D 3.0 ODAM Specifications Applicability

The ODAM Specifications are subject to Technical Specifications Section 3.0, "Limiting Condition for Operation (LCO) Applicability and Surveillance Requirement (SR) Applicability," with the following exceptions:

1. LCO 3.0.6 is not applicable to ODAM Specifications.

2. LCO 3.0.7 is not applicable to ODAM Specifications.

3. Section 3.0 requirements are not applicable when so stated in notes within individual specifications.

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 X 10^{-7} µCi/ml (i)
				I-131	1 X 10^{-6} µCi/ml
	Grab sample	One batch/ 31 days (a)	31 days (b)	Dissolved and Entrained Gases (gamma emitters)	1 X 10^{-5} µCi/ml
	Proportional Composite of grab samples (f)	Each batch (a)	31 days (b)	H-3	1 X 10^{-5} µCi/ml
				Gross Alpha	1 X 10^{-7} µCi/ml
			92 days (b)	Sr-89	5 X 10^{-8} µCi/ml
				Sr-90	5 X 10^{-8} µCi/ml
				Fe-55	1 X 10^{-6} µCi/ml
Plant Service Water Effluent (d)	Grab Sample	7 days	7 days (b)	Principal Gamma Emitters (j)(k)	5 X 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 X 10^{-7} µCi/ml (i)
				I-131	1 X 10^{-6} µCi/ml
	Grab Sample	31 days	31 days (b)	Dissolved and Entrained Gases (gamma emitters)	1 X 10^{-5} µCi/ml
	Proportional Composite of Grab Samples (g)	24 hours	31 days (b)	H-3	1 X 10^{-5} µCi/ml
				Gross Alpha	1 X 10^{-7} µCi/ml
			92 days (b)	Sr-89	5 X 10^{-8} µCi/ml
				Sr-90	5 X 10^{-8} µCi/ml
				Fe-55	1 X 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 X 10^{-6} µCi/ml, or in the event the plant service water effluent monitor indicates the presence of an activity concentration greater than 3 X 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.

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

(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 $\leq 1 \times 10^{-8}$ 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 ≤ .01 μCi/ml, excluding tritium and noble gases.

APPLICABILITY: At all times.

ACTIONS

-----------------------------------NOTES-----------------------------------

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Concentration of radioactive materials in liquid wastes from pre-release analysis > .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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met. <u>AND</u> Radioactive liquid waste being discharged without treatment in excess of .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

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

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

(continued)

ACTIONS (coninued)

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 ≤ 75 mrem to the thyroid and ≤ 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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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.	Immediately
	<u>AND</u>	
	A.2 Begin measures to reduce content to within the limits of TS 5.5.8.b.	Immediately
	<u>AND</u>	
	A.3 Describe the events leading to the condition in the Radioactive Effluent Release Report.	Prior to submittal of next Radioactive Effluent Release Report

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, ≤ 500 mrem per year to the total body and ≤ 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 ≥ 8 days, ≤ 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 ≤ 3 months and;

2. The dose rate due to other radionuclides is averaged over ≤ 31 days.

APPLICABILITY: At all times.

ACTIONS

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

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, ≤ 5 mrad during any calender quarter and ≤ 10 mrad during any calender year; and

b. For beta radiation, ≤ 10 mrad during any calender quarter and ≤ 20 mrad during any calender year.

APPLICABILITY: At all times.

ACTIONS

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable

2. LCO 3.0.4 is not applicable.

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

(continued)

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

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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.

(continued)

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 X 10^{-10} μCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1 X 10^{-11} μCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1 X 10^{-11} μCi/ml
				Sr-90	1 X 10^{-11} μCi/ml
				Gross Alpha	1 X 10^{-11} μCi/ml
	Noble Gas Monitor	Continuous (b)	Continuous	Gross Noble Gases (g) (Beta, Gamma)	1 X 10^{-6} μCi/ml
4. Turbine Building Vent (Gaseous)	Grab Sample	31 days (c)	31 days (c)	Principal Gamma Emitters (f)	1 X 10^{-4} μCi/ml (i)
	Grab Sample	92 days (a)	92 days	H-3	1 X 10^{-6} μCi/ml
	Charcoal Sample	Continuous (b)	7 days (d)	I-131	1 X 10^{-12} μCi/ml
				I-133	1 X 10^{-10} μCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1 X 10^{-11} μCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1 X 10^{-11} μCi/ml
				Sr-90	1 X 10^{-11} μCi/ml
				Gross Alpha	1 X 10^{-11} μCi/ml
	Noble Gas Monitor	Continuous (b)	Continuous	Gross Noble Gases (g) (Beta, Gamma)	1 X 10^{-6} μCi/ml
5. Multi Purpose Facility (MPF) Building Vent (Gaseous)	Charcoal Sample	Continuous (b)	7 days (d)	I-131	1 X 10^{-12} μCi/ml
				I-133	1 X 10^{-10} μCi/ml
	Particulate Sample	Continuous (b)	7 days (d)	Principal Gamma Emitters (f) (I-131, Others)	1 X 10^{-11} μCi/ml (i)
	Composite Particulate Sample (e)	Continuous (b)	92 days	Sr-89	1 X 10^{-11} μCi/ml
				Sr-90	1 X 10^{-11} μCi/ml
				Gross Alpha	1 X 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):

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

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., $1 \times 10^{-4}/I$, 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

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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. <u>AND</u> 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 ≤ 2% by volume.

APPLICABILITY: During augmented offgas treatment system operation.

ACTIONS

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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.

---------------------------NOTE---------------------------

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

---- --------------------------------NOTE------------------------------------

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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 <u>AND</u> 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

-----------------------------------NOTES-----------------------------------

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

3. 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.	Immediately
	<u>OR</u>		
	A.2	Declare channel inoperable.	Immediately

(continued)

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.
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.	Prior to initiating a release
	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.

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 <u>AND</u> 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 <u>AND</u> Once per 4 hours thereafter
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
G. Required Action and associated Completion Time for Condition D not met.	G.1 Enter the problem into the Corrective Action Program for investigation of the compliance failure.	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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

-----NOTES-----

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

3. 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.	Immediately
	<u>OR</u>		
	A.2	Declare channel inoperable.	Immediately

(continued)

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
	<u>AND</u>		
	B.2.1	Restore inoperable channel(s) to OPERABLE status.	31 days
		<u>OR</u>	
	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.

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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. AND C.2 Ensure the Elevated Release Point Monitoring noble gas activity monitor is OPERABLE. AND C.3 Restore inoperable channels to OPERABLE status.	Immediately Immediately 72 hours
D. Required Action and associated Completion Time for Condition C not met.	D.1 Be in MODE 2	12 hours
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

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action B.1 and referenced in Table D3.3.2-1.	F.1	Take grab samples.	24 hours AND Once per 24 hours thereafter
	AND		
	F.2	Analyze for gross activity.	24 hours from time of sampling completion
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	Immediately
		AND	
	G.1.2	Monitor recombiner exhaust temperature	Immediately
	OR		
	G.2.1.1	Collect gas sample	24 hours
		AND	AND Once per 24 hours thereafter
	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	Immediately
		OR	
	G.2.2.2	Monitor recombiner exhaust temperature	Immediately

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Required Action and associated Completion Time for Condition G not met.	H.1 Discontinue operation of the augmented offgas treatment system.	Immediately
I. As required by Required Action B.1 and referenced in Table D3.3.2-1.	I.1 Continuously collect samples with auxiliary sampling equipment as required in Table D3.2.3-1. 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 I.2.2 Report this event in the Radioactive Effluent Release Report.	4 Hours Immediately 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
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 Close the offgas isolation valve AND K.2 Initiate reactor shutdown AND K.3 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

(continued)

SURVEILLANCE		FREQUENCY
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
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)	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)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b)	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.

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)	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)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b)	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)	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)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b)	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)	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)	1	I	DSR 3.3.2.2

(continued)

(b) During releases via this pathway

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)	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)	1	I	DSR 3.3.2.2
b. Particulate Sampler Filter	(b)	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

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

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

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

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

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/m^3)	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

-----NOTES-----

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
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

-----NOTES-----

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

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.	92 days
	<u>AND</u>		
	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

NOTES

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

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.

TABLE D5.2-1
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 5.0 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/m^3)	FISH (pCi/Kg, Wet)	MILK (pCi/l)	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).

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B 3.0 ODAM Specification Applicability

BASES

As stated, the Technical Specification Section 3.0 requirements apply to the ODAM except for LCO 3.0.6 and LCO 3.0.7 (or otherwise excepted in ODAM Notes). LCO 3.0.6 and LCO 3.0.7 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.*) Since 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.

LCO 3.0.7 provides for explicit changes to specified Technical Specifications by the Section 3.10 Specifications. Specifically, 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 x 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

(continued)

BASES

(continued)

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:

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

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

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

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

$$C = 0.013\ \mu\text{Ci/ml}$$

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

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

$$C = 0.025\ \mu\text{Ci/ml}$$

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

(continued)

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

(continued)

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).

(continued)

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), <u>no recommendation</u> 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 ≤ 500 mrem/year to the total body or to ≤ 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, a simplified flow diagram showing gaseous effluent treatment and monitoring equipment. 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.

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

Custom Technical Specifications Bases did not exist.

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.

PROCESS CONTROL PROGRAM

FOR

COOPER NUCLEAR STATION

Cooper Nuclear Station

Process Control Program

List of Effective Pages

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A. Purpose:

The Cooper Nuclear Station (CNS) Process Control Program (PCP) establishes the processing conditions for assuring the SOLIDIFICATION, dewatering or stabilization of CNS radioactive waste streams produced from the CNS liquid radioactive waste treatment system and from activities producing radioactive waste requiring SOLIDIFICATION, dewatering or stabilization such as decontamination system resins, irradiated components and highly contaminated equipment.

The CNS PCP is comprised of the Dewatering Process Control Program (DPCP), SOLIDIFICATION Process Control Program (SPCP) and vendor Process Control Programs. The DPCP utilizes the CNS dewatering system to process solid wet waste streams from the CNS liquid radioactive waste treatment system or from chemical decontamination resins. The SPCP utilizes the cement solidification system to process solid wet waste from the liquid radioactive waste treatment system. Vendor Process Control Programs utilize NRC approved PCP's, stabilization processes and High Integrity Containers to process various forms of solid and liquid radioactive wastes at CNS.

B. Definitions:

1. SOLIDIFICATION, SOLIDIFY, SOLIDIFIED - The conversion of radioactive wastes from liquid systems to a solid which is as uniformly distributed as reasonably achievable with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

2. Stability - Structural stability which ensures that the waste does not degrade and (a) promote slumping, collapse, or other failure of the cap or cover over a near-surface disposal facility and thereby lead to water infiltration, or (b) impart a substantial increase in surface area of the waste form that could lead to an increase in leach rate. Stability is also a factor in limiting exposure to an inadvertent intruder since it provides greater assurance that the waste form will be recognizable and nondispersable during its hazardous lifetime. Structural stability of a waste form can be provided by the waste form itself (as with activated stainless steel components),

by processing the waste to a stable form (e.g., stabilization), or by placing the waste in a container or structure that provides stability (e.g., high integrity container or engineered structure).

C. Discussion:

The regulation, "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR Part 61, establishes a waste classification system based on the radionuclide concentrations in the waste. All Class B and C wastes are required to be stabilized. Class A wastes have lower concentrations and may be segregated without stabilization. All Class A liquid wastes, however, require solidification or absorption to meet the free liquid requirements. Class A wastes may also be stabilized and disposed of with stabilized Class B and C wastes.

D. References:

1. CNS Operating Procedure 2.5.4.1, Solid Wet Waste Packaging, Storage and Transfer System.

2. CNS Operating Procedure 2.5.4.2, Solid Wet Waste Drum Filling.

3. CNS Operating Procedure 2.5.4.3, Radwaste Drum Mixer Operation.

4. CNS Operating Procedure 2.5.4.4, NuPac Dewatering System.

5. CNS Operating Procedure 2.5.1.3, Radwaste Low Conductivity Liquid Waste Spent Resin Tank Fluid Transfer.

6. CNS Operating Procedure 2.5.2.1, Radwaste High Conductivity Liquid Waste and Waste Sludge Tank Fluid Transfer.

7. CNS Operating Procedure 2.5.1.2, Radwaste Low Conductivity Liquid Waste Condensate Phase Separator Fluid Transfer.

8. CNS Operating Procedure 2.5.1.7, RWCU Phase Separator Tank Transfer.

9. CNS Radiological Protection Procedure 9.RW.1, Radioactive Shipments.

10. CNS Radiological Protection Procedure 9.RW.2, Condensate Waste Resins, Spent Resins, RWCU Resins and Waste Sludge Classification and Listing.

11. CNS Radiological Protection Procedure 9.RW.7, Waste Stream Sampling.

12. Cooper Nuclear Station Offsite Dose Assessment Manual, D.3.5.1 - Solid Radioactive Waste.

13. Cooper Nuclear Station Technical Specifications,5.6.3 - Radioactive Effluent Release Report.

14. Cooper Nuclear Station Updated Safety Analysis Report, XIII-13.0 - Process Control Program.

15. 10CFR61 Licensing Requirements for Land Disposal of Radioactive Waste.

16. Waste Form Technical Position, Revision 1, January 1991, U. S. NRC.

17. Certificate of Compliance WN-EB-01, State of Washington Department of Health.

18. Certificate of Compliance WN-EB-02, State of Washington Department of Health.

19. Safety Evaluation Report TP-02-P, Revision 1, U. S. NRC.

20. Cooper Nuclear Station Offsite Dose Assessment Manual, D.5.3. - Radioactive Effluent Release Report.

E. Process Descriptions:

1. SOLIDIFICATION Process Control Program

a. Overview

The Cooper Nuclear Station (CNS) SOLIDIFICATION Process Control Program (SPCP) is a program of sampling, processing, analysis and formulation determination by which SOLIDIFICATION of radioactive waste from liquid systems is assured to be consistent with the CNS Offsite Dose Assessment Manual. Compliance with the SPCP ensures that the resultant waste form characteristics are acceptable for burial at a licensed low level radioactive waste burial facility.

The SPCP is not intended to be a substitute for the CNS Operations Manual. The SPCP outlines the general methods of sampling, processing, analysis and waste formulation during the SOLIDIFICATION of solid wet radioactive waste. The CNS Operations Manual details the actual methods by which solid wet waste is SOLIDIFIED.

Solid wet waste streams produced from the liquid radioactive waste treatment system can be SOLIDIFIED utilizing the CNS SPCP. These wet waste streams are comprised of condensate resins, spent resins, reactor water cleanup resins and waste sludges. When regulations require SOLIDIFICATION, these waste streams are SOLIDIFIED in accordance with SPCP requirements. The waste processing is performed with equipment installed at CNS under the direct supervision of CNS personnel utilizing CNS approved procedures. ALARA considerations and radiological controls are incorporated into these procedures. After the waste has been processed in its burial container, the waste classification and type is determined so that the waste may be shipped to a licensed burial site in accordance with appropriate NRC, DOT, state and burial site regulations.

b. Operation - General:

The solid wet radioactive waste SOLIDIFICATION process is operated on a batch basis. A batch consists of all the resulting drums sequentially processed from the contents of a single source such as a phase separator. The batch is completed when the waste source is empty or the batch process is terminated due to radioactive waste system operational constraints. Solid wet waste sources which are typically processed on a batch basis for SOLIDIFICATION include the condensate phase separators, reactor water clean-up phase separators, waste sludge tank or spent resin tank.

Waste steams being SOLIDIFIED from the condensate or reactor water cleanup phase separators, waste sludge tank, or spent resin tank are routed through centrifuge units for the purpose of dewatering the waste slurry. The wastes enter a storage hopper after dewatering in the centrifuge units. The waste in the storage hopper ranges from a relatively dry granular consistency to a wet, putty-like consistency, depending on the waste material (e.g., sludge, resin). Department of Transportation (DOT) 17H specification 55 gallon drums containing dry Portland Type I cement are transferred under the hopper and filled with the dewatered waste and demineralized water. The drum then progresses to the drum mixing section where an in-drum mixes the cement and waste.

During the mixing process additional water is added to the mixture in quantities sufficient to ensure SOLIDIFICATION. Due to the varying degrees of moisture content subsequent to the centrifuge dewatering process, a visual inspection of the initial drums being mixed is necessary to establish the exact amount of water required to SOLIDIFY the waste batch. Once the correct volume of water for the batch is determined the mixing system programmer is adjusted to add the correct amount of mixing water automatically. The water, cement and waste are thoroughly mixed by the mixing system's programmed sequence. After the mixing sequence is

complete the radiation levels are measured on each drum. The drum is then transferred to the drum storage line.

The filled drums are allowed to SOLIDIFY a minimum of twenty-four hours before being transferred from the drum storage line to the drum capping station. Here the drums are visually inspected for freestanding liquid and SOLIDIFICATION. Drums having no visible freestanding liquid and meeting SOLIDIFICATION acceptance criteria are capped, decontaminated and returned to the drum storage line in preparation for shipment to a licensed burial facility. Drums not complying with freestanding liquid of SOLIDIFICATION requirements are retained for corrective disposition prior to capping and storage.

c. <u>Operation - Specific</u>:

1.8 ± 0.3 cubic feet of dry Portland Type I cement is added to each DOT 17H specification 55 gallon drum. It has been demonstrated that this volume of cement will properly SOLIDIFY the waste streams processed by the drum mixing system. The waste to be SOLIDIFIED is added in quantities sufficient to fill the drum along with approximately seven gallons of demineralized water. When the drum filling is complete the drum is transferred to the drum mixing station. A representative waste sample from one drum of each batch is taken and analyzed for pH. Experience has conclusively shown that the SOLIDIFICATION process is unaffected by pH as long as the pH of the waste remains within the range of 2 to 13.

The filled drum is mixed by an automatic, sequence controlled, in-drum mixer. During the mixing sequence approximately five gallons of additional demineralized water are added to the filled drum. When the water addition is complete and the final mixing steps are sequencing to completion, the mixer motor amps are checked to ensure sufficient water has been added. Slight changes in the amount of water being added during the mixing sequence may be made to adjust the mixer motor amps to approximately

seven amps. These adjustments are made on the first several drums being processed from the waste batch, and are then programmed into the drum mixing sequence for the remaining drums of the batch. Upon completion of the mixing process, and prior to the cement setting, one drum from the batch is sampled in order to determine its isotopic content and distribution. The drum's radiation level is also measured. By comparing this drum's isotopic distribution, concentration, and radiation level with the radiation readings on each of the other drums in the batch the total concentration of the radionuclides present in each drum can be determined. This data comparison method may also be used to determine waste carry-over from previous batches into the batch currently being processed.

Following a minimum of twenty-four hours after mixing, each SOLIDIFIED drum is visually inspected for freestanding liquid and SOLIDIFICATION. Every tenth drum of the batch is quality control checked for resistance to penetration to verify SOLIDIFICATION. The penetration test verifies that the SOLIDIFIED waste has a minimum compressive strength of 50 psi. Filled drums meeting the freestanding liquid and SOLIDIFICATION criteria are capped and transferred to the drum storage line for later shipment to a licensed burial site. The packaging, classification and shipping of wastes processed via the CNS SPCP are in accordance with the applicable sections of 10CFR61, 10CFR71 and 49CFR Parts 171 through 178.

d. <u>Qualification</u>:

Qualification of the SPCP process for stabilization must be conducted for each waste stream requiring stabilization (e.g., Class B waste, Class C waste) prior to burial. Qualification shall be conducted in accordance with the stabilization testing recommendations of the "Waste Form Technical Position" Revision 1, January 1991, U. S. NRC. The SPCP stabilization qualification testing is typically performed by a CNS approved vendor.

e. Scaling Factors:

As permitted by 10CFR61.55, waste stream radionuclide scaling factors based on certain gamma emitting nuclides (e.g., Co-60, Cs-137) may be used to calculate the concentrations of difficult-to-measure radionuclides. Scaling factor determinations are made using a detailed radionuclide analysis of waste stream samples conducted by a CNS approved vendor.

f. Reports:

The volume, curie content, principle nuclides, type of waste and solidification agent for the wet wastes SOLIDIFIED at Cooper Nuclear Station are documented in the Cooper Nuclear Station Radioactive Effluent Release Report. This information is listed in the format outlined in Revision 1 of Regulatory Guide 1.21.

F. Dewatering:

a. General

The CNS Dewatering PCP (DPCP) utilizes dewatering equipment and disposable waste containers to dewater solid wet waste streams produced from the liquid radioactive waste treatment system or from chemical decontamination resins. Compliance with the DPCP ensures that the resultant waste meets the free standing liquid criteria set forth in 10CFR61. The DPCP is not intended to be a substitute for the CNS Operations Manual nor does the process meet the waste stability form requirements of 10CFR61. Stabilization/solidification of the dewatered waste is provided by High Integrity Containers (HICs) and/or Engineered Concrete Barriers. Solidification of the dewatered waste can also be accomplished by dewatering the waste in metal liners.

Solid wet waste streams produced from the liquid radioactive waste treatment system and chemical decontamination resins can be dewatered utilizing the CNS

DPCP. The liquid radioactive waste treatment streams include condensate resins, spent resins, reactor water cleanup resins and waste sludges. When regulations require solidification or stabilization, these waste streams can be dewatered in HICs and burial liners per DPCP requirements. The waste processing is performed with the Nuclear Packaging, Inc. dewatering system equipment installed at CNS, under the direct supervision of CNS personnel utilizing CNS approved procedures. The design and arrangement of the CNS dewatering system components are based on maintaining the operator radiation exposure ALARA. ALARA considerations and radiological controls are also incorporated into the system operating procedures. After the waste has been processed in its burial liner the waste classification and type are determined so that the waste may be shipped to a licensed burial site in accordance with appropriate NRC, DOT, state and burial site regulations.

b. Operation - General:

The solid wet radioactive waste dewatering process is operated on a batch basis. A batch consists of all the resulting dewatered liners processed from the contents of a waste source such as a phase separator. The batch is complete when the waste source is empty or the batch process is terminated due to radioactive waste system operational constraints. Solid wet waste sources which are typically dewatered on a batch basis include the spent resin tank, waste sludge tank, condensate phase separators, reactor water clean-up phase separators, decontamination resins, or a combination of these waste sources. Prior to dewatering the solid wet waste, verification is made that the waste being dewatered is characterized in accordance with the NUPAC Services Division, Inc. Waste Characterization Form. Selection of the burial liner (steel liner or HIC) utilized in the dewatering process is based upon the classification and type of waste being processed.

The CNS dewatering system consists of a dewatering waste container, a dewatering pump, an off-gas vent unit, a container level indicator, a waste fill head, a water separator with water chiller unit, an air blower, a relative humidity instrument, a control panel, and interconnecting piping and valves.

After wet radioactive waste is charged into a waste container, dewatering is achieved with continuous suction on the waste container by the dewatering pump. Various types, numbers, and configurations of filters are used within the waste container to retain spent resin and filter precoat materials. Water removed from the waste container is returned to the liquid radwaste system.

The dewatering pump is operated for given time internals in accordance with the NuPac Services Division, Inc. Waste Characterization Form. The pumping time may range from eight to sixteen hours depending upon the waste stream and waste container. After most of the free water in the waste container has been removed, drying air is continuously recirculated in a loop from the air blower to the waste container through the water separator to remove any residual free water in the waste container. The dewatering system is provided with temperature instrumentation which is interlocked to automatically shut down the dewatering process on high air temperature due to the potential for exothermic reactions during ion exchange resin dewatering. The waste container is considered dewatered when at least 8 hours of pumping time has been completed and, the relative humidity in the recirculating drying air meets the acceptance criteria specified in the NUPAC Services Division, Inc. Waste Characterization Form. A relative humidity instrument and monitor are provided to remotely and continuously monitor the waste container outlet air. This instrument is used to establish the positive end point to the dewatering process and the point at which the free standing liquid criteria of 10CFR61 is met. Filled waste containers which do not meet the dewatering end point criteria are retained until a satisfactory corrective disposition is determined. Stability or solidification of the waste is provided by the waste container and/or engineered barriers such as concrete vaults.

No airborne or liquid radwastes are released to the environment from the dewatering operation. The dewatered liquid radwastes are routed to the liquid radioactive waste treatment system and resin drying air is vented through a HEPA system. The dewatering system is designed to prevent the uncontrolled release of radioactive materials by monitoring liquid levels in the waste container by a level indicator. During the waste filling operation, the operator visually monitors the

waste transfer process by observing the dewatering control panel video monitor and the local area radiation monitors. Potential inadvertent spills and overflows are contained in the Augmented Radioactive Waste Building and can be routed back to the liquid radioactive waste treatment system via floor drains.

c. <u>Operation - Specific</u>:

The CNS Dewatering System fillhead is installed on a waste container compatible with the dewatering fillhead. The waste container typically has four filter levels. The filters provide the flow path from the water container, through the fillhead, to the dewatering pump. The fillhead is fitted with connections to transfer solid wet waste from the Waste Transfer System to the waste container. The fillhead also contains a remote video camera for internal monitoring of the waste container, dewatering pump suction connections, electrical service connections for the waste container level indication system, a service air supply to provide cooling, a spray ring header for flushing, a pressure switch to monitor container pressure, and connections for the waste container temperature indicating system.

Once the fillhead is installed and the bottom filter level dewatering pump suction valve is opened, solid wet waste from the Condensate Phase Separator, Spent Resin Tank, or Reactor Water Cleanup Phase Separator is transferred as a homogeneously mixed slurry to the waste container. During the waste transfer the fillhead video monitor is used to verify that the waste container is being filled.

When the waste slurry reaches the second filter level, the dewatering pump is started. The dewatering pump is cycled on and off to maintain the water level in the waste container slightly above the level of the waste. When the slurry level reaches the third level of filters the second level dewatering pump suction valve is opened and the first level dewatering suction valve is closed. At this point a sample of the waste slurry is drawn for gamma isotopic analysis. As the waste slurry level reaches the fourth filter level the third level dewatering pump suction valve is opened. When the slurry level reaches the fill line deflector plate the fourth level dewatering pump suction valve is opened. Waste slurry transfer is continued until

the waste container is full. Demineralized water is then used to flush the residue in the waste transfer lines and fill head connections to the waste container. The dewatering pump is stopped, the second and third level dewatering suction valves closed, and the Waste Transfer System lineup secured. The dewatering cycle is then commenced by starting the dewatering pump. When the fourth level dewatering pump suction line vacuum fluctuates rhythmically with the dewatering pump and/or drops to approximately 5 inches Hg the fourth level dewatering suction valve is closed and the third level suction valve is opened. As the third level dewatering pump suction line vacuum begins to fluctuate rhythmically with the dewatering pump and/or drops to approximately 5 inches Hg the third level dewatering suction valve is closed and the second level dewatering suction valve is opened. Similarly, as the second level dewatering pump suction line vacuum begins to fluctuate rhythmically with the dewatering pump and/or drops to approximately 5 inches Hg the second level dewatering suction valve is closed and the first level dewatering suction valve is opened. When the first level dewatering pump suction line vacuum fluctuates rhythmically with the dewatering pump and/or drops to approximately 5 inches Hg the first level dewatering suction valve is closed and the dewatering pump secured.

After the temperature of the radioactive waste in the waste container is measured and recorded the drying cycle is commenced. The drying cycle utilizes a blower to blow warm dry air through the waste to evaporate the residual water remaining in the waste container. The warm moist air which exits from the waste container is pumped through an entrainment separator tank where refrigeration coils condense the water vapor and the entrained water is removed. The water collected in the entrainment separator tank is pumped to the radioactive waste floor drain system. The dehumidified warm air is recirculated through the waste until the endpoint humidity and minimum recirculation dry time requirements of the NuPac Waste Characterization Form are met.

The drying cycle begins by closing the fillhead waste isolation valve, aligning the blower to the waste container and starting the dewatering system chiller. The dewatering pump and blower are started and the blower inlet pressure, blower

discharge pressure, dewatering pump suction pressure, blower inlet temperature, blower discharge temperature, waste container temperature parameters are then monitored and recorded each hour. After 3 hours of blower operation the waste level in the waste container is checked. Due to volume reduction inherent with the drying cycle, additional waste may have to be added to the waste container. To add waste, the drying cycle is secured and waste is added per the filling and dewatering cycles. Once additional waste is added the drying cycle is resumed. Several iterations of blower operation and waste addition may be required.

When 3 hours of blower operation have been completed, and waste addition is not required, the Relative Humidity Monitoring System is placed in operation. This system monitors the relative humidity of the air stream from the waste container before the air enters the entrainment separator tank. The sample tubing is wrapped with heat tape and insulated to prevent condensation in the sample lines. The air sample flows through a filter/separator and then into the sample chamber. Air flow is maintained and monitored with a flow meter. A General Eastern 100DP Dew Point Hygrometer measures the system air temperature and the system air dew point temperature. From this data, the percent (%) relative humidity of the entrainment separator inlet air is determined.

The air temperature and dew point temperature of the air stream at the inlet of the entrainment separator tank is then monitored and recorded hourly in addition to the temperature and pressure parameters previously noted. The percent relative humidity of the air stream is calculated using the air temperature and dew point temperatures. The drying cycle is considered complete when the percent relative humidity of the air stream reaches the end point relative humidity determined by the NuPac Services Division, Inc. Waste Characterization Form and the NuPac Services Division, Inc. Humidity Endpoint Graph and, when at least eight hours of blower operation has been completed. Once the drying cycle is successfully completed the CNS Dewatering System is shutdown, the fillhead removed from the waste container and the container primary and secondary closure devices installed. If the percent relative humidity endpoint can not be attained the waste container is retained for corrective disposition. Waste containers which have been successfully

dewatered are classified and shipped to a licensed burial site in accordance with the applicable sections of 10CFR61, 10CFR71, and 49CFR Parts 171 through 178.

d. Qualification:

Qualification of the DPCP process for dewatering was conducted in accordance with reference 19. Qualification of High Integrity Containers used for stabilization was conducted in accordance with references 17 and 18.

e. Scaling Factors:

As permitted by 10CFR61.55, waste stream radionuclide scaling factors based on certain gamma emitting nuclides (e.g., Co-60, Cs-137) may be used to calculate the concentrations of difficult-to-measure radionuclides. Scaling factor determinations are made using a detailed radionuclide analysis of waste stream samples, by a CNS approved vendor.

f. Reports:

The volume, curie content, principle nuclides, and type of waste for the wet wastes dewatered at Cooper Nuclear Station are documented in the Cooper Nuclear Station Radioactive Effluent Release Report. This information is listed in the format outlined in Revision 1 of Regulatory Guide 1.21.

G. Vendor Process Control Programs, Stabilization Processes, and High Integrity Contain Qualifications:

Vendor supplied Process Control Programs, Stabilization Processes and High Integrity Containers which have NRC approved Topical Reports or, which are approved in Low-Level Radioactive Waste Disposal Facility licenses, can be utilized to process radioactive waste at CNS.

H. District Initiated Changes

1. Shall be submitted to the Commission by inclusion in the Radioactive Effluent Release Report for the period in which the change(s) was made effective and shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.