discussed in the bases for Specification D 3.1.3 and Surveillances DSR 3.1.3.1 and 3.1.3.2.

B 3.2 GASEOUS EFFLUENT

B 3.2.4 Offgas Treatment System

BASES

The OPERABILITY of the gaseous radwaste treatment system ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system 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 specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of this system are 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.

B 3.2 GASEOUS EFFLUENTS

B 3.2.5 Exhaust Ventilation Treatment Systems

BASES

An Exhaust Ventilation Treatment System (EVTS) is a system intended to remove radioiodine or radioactive material in particulate form from gaseous effluent by passing exhaust ventilation air through charcoal absorbers and/or HEPA filters before exhausting the air to the environment. An EVTS is not intended to affect noble gas in gaseous effluent. Engineered Safety Feature (ESF) gaseous treatment systems are not considered to be EVTS. The Standby Gas Treatment System is an ESF and not an EVTS. EVTS are specifically identified in ODAM Figure 3-1.

The OPERABILITY of the exhaust ventilation treatment systems ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement 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 specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the system are 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.

B 3.2   GASEOUS EFFLUENTS

B 3.2.6   Hydrogen Concentration

BASES

This specification is provided to ensure that the concentration of
potentially explosive gas mixtures contained in the waste gas treatment system
is maintained below the flammability limits of hydrogen and oxygen. While the
Augmented Treatment System is in service the hydrogen and oxygen
concentrations are prevented from reaching the flammability limits.
Maintaining the concentration of hydrogen below its flammability limit
provides assurance that the releases of radioactive materials will be
controlled in conformance with the requirements of General Design Criterion 60
of Appendix A to 10 CFR 50.

For this specification, reporting by Special Reports and the other reports
required by the ODAM and Section 5.6 of Technical Specifications is used in
lieu of reporting per 10 CFR 50.73.

B 3.2 GASEOUS RELEASES

B 3.2.7 Primary Containment Venting and Purging

BASES

This specification provides reasonable assurance that releases of Iodine from drywell purging during power operations, and during startup while performing primary containment inerting within 24 hours after shutdown will not be excessively large, particularly due to Iodine spiking. The exemptions to using the SBGT system are intended to minimize the time the SBGT system is on line while coolant temperature is greater than 200°F, hence to decrease the probability of damage to the SBGT filters that could occur from overpressurization due to a LOCA and the main purge and vent valves open.

For this specification, reporting by Special Reports and the other reports required by the ODAM and Section 5.6 of Technical Specifications is used in lieu of reporting per 10 CFR 50.73.

B 3.3 INSTRUMENTATION

B 3.3.1 Liquid Effluent Monitoring

BASES

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the release of radioactive material in liquid effluents. The OPERABILITY and use of these instruments implements the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64. The alarm and/or trip setpoints for these instruments are calculated in the manner described in the ODAM to assure that the alarm and/or trip will occur before the limit specified in 10 CFR Part 20.1302 is exceeded. Control of the normal liquid discharge pathway is assured by station procedures governing locked discharge valves and valve line-up verification.

The liquid radwaste monitor assures that all liquid discharged to the discharge canal does not exceed the limits of Specification D 3.1.1. Upon sensing a high discharge level, an isolation signal is generated which closes the radwaste discharge valve. The set point is adjustable to compensate for variable isotopic discharges and dilution flow rates.

For this specification, reporting by Special Reports and the other reports required by the ODAM and Section 5.6 of Technical Specifications is used in lieu of reporting per 10 CFR 50.73.

B 3.3 INSTRUMENTATION

B 3.3.2 Gaseous Effluent Monitoring

BASES

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 of gaseous effluents. The location of this instrumentation is indicated by a Figure in the ODAM (page 48). The simplified flow diagram shows the gaseous effluent treatment and monitoring equipment and defines the delay system (30 minute hold up line) and the treatment system (AOG charcoal beds). The alarm/trip setpoints for these instruments shall be calculated in accordance with methods in the ODAM, which have been reviewed by NRC, to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20. The process monitoring instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the augmented offgas treatment system. 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.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay accounted for by the 30-minute holdup time of the off-gas before it reaches the stack. Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip setting of 1.0 ci/sec (prior to 30 min. delay) provides an improved capability to detect fuel pin cladding failures to allow prevention of serious degradation of fuel pin cladding integrity which might result from plant operation with a misoriented or misloaded fuel assembly. This limit is more restrictive than 0.39 ci/sec noble gas release rate at the air ejectors (after 30 min. delay) which was used as the source term for an accident analysis of the augmented off-gas system. Using the .39 ci/sec source term, the maximum off-site total body dose would be less than the .5 rem limit.

A footnote has been added to Table D3.3.2-1 to indicate that a channel of the Noble Gas Activity Monitor, the Iodine Sampler Cartridge, or the Particulate Sampler Filter, in the ventilation monitoring systems for the Reactor Building, Radwaste Building, Turbine Building, Elevated Release Point, and Multi Purpose Facility, may be removed from service for up to 30 minutes for changing particulate filters or iodine cartridges or for the low flow alarm check without entering the conditions and required actions. The channel will continue to be declared inoperable when removed from service, but the Conditions and Required Actions will not be entered due to the short duration that the channel is inoperable and the facts that the system must be removed from service to perform the required ODAM surveillance and is returned to an OPERABLE condition prior to expiration of the Completion Time for initiating the Required Action.

A note has been added to Required Action I.1 stating that continuous monitoring may be discontinued for up to 30 minutes only for changing particulate filters and iodine cartridges

when the backup monitoring system is in service. Replacement of these filters and cartridges is a normal activity, and not a condition that is appropriate for entry into the Corrective Action Program or for including in the Radioactivity Effluent Release Report. This allowance to discontinue the sampling is acceptable based on (1) four hours are allowed to begin the continuous collection of samples with auxiliary equipment when the primary monitoring system is inoperable, (2) occasions when the primary monitoring system is inoperable and the backup monitoring system is in service are expected to be rare, and (3) the short period of time (30 minutes) that the monitor is allowed to be out of service.

In the event no flow rate measurement device is operable on a gaseous stream, alternative 24-hour estimates are adequate since the system design is constant flow and loss of flow is alarmed in the control room.

For this specification, reporting by Special Reports and the other reports required by the ODAM and Section 5.6 of Technical Specifications is used in lieu of reporting per 10 CFR 50.73.

B 3.4 LIQUID/GASEOUS DOSE

B 3.4.1 Liquid/Gaseous Effluents Dose

BASES

This specification is provided to meet the reporting requirements of 40 CFR Part 190. In the event an analysis is required to determine compliance with 40 CFR 190. the dose to a member of the public due to radiation direct from the station will be estimated with the aid of environmental TLD. PIC. or similar environmental radiation dosimetry. A contribution from another fuel cycle facility is not added since there is no licensed fuel cycle facility within 50 miles of Cooper Station.

B 3.5 SOLID RADIOACTIVE WASTE

B 3.5.1 Solid Radioactive Waste

BASES

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require materials processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

B 4.0 MONITORING PROGRAM

B 4.1 Monitoring Program Compliance

BASES

The radiological environmental monitoring program, including the land use census, is conducted to satisfy the requirements of 10 CFR Part 50, Appendix I, Section IV.B.2 and 3. The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measureable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

The environmental monitoring program described in Table D4.1-1 is the minimum program which will be maintained. The Offsite Dose Assessment Manual(ODAM) describes in detail the actual monitoring program which is performed to ensure compliance with the specified minimum program.

The land use census is conducted annually to identify changes in use of the unrestricted area in order to recommend modifications in monitoring programs for evaluating individual doses from principal exposure pathways.

The need to adjust the program to current conditions and to assure that the integrity of the program is maintained are thereby provided. Restricting the census to gardens of greater than 500 square feet 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 used. 1) that 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/square meter.

B 4.0 MONITORING PROGRAM

B 4.2 Monitoring Program Concentration

BASES

Custom Technical Specifications Bases did not exist.

B 4.0 MONITORING PROGRAM

B 4.3 Monitoring Program Dose

BASES

Like pathways monitored (sampled) at a location, excluding the control station location(s), having the lowest associated calculated personnal dose may be deleted from Table D4.1-1 at the time the new pathway(s) and locations are added.

B 5.0 MISCELLANOUS PROGRAMS/REPORTS

B 5.1 Interlaboratory Comparison Program

BASES

The requirement for participation in a 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 a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

B 5.0 MISCELLANOUS PROGRAMS/REPORTS

B 5.2 Annual Radiological Environmental Report

BASES

Custom Technical Specifications Bases did not exist.

B 5.0 MISCELLANOUS PROGRAMS/REPORTS

B 5.3 Annual Radiological Effluent Release Report

BASES

Custom Technical Specifications Bases did not exist.

B 5.0 MISCELLANOUS PROGRAMS/REPORTS

B 5.4 Special Reports

BASES

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Custom Technical Specifications Bases did not exist.

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B 5.0 MISCELLANOUS PROGRAMS/REPORTS

B 5.5 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous, and Solid)

BASES

Custom Technical Specifications Bases did not exist.