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0CAN042202 10 CFR 50.36a

April 14, 2022

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Radioactive Effluent Release Report for 2021

Arkansas Nuclear One, Units 1 and 2
NRC Docket Nos. 50-313 and 50-368
Renewed Facility Operating License Nos. DPR-51 and NPF-6

Reference: Entergy Operations, Inc. letter to U. S. Nuclear Regulatory Commission, "Radioactive Effluent Release Report for 2020," (ADAMS Accession No. ML21111A252), dated April 21, 2021

Arkansas Nuclear One, Units 1 and 2 (ANO-1 and ANO-2) Technical Specifications (TSs) 5.6.3 and 6.6.3, respectively, require the submittal of a Radioactive Effluent Release Report annually. The information which fulfills this reporting requirement for ANO-1 and ANO-2 for the 2021 calendar year is enclosed.

ANO-1 TS 5.6.3 and ANO-2 TS 6.6.3 require this report to be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. 10 CFR 50.36a(a)(2) requires that the time between submission of the reports must be no longer than 12 months. The reference document is the ANO Radioactive Effluent Release Report for 2020.

Liquid and gaseous release data show that the dose from both ANO-1 and ANO-2 was considerably below the Offsite Dose Calculation Manual limits. The data reveals that radioactive effluents had an overall minimal dose contribution to the surrounding environment.

This letter contains no new regulatory commitments.

Should you have any questions or require additional information, please contact Riley Keele, Manager, Regulatory Assurance, at (479) 858-7826.

Respectfully,

Riley D Keele Jr.

Riley Keele

RDK/rwc

Enclosure: Radioactive Effluent Release Report

Attachments to Enclosure:

1. Offsite Dose Calculation Manual
2. Process Control Program

cc: NRC Region IV Regional Administrator

NRC Senior Resident Inspector – Arkansas Nuclear One

NRC Project Manager – Arkansas Nuclear One

Designated Arkansas State Official

ENCLOSURE

0CAN042202

RADIOACTIVE EFFLUENT RELEASE REPORT

Plant: Arkansas Nuclear One	Page 1 of 32
	YEAR: 2021
Document Number: 0CAN042202	
Annual Radioactive Effluent Release Report	

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1.0 INTRODUCTION

Arkansas Nuclear One (ANO) is a two unit site consisting of a Babcock & Wilcox (B&W) (Unit 1) and a Combustion Engineering (CE) (Unit 2) nuclear steam supply system. Both liquid and gaseous effluents are released in accordance with the ANO Offsite Dose Calculation Manual (ODCM). This report is a summary of the effluent data in accordance with Unit 1 Technical Specification (TS) 5.6.3 under License Number DPR-51 and Unit 2 TS 6.6.3 under License Number NPF-6. ANO-1 capacity factor for 2021 was 0.907 and ANO-2 was 0.824.

2.0 SUPPLEMENTAL INFORMATION

2.1 Regulatory Limits

The ODCM contains the limits to which ANO must adhere. Because of the "as low as reasonably achievable" (ALARA) philosophy at ANO, actions are taken to reduce the amount of radiation released to the environment. Liquid and gaseous release data show that the dose from ANO is considerably lower than the ODCM limits. This data reveals that the radioactive effluents have an overall minimal dose contribution to the surrounding environment. The following are the limits required by the ODCM:

1. Fission and activation gases:

   a. Noble gases dose rate due to radioactive materials released in gaseous effluents from the areas at and beyond the site boundary shall be limited to the following:

      - Less than or equal to 500 mrem/year to the total body
      - Less than or equal to 3000 mrem/year to the skin

   b. Noble gas air dose due to noble gases released in gaseous effluents to areas at and beyond the site boundary shall be limited to the following:

      1) Quarterly

         - Less than or equal to 5 mrads gamma
         - Less than or equal to 10 mrads beta

      2) Yearly

         - Less than or equal to 10 mrads gamma
         - Less than or equal to 20 mrads beta

2. Iodine, tritium, and all radionuclides in particulate form with half-lives greater than 8 days.

   a. The dose rate for Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the site boundary shall be limited to the following:

      - Less than or equal to 1500 mrem/year to any organ

   b. The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the site boundary shall be limited to the following:

      1) Quarterly

         - Less than or equal to 7.5 mrem to any organ

      2) Yearly

         - Less than or equal to 15 mrem to any organ

3. Liquid Effluents Dose

   a. The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to unrestricted areas shall be limited to the following:

      1) Quarterly

         - Less than or equal to 1.5 mrem total body
         - Less than or equal to 5 mrem critical organ

      2) Yearly

         - Less than or equal to 3 mrem total body
         - Less than or equal to 10 mrem critical organ

4. Total Dose (40 CFR 190)

   a. The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to the following:

      - Less than or equal to 25 mrem, Total Body or any Organ except Thyroid.
      - Less than or equal to 75 mrem, Thyroid

2.2 Maximum Permissible Concentrations

1. Fission and Activation Gases, Iodines, and Particulates with Half Lives > Eight (8) Days

   For gaseous effluents, maximum permissible concentrations are not directly used in release rate calculations since the applicable limits are expressed in terms of dose rate at the site boundary.

2. Liquid Effluents

   The concentration of radioactive material released shall be limited to the concentration specified in 10 CFR 20, Appendix B, Table 2, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the total concentration released shall be limited to 2.0E-4 microcuries/ml.

2.3 Measurements and Approximations of Total Radioactivity

1. Gaseous Effluents

   a. Fission and activation gases

      Gas samples are collected weekly and are counted on a high purity germanium detector (HPGe) for principal gamma emitters. The samples are collected from the containment vent, auxiliary building, piping penetration room, and spent fuel pool areas for ANO-1 and ANO-2. Additionally, the Auxiliary Building Extension and low level radwaste storage are sampled. All effluent waste streams are only sampled when the ventilation is active. TS release points are continuously monitored and the average release flow rates for each release point are used to calculate the total activity released during a given time period.

   b. Iodines

      Iodine is continuously collected on F&J TE2C cartridge filter via an isokinetic sampling assembly from each release point. Filters are exchanged once per week and then analyzed on an HPGe system. The flow rates for each release point are averaged over the duration of the sampling period and these results - along with specific isotopic concentrations - are then used to determine the total activity released during the time period in question.

   c. Particulates (half-lives > 8 days)

      Particulates are continuously collected on a filter paper via an isokinetic sampling assembly on each release point. Filters are exchanged once per week and then analyzed on an HPGe system. The flow rates for each release point are averaged over the duration of the sampling period and these results - along with specific isotopic concentrations - are then used to determine the total activity released during the time period in question.

Section 2.3 (Continued)

d. Tritium

Tritium is collected by passing a known volume of the sample stream through a calcium chloride desiccant on a weekly basis from each release point. The collected samples are distilled and analyzed by liquid scintillation. The tritium released was calculated for each release point from the measured tritium concentration, the volume of the sample, the tritium collection efficiency, and the respective stack exhaust flow rates.

e. Carbon-14

1) ANO-1

Carbon-14 release values were estimated using the methodology included in the EPRI Technical Report 1021106, using the 2021 normalized Carbon-14 production rate of 3.4 Ci/GWtyr, a gaseous release fraction of 98 percent, a Carbon-14 carbon dioxide fraction of 30 percent, a reactor power rating 2568 MWt, and equivalent full power operation of approximately 331 days.

2) ANO-2

Carbon-14 release values were estimated using the methodology included in the EPRI Technical Report 1021106, using the 2021 normalized Carbon-14 production rate of 3.9 Ci/GWtyr, a gaseous release fraction of 98 percent, a Carbon-14 carbon dioxide fraction of 30 percent, a reactor power rating of 3026 MWt, and equivalent full power operation of approximately 301 days.

2. Liquid Effluents

a. Batch Releases

Each tank of liquid radwaste is sampled and analyzed for principal gamma emitters prior to release. Each sample tank is recirculated for a sufficient amount of time prior to sampling ensuring that a representative sample is obtained. Samples are then analyzed on an HPGe system and liquid release permits are generated based upon the values obtained from the isotopic analysis and the most recent values for H3, gross alpha, Fe-55, Sr-89, and Sr-90. An aliquot based on release volume is saved and added to composite containers. The concentrations of composited isotopes and the volumes of the releases associated with these composites establish the proportional relationships that are then utilized for calculating the total activity released for these isotopes.

Section 2.3 Step 2 (Continued)

b. Continuous Releases

Samples are taken on ANO's two continuous liquid release streams (turbine building sumps) once per 24 hours. Samples are then analyzed on an HPGe system and liquid release permits are generated based upon the values obtained from the isotopic analysis and the most recent values for H3, gross alpha, Fe-55, Sr-89, and Sr-90. An aliquot based on release volume is saved and added to composite containers. The concentrations of composited isotopes and the volumes of the releases associated with these composites establish the proportional relationships that are then utilized for calculating the total activity released for these isotopes.

3. Estimated Total Error Present

a. Estimates of measurement and analytical error for gaseous and liquid effluents are calculated as follows:

$$E_T = \sqrt{[(E_1)^2 + (E_2)^2 + \ldots (E_n)^2]}$$

Where:

E_T = total percent error

E_1 ... E_n = percent error due to calibration standards, Laboratory analysis, instruments, sample flow, etc.

2.4 Batch Releases

2.4.1 Liquid

2.4.2 Unit-1

1. Number of batch releases: 105
2. Total time period for a batch release: 2.61E+04 minutes
3. Maximum time period for a batch release: 1.69E+03 minutes
4. Average time period for a batch release: 2.49E+02 minutes
5. Minimum time period for a batch release: 2.00E+00 minutes

2.4.3 Unit-2

1. Number of batch releases: 42
2. Total time period for a batch release: 1.52E+04 minutes
3. Maximum time period for a batch release: 7.64E+02 minutes
4. Average time period for a batch release: 3.62E+02 minutes
5. Minimum time period for a batch release: 5.50E+01 minutes

2.4.4 Gaseous

2.4.5 Unit 1

1. Number of batch releases: 103
2. Total time period for a batch release: 8.91E+05 minutes
3. Maximum time period for a batch release: 1.05E+04 minutes
4. Average time period for a batch release: 8.65E+03 minutes
5. Minimum time period for a batch release: 6.70E+01 minutes

2.4.6 Unit 2

1. Number of batch releases: 127
2. Total time period for a batch release: 1.11E+06 minutes
3. Maximum time period for a batch release: 1.05E+04 minutes
4. Average time period for a batch release: 8.73E+03 minutes
5. Minimum time period for a batch release: 6.00E+01 minutes

2.5 Continuous Releases

2.5.1 Liquid

2.5.2 Unit 1

1. There were zero continuous releases.

2.5.3 Unit 2

1. Number of continuous releases: 1
2. Total time period for a continuous release: 1.04E+04 minutes
3. Maximum time period for a continuous release: 1.04E+04 minutes
4. Average time period for a continuous release: 1.04E+04 minutes
5. Minimum time period for a continuous release: 1.04E+04 minutes

2.5.4 Gaseous

2.5.5 Unit 1:

There were zero continuous releases.

2.5.6 Unit 2:

There were zero continuous releases.

2.6 Abnormal Releases

2.6.1 ANO-1

1. Liquid
   a. Number of releases: 0
2. Gaseous
   a. Number of releases: 0

2.6.2 ANO-2

1. Liquid
   a. Number of releases: 0
2. Gaseous
   a. Number of releases: 0

2.7 Non-routine, Planned Discharges

2.7.1 ANO-1 Non-routine Discharges

1. There were zero non-routine discharges for Unit 1.

2.7.2 ANO-2 Non-routine Discharges

1. There were two discharge points that are considered non-routine because it is not a significant release point for the site (<1 percent total dose) for ANO-2. The discharge point is the steam release off the emergency feedwater pump 2P-7A and steam leak-by on Main Steam Header 1 Steam Dump and Bypass Valve (2CV-0301).
   - Later in the operating cycle for ANO-2, tritium concentrations are greater than the minimum detectable activity (MDA) in the secondary water. This is due to the U-tube designed steam generators where the tritium from the reactor coolant migrates across the tubes. When 2P-7A is run for surveillances there is a steam release with tritiated water to the atmosphere. A release permit is generated to capture the dose released.
   - There is approximately 326 CFM leak by valve 2CV-0301 from the steam generating system. Due to tritium concentration in the secondary system a release permit is generated each month to account for this release. Repairs of 2CV-0301 were completed in refueling outage (2R29).

2.8 Radioactive Waste Treatment System Changes

1. There were zero changes to the radioactive waste treatment systems for either liquid or gases.

2.9 Land Use Census Changes

1. There were zero changes to receptor locations or routes of exposure as a result of the 2021 land use census. ANO performs land use census once every two years.

2.10 Effluent Monitor Instrument Inoperability

1. There were zero ODCM required radmonitors out-of-service greater than 30 days.

2.11 Offsite Dose Calculation Manual Changes

1. There were zero changes to ANO's ODCM in 2021.

2. Attachment 1, Revised Offsite Dose Calculation Manual is the most recent copy of the ODCM.

2.12 Process Control Program (PCP) Changes

1. There were zero changes to Entergy's Process Control Program procedure EN-RW-105 in 2021.

2. Attachment 2, Process Control Program is the most recent copy of EN-RW-105.

2.13 NON-REMP Groundwater Monitoring Results (NEI 07-07)

1. ANO has a total of 16 Non-REMP wells as part of the NEI 07-07 Ground Water Protection Program. There are 11 wells that are sampled on an annual basis and 5 that are performed quarterly. The 5 wells that are performed quarterly have a higher potential to become contaminated due to their proximity to the plant.

2. There were a total of 36 (includes duplicates) samples taken in 2021 for predominant gamma emitters (Mn-54, Co-58, Fe-59, Co-60, Zn-65, Nb-95, Zr-95, I-131, Cs-134, Cs-137, Ba-140 and La-140) with zero samples indicating results greater than the minimum detectable concentration.

3. There were a total of 36 (includes duplicates) samples taken in 2021 for tritium (H-3) with 3 samples indicating a positive result from Monitoring Well 17. Due to the well location in proximity to the Unit 1 containment building, it is most likely tritium recapture from rainwater. The sample results are below and do not exceed any reporting criteria or any federal limit.

Section 2.13 (Continued)

Table 1, Positive Well Results

Well	Sample Date	Analysis	Result	Error	MDC	Units
MW-17	9/8/2021 10:27	H-3	383	254	364	pCi/L
MW-17 (Duplicate)	9/8/2021 10:35	H-3	452	262	369	pCi/L
MW-17	12/8/2021 8:42	H-3	330	226	323	pCi/L

[MDC – Minimum Detectable Concentration]

4. There were zero spills that required entry into ANO's 10 CFR 50.75 (g) document for future decommissioning.

2.14 LLD Levels

In accordance with ODCM Appendix 1, lower limits of detection (LLDs) higher than required shall be documented in the Annual Radioactive Effluent Release Report (ARERR):

1. There were zero instances in which ANO did not meet their required sample LLDs.

2.15 Errata/Corrections to Previous ARERRs

1. There are zero errata to be submitted with the 2021 ARERR.

3.0 GASEOUS EFFLUENTS

3.1 Gas Effluent and Waste Disposal Report

3.1.1 ANO-1 Data

Table 2, Gaseous Effluents-Summation of All Releases (ANO-1)

		Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Est. Total Error %
A.	**Fission and Activation Gases**						
1.	Total Release	Ci	0.00E+00	2.09E-02	0.00E+00	0.00E+00	24
2.	Average release rate for the period	μCi/sec	0.00E+00	2.65E-03	0.00E+00	0.00E+00	
B.	**Iodine**						
1.	Total Iodine – 131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	20
2.	Average release rate for the period	μCi/sec	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
C.	**Particulates**						
1.	Particulates with half-lives > 8 days	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	22
2.	Average release rate for the period	μCi/sec	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
D.	**Tritium**						
1.	Total Release	Ci	2.58E+00	4.74E+00	4.31E+00	3.91E+00	21
2.	Average release rate for the period	μCi/sec	3.27E-01	6.01E-01	5.47E-01	4.95E-01	
E.	**Gross Alpha**						
1.	Total Release	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	31
2.	Average release rate for the period	μCi/sec	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
F.	**Carbon-14**						
1.	Total Release	Ci	2.04E+00	1.45E+00	2.16E+00	2.11E+00	
2.	Average release rate for the period	μCi/sec	2.68E-07	1.89E-07	2.77E-07	2.71E-07	

% of limit is on the Radiological Impact to Man Table (Section 6.0)

Table 3, Gaseous Effluents – Ground Level Release - Batch Mode (ANO-1)

Nuclides Released	Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
Fission Gases						
Ar-41	Ci	0.00E+00	1.90E-02	0.00E+00	0.00E+00	1.90E-02
Kr-85	Ci	0.00E+00	1.64E-03	0.00E+00	0.00E+00	1.64E-03
Xe-133	Ci	0.00E+00	2.45E-04	0.00E+00	0.00E+00	2.45E-04
Total for Period	**Ci**	**0.00E+00**	**2.09E-02**	**0.00E+00**	**0.00E+00**	**2.09E-02**
Iodines						
I-131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total for Period	**Ci**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
Particulates						
None	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total for Period	**Ci**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
Tritium						
H-3	Ci	2.58E+00	4.74E+00	4.31E+00	3.91E+00	1.55E+01
Gross Alpha						
Alpha	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Carbon-14						
C-14	Ci	2.04E+00	1.45E+00	2.16E+00	2.11E+00	7.76E+00

3.1.2 ANO-2 Data

Table 4, Gaseous Effluents-Summation of All Releases (ANO-2)

A.	Fission & Activation Gases	Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Est. Total Error %
1.	Total Release	Ci	0.00E+00	0.00E+00	1.59E+00	3.48E+00	24
2.	Average release rate for the period	μCi/sec	0.00E+00	0.00E+00	2.02E-01	4.42E-01	
B.	**Iodine**						
1.	Total Iodine – 131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	20
2.	Average release rate for the period	μCi/sec	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
C.	**Particulates**						
1.	Particulates with half-lives > 8 days	Ci	0.00E+00	0.00E+00	4.91E-07	0.00E+00	22
2.	Average release rate for the period	μCi/sec	0.00E+00	0.00E+00	6.22E-08	0.00E+00	
D.	**Tritium**						
1.	Total Release	Ci	4.30E+00	6.62E+00	8.22E+00	1.15E+01	21
2.	Average release rate for the period	μCi/sec	5.45E-01	8.40E-01	1.04E+00	1.45E+00	
E.	**Gross Alpha**						
1.	Total Release	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	31
2.	Average release rate for the period	μCi/sec	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
F.	**Carbon-14**						
1.	Total Release	Ci	2.85E+00	2.88E+00	2.72E+00	1.08E+00	
2.	Average release rate for the period	μCi/sec	3.74E-07	3.74E-07	3.50E-07	1.38E-07	

% of limit is on the Radiological Impact to Man Table (Section 6.0)

Table 5, Gaseous Effluents – Ground Level Release - Batch Mode (ANO-2)

Nuclides Released	Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
Fission Gases						
Ar-41	Ci	0.00E+00	0.00E+00	3.07E-02	3.48E+00	3.52E+00
Xe-133	Ci	0.00E+00	0.00E+00	1.79E-04	0.00E+00	1.79E-04
Xe-135	Ci	0.00E+00	0.00E+00	1.56E+00	0.00E+00	1.56E+00
Total for Period	**Ci**	**0.00E+00**	**0.00E+00**	**1.59E+00**	**3.48E+00**	**5.08E+00**
Iodines						
I-131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total for Period	**Ci**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
Particulates						
Co-58	Ci	0.00E+00	0.00E+00	4.91E-07	0.00E+00	4.91E-07
Total for Period	**Ci**	**0.00E+00**	**0.00E+00**	**4.91E-07**	**0.00E+00**	**4.91E-07**
Tritium						
H-3	Ci	4.30E+00	6.62E+00	8.22E+00	1.15E+01	3.06E+01
Gross Alpha						
Alpha	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Carbon-14						
C-14	Ci	2.85E+00	2.88E+00	2.72E+00	1.08E+00	9.54E+00

4.0 LIQUID EFFLUENTS

4.1 Liquid Effluent and Waste Disposal Report

4.1.1 ANO-1

Table 6, Liquid Effluents-Summation of All Releases (ANO-1)

		Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Est. Total Error %
A.	**Fission & Activation Products**						
1.	Total Release (not including tritium, gases or alpha)	Ci	3.49E-02	8.69E-03	5.74E-03	2.78E-03	21
2	Average diluted concentration during period	μCi/mL	1.12E-10	2.84E-11	1.46E-11	7.47E-12	
B.	**Tritium**						
1.	Total Release	Ci	2.37E+02	3.34E+01	3.16E+01	6.89E+01	12
2.	Average diluted concentration during period.	μCi/mL	7.64E-07	1.10E-07	8.05E-08	1.85E-07	
C.	**Dissolved & Entrained Gases**						
1.	Total Release	Ci	1.11E-02	5.18E-04	4.77E-05	2.17E-04	22
2.	Average diluted concentration during period	μCi/mL	3.57E-11	1.69E-12	1.22E-13	5.85E-13	
D.	**Gross Alpha Activity**						
1.	Total Release	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	27
E.	**Volume Of Waste Released (prior to dilution)**	Liters	2.48E+06	1.91E+06	2.58E+06	3.57E+05	
F.	**Volume Of Dilution Water Used During Period**	Liters	3.11E+11	3.05E+11	3.92E+11	3.72E+11	

% of limit is on the Radiological Impact to Man Table (Section 6.0)

Table 7, Batch Mode Liquid Effluents (ANO-1)

Nuclides Released		Batch Mode				
	Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
Na-24	Ci	2.16E-06	6.95E-04	0.00E+00	0.00E+00	6.97E-04
Cr-51	Ci	4.35E-03	6.66E-04	1.19E-04	5.65E-04	5.70E-03
Mn-54	Ci	2.88E-04	3.76E-05	1.83E-05	2.84E-05	3.72E-04
Fe-59	Ci	1.83E-05	9.49E-06	0.00E+00	0.00E+00	2.78E-05
Co-56	Ci	0.00E+00	2.78E-06	0.00E+00	0.00E+00	2.78E-06
Co-58	Ci	9.70E-03	2.19E-03	5.07E-04	5.88E-04	1.30E-02
Co-60	Ci	7.09E-03	1.36E-03	5.84E-04	5.99E-04	9.63E-03
Zn-65	Ci	0.00E+00	0.00E+00	0.00E+00	3.96E-06	3.96E-06
Sr-92	Ci	2.42E-05	0.00E+00	0.00E+00	1.83E-06	2.60E-05
Y-91m	Ci	8.63E-06	0.00E+00	0.00E+00	0.00E+00	8.63E-06
Zr-95	Ci	1.91E-03	3.85E-04	8.97E-05	8.97E-05	2.47E-03
Nb-95	Ci	3.31E-03	6.46E-04	1.43E-04	1.61E-04	4.26E-03
Nb-97	Ci	5.52E-06	2.17E-05	5.17E-06	0.00E+00	3.24E-05
Mo-99	Ci	0.00E+00	8.42E-05	0.00E+00	0.00E+00	8.42E-05
Tc-99m	Ci	0.00E+00	6.86E-05	0.00E+00	0.00E+00	6.86E-05
Ru-105	Ci	2.88E-05	2.83E-05	0.00E+00	0.00E+00	5.71E-05
Ag-110m	Ci	1.74E-03	2.22E-04	5.33E-05	4.37E-05	2.05E-03
Cd-113m	Ci	1.51E-05	0.00E+00	0.00E+00	0.00E+00	1.51E-05
Sn-117m	Ci	0.00E+00	5.27E-06	0.00E+00	0.00E+00	5.27E-06
Sb-122	Ci	0.00E+00	3.61E-05	0.00E+00	0.00E+00	3.61E-05
Sb-124	Ci	3.24E-03	1.22E-03	1.01E-03	1.07E-04	5.57E-03
Sb-125	Ci	1.53E-03	4.06E-04	6.00E-04	6.02E-05	2.59E-03
Te-132	Ci	0.00E+00	2.30E-07	0.00E+00	0.00E+00	2.30E-07
I-131	Ci	0.00E+00	1.65E-05	0.00E+00	0.00E+00	1.65E-05
I-133	Ci	0.00E+00	5.67E-07	0.00E+00	0.00E+00	5.67E-07
I-134	Ci	0.00E+00	1.12E-05	0.00E+00	0.00E+00	1.12E-05
Cs-134	Ci	7.45E-05	8.50E-05	8.43E-05	2.39E-05	2.68E-04
Cs-137	Ci	1.61E-03	4.92E-04	2.53E-03	5.04E-04	5.13E-03
Total For Period	**Ci**	**3.49E-02**	**8.69E-03**	**5.74E-03**	**2.78E-03**	**5.21E-02**
H-3	Ci	2.37E+02	3.34E+01	3.16E+01	6.89E+01	3.71E+02
Total For Period	**Ci**	**2.37E+02**	**3.34E+01**	**3.16E+01**	**6.89E+01**	**3.71E+02**
Ar-41	Ci	8.50E-08	0.00E+00	0.00E+00	0.00E+00	8.50E-08
Kr-88	Ci	0.00E+00	1.68E-05	1.05E-05	0.00E+00	2.73E-05
Xe-133		1.09E-02	5.01E-04	3.72E-05	1.51E-04	1.16E-02

Table 7, Batch Mode Liquid Effluents (ANO-1)						
Nuclides Released		**Batch Mode**				
Xe-135		1.88E-04	0.00E+00	0.00E+00	0.00E+00	1.88E-04
Xe-138		0.00E+00	0.00E+00	0.00E+00	6.65E-05	6.65E-05
Total For Period	**Ci**	**1.11E-02**	**5.18E-04**	**4.77E-05**	**2.17E-04**	**1.19E-02**

4.1.2 ANO-2

Table 8, Liquid Effluents-Summation of All Releases (ANO-2)

A.	Fission & Activation Products	Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Est. Total Error %
1.	Total Release (not including tritium, gases or alpha)	Ci	1.60E-03	1.32E-03	1.59E-03	3.23E-03	21
2	Average diluted concentration during period	µCi/mL	5.15E-12	4.32E-12	4.06E-12	8.70E-12	
B.	**Tritium**						
1.	Total Release	Ci	2.49E+02	1.04E+02	2.42E+02	7.08E+01	12
2.	Average diluted concentration during period.	µCi/mL	8.02E-07	3.42E-07	6.16E-07	1.91E-07	
C.	**Dissolved & Entrained Gases**						
1.	Total Release	Ci	7.63E-04	5.03E-04	1.44E-03	2.12E-03	22
2.	Average diluted concentration during period	µCi/mL	2.46E-12	1.65E-12	3.66E-12	5.70E-12	
D.	**Gross Alpha Activity**						
1.	Total Release	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00	27
E.	**Volume Of Waste Released (prior to dilution)**	Liters	7.53E+05	3.57E+05	2.24E+06	1.27E+06	
F.	**Volume Of Dilution Water Used During Period**	Liters	3.11E+11	3.05E+11	3.92E+11	3.72E+11	

% of limit is on the Radiological Impact to Man Table (Section 6.0)

Table 9, Batch Mode Liquid Effluents (ANO-2)

Nuclides Released		Batch Mode				
	Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
Cr-51	Ci	4.58E-05	0.00E+00	0.00E+00	6.29E-04	6.75E-04
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	4.33E-05	4.33E-05
Co-58	Ci	6.07E-05	4.51E-05	1.66E-04	1.80E-03	2.07E-03
Co-60	Ci	3.25E-05	2.29E-05	4.66E-04	6.55E-04	1.18E-03
Zr-95	Ci	1.75E-05	4.76E-06	0.00E+00	0.00E+00	2.23E-05
Nb-95	Ci	3.09E-05	1.02E-05	0.00E+00	7.12E-06	4.83E-05
Nb-97	Ci	0.00E+00	1.98E-06	5.86E-06	1.06E-05	1.84E-05
Ag-110m	Ci	6.27E-06	9.57E-06	9.13E-05	5.71E-05	1.64E-04
Sb-124	Ci	4.35E-05	5.78E-05	8.50E-06	0.00E+00	1.10E-04
Sb-125	Ci	7.56E-04	5.29E-04	2.10E-04	0.00E+00	1.49E-03
I-133	Ci	0.00E+00	0.00E+00	7.04E-06	0.00E+00	7.04E-06
I-134	Ci	0.00E+00	0.00E+00	3.04E-05	0.00E+00	3.04E-05
Cs-134	Ci	0.00E+00	0.00E+00	2.58E-06	0.00E+00	2.58E-06
Cs-137	Ci	6.08E-04	6.37E-04	6.06E-04	3.46E-05	1.89E-03
Total For Period	**Ci**	**1.60E-03**	**1.32E-03**	**1.59E-03**	**3.23E-03**	**7.75E-03**
H-3	Ci	2.49E+02	1.04E+02	2.42E+02	7.08E+01	6.66E+02
Total For Period	**Ci**	**2.49E+02**	**1.04E+02**	**2.42E+02**	**7.08E+01**	**6.66E+02**
Ar-41	Ci	0.00E+00	0.00E+00	0.00E+00	2.28E-05	2.28E-05
Xe-133	Ci	7.63E-04	5.03E-04	1.43E-03	2.08E-03	4.77E-03
Xe-135	Ci	0.00E+00	0.00E+00	1.13E-05	1.68E-05	2.81E-05
Total For Period	**Ci**	**7.63E-04**	**5.03E-04**	**1.44E-03**	**2.12E-03**	**4.82E-03**

Table 10, Continuous Mode Liquid Effluents (ANO-2)

Nuclides Released		Continuous Mode				
	Unit	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
H-3	Ci	0.00E+00	0.00E+00	3.61E-03	0.00E+00	3.61E-03
Total For Period	**Ci**	**0.00E+00**	**0.00E+00**	**3.61E-03**	**0.00E+00**	**3.61E-03**

5.0 SOLID WASTE SUMMARY

5.1 Solid Waste Shipped Offsite for Burial or Disposal (Not Irradiated Fuel) ANO-1

5.1.1 Types of Waste Summary

Table 11, Types of Solid Waste Summary (ANO-1)			
Types of Waste	**Total Quantity (m^3)**	**Total Activity (Ci)**	**Est. Total Error (%)**
a. Spent resins, filter sludges, evaporator bottoms, etc.	2.24E+01	2.25E-02	25
b. Dry compressible waste, contaminated equip, etc.	2.00E+02	7.18E-01	25
c. Irradiated components, control rods, etc.	0.00E+00	0.00E+00	-
d. Other	0.00E+00	0.00E+00	-

5.1.2 Estimate of major nuclide composition (by waste type) only >1% [Note 1] are reported.

Table 12, Major Nuclides Summary (ANO-1)		
Major Nuclide Composition	**%**	**Curies**
a. Spent resins, filter sludges, evaporator bottoms, etc.		
Co-60	1.35	3.02E-04
Cs-137	98.65	2.22E-02
b. Dry compressible waste, contaminated equip, etc.	**%**	**Curies**
C-14	4.46	3.21E-02
Cr-51	5.56	4.00E-02
Mn-54	2.58	1.86E-02
Fe-55	12.73	9.17E-02
Co-58	8.58	6.18E-02
Co-60	39.11	2.82E-01
Ni-63	14.09	1.02E-01
Zr-95	2.25	1.62E-02
Nb-95	4.38	3.16E-02
Cs-137	4.22	3.04E-02

Table 12, Major Nuclides Summary (ANO-1)		
c. Irradiated components, control rods, etc.	**%**	**Curies**
None	-	-
d. Other	**%**	**Curies**
None	-	-

[Note 1] – "Major" radionuclide is equivalent to a "principle" radionuclide, i.e., greater than 1 percent of total activity.

5.1.3 Solid Waste Disposition

Table 13, Solid Waste Disposition (ANO-1)		
Number of Shipments	**Mode of Transportation**	**Destination**
8	Hittman Transport	Bear Creek Operations.
2	Hittman Transport	Gallaher Road Operations

Table 14, Irradiated Fuel Shipments Disposition (ANO-1)		
Number of Shipments	**Mode of Transportation**	**Destination**
0	-	-

5.2 Solid Waste Shipped Offsite for Burial or Disposal (Not Irradiated Fuel) ANO-2

5.2.1 Types of Waste

Table 15, Types of Solid Waste Summary (ANO-2)			
Types of Waste	**Total Quantity (m^3)**	**Total Activity (Ci)**	**Est. Total Error (%)**
a. Spent resins, filter sludges, evaporator bottoms, etc.	1.44E+01	5.51E-03	25
b. Dry compressible waste, contaminated equip, etc.	1.97E+02	2.93E-01	25
c. Irradiated components, control rods, etc.	0.00E+00	0.00E+00	-
d. Other	0.00E+00	0.00E+00	-

5.2.2 Estimate of major nuclide composition (by waste type) only >1% [Note 1] are reported.

Table 16, Major Nuclides (ANO-2)		
Major Nuclide Composition	**%**	**Curies**
a. Spent resins, filter sludges, evaporator bottoms, etc.		
Co-60	43.65	2.41E-03
Cs-137	56.35	3.11E-03
b. Dry compressible waste, contaminated equip, etc.	**%**	**Curies**
C-14	3.49	1.03E-02
Cr-51	14.5	4.27E-02
Mn-54	2.33	6.87E-03
Fe-55	10.44	3.08E-02
Co-58	12.17	3.59E-02
Co-60	31.34	9.23E-02
Ni-63	11.03	3.25E-02
Zr-95	3.37	9.95E-03
Nb-95	6.14	1.81E-02
Cs-137	3.31	9.76E-03
c. Irradiated components, control rods, etc.	**%**	**Curies**
None	-	-

Table 16, Major Nuclides (ANO-2)		
d. Other	**%**	**Curies**
None	-	-

[Note 1] – “Major" radionuclide is equivalent to a “principle” radionuclide, i.e., greater than 1 percent of total activity.

5.2.3 Solid Waste Disposition

Table 17, Solid Waste Disposition (ANO-2)		
Number of Shipments	**Mode of Transportation**	**Destination**
5	Hittman Transport	Bear Creek Operations
1	Hittman Transport	Gallaher Road Operations
1	Landstar	Bear Creek Operations

Table 18, Irradiated Fuel Shipments Disposition (ANO-2)		
Number of Shipments	**Mode of Transportation**	**Destination**
0	-	-

5.3 Solid Waste Shipped Offsite for Burial or Disposal (Not Irradiated Fuel) ANO-Common

5.3.1 Types of Waste

Table 19, Types of Solid Waste Summary (ANO-Common)

Types of Waste	Total Quantity (m^3)	Total Activity (Ci)	Est. Total Error (%)
a. Spent resins, filter sludges, evaporator bottoms, etc.	0.00E+00	0.00E+00	-
b. Dry compressible waste, contaminated equip, etc.	1.10E+02	1.38E-01	25
c. Irradiated components, control rods, etc.	0.00E+00	0.00E+00	-
d. Other	1.87E+00	1.12E-02	25

5.3.2 Estimate of major nuclide composition (by waste type) only >1% [Note 1] are reported.

Table 20, Major Nuclides (ANO-Common)

Major Nuclide Composition	%	Curies
a. Spent resins, filter sludges, evaporator bottoms, etc.		
None	-	-
b. Dry compressible waste, contaminated equip, etc.	%	Curies
C-14	3.88	5.36E-03
Cr-51	9.98	1.38E-02
Mn-54	2.48	3.42E-03
Fe-55	11.43	1.58E-02
Co-58	11.08	1.53E-02
Co-60	34.58	4.78E-02
Ni-63	12.26	1.70E-02
Zr-95	3.01	4.16E-03
Nb-95	5.66	7.83E-03
Cs-137	3.68	5.08E-03

c. Irradiated components, control rods, etc.	%	Curies
None	-	-

d. Other	%	Curies
Co-60	3.05	3.42E-04
Cs-137	96.95	1.09E-02

[Note 1] – “Major" radionuclide is equivalent to a “principle” radionuclide, i.e., greater than 1 percent of total activity.

5.3.3 Solid Waste Disposition

Table 21, Solid Waste Disposition (ANO-Common)		
Number of Shipments	**Mode of Transportation**	**Destination**
1	Barnhart Crane & Rigging	Energy Solutions (Memphis)
5	Hittman Transport	Bear Creek Operations

Table 22, Irradiated Fuel Shipments Disposition (ANO-Common)		
Number of Shipments	**Mode of Transportation**	**Destination**
0	-	-

6.0 RADIOLOGICAL IMPACT TO MAN

6.1 10 CFR Part 50, Appendix I Evaluation

6.1.1 ANO–1 Assessment

Table 23, Dose Assessment (ANO-1)

	Quarter 1	**Quarter 2**	**Quarter 3**	**Quarter 4**	**Annual**
Liquid Effluent Dose Limit, Total Body	1.5 mrem	1.5 mrem	1.5 mrem	1.5 mrem	3 mrem
Total Body Dose	1.23E-03	3.96E-04	1.08E-03	3.13E-04	3.02E-03
% of Limit	**8.21E-02**	**2.64E-02**	**7.20E-02**	**2.08E-02**	**1.01E-01**
Liquid Effluent Dose Limit, Any Organ	5 mrem	5 mrem	5 mrem	5 mrem	10 mrem
Maximum Organ Dose	1.63E-03	5.39E-04	1.61E-03	4.23E-04	4.20E-03
% of Limit	**3.26E-02**	**1.08E-02**	**3.22E-02**	**8.46E-03**	**4.20E-02**
Gaseous Effluent Dose Limit, Gamma Air	5 mrad	5 mrad	5 mrad	5 mrad	10 mrad
Gamma Air Dose	0.00E+00	1.12E-04	0.00E+00	0.00E+00	1.12E-04
% of Limit	**0.00E+00**	**2.25E-03**	**0.00E+00**	**0.00E+00**	**1.12E-03**
Gaseous Effluent Dose Limit, Beta Air	10 mrad	10 mrad	10 mrad	10 mrad	20 mrad
Beta Air Dose	0.00E+00	4.18E-05	0.00E+00	0.00E+00	4.18E-05
% of Limit	**0.00E+00**	**4.18E-04**	**0.00E+00**	**0.00E+00**	**2.09E-04**
Gaseous Effluent Organ Dose Limit (Iodine, Tritium, Particulates with > 8 day half-life)	7.5 mrem	7.5 mrem	7.5 mrem	7.5 mrem	15 mrem
Gaseous Effluent Organ Dose (Iodine, Tritium, Particulates with > 8 day half-life)	1.14E-02	2.08E-02	1.90E-02	1.72E-02	6.83E-02
% of Limit	**1.51E-01**	**2.78E-01**	**2.53E-01**	**2.29E-01**	**4.56E-01**

6.1.2 ANO–2 Assessment

Table 24, Dose Assessment (ANO-2)

	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Annual
Liquid Effluent Dose Limit, Total Body	1.5 mrem	1.5 mrem	1.5 mrem	1.5 mrem	3 mrem
Total Body Dose	6.32E-04	5.59E-04	5.67E-04	1.21E-04	1.88E-03
% of Limit	**4.21E-02**	**3.73E-02**	**3.78E-02**	**8.07E-03**	**6.26E-02**
Liquid Effluent Dose Limit, Any Organ	5 mrem	5 mrem	5 mrem	5 mrem	10 mrem
Maximum Organ Dose	7.75E-04	7.57E-04	6.94E-04	1.28E-04	2.35E-03
% of Limit	**1.55E-02**	**1.51E-02**	**1.39E-02**	**2.56E-03**	**2.35E-02**
Gaseous Effluent Dose Limit, Gamma Air	5 mrad	5 mrad	5 mrad	5 mrad	10 mrad
Gamma Air Dose	0.00E+00	0.00E+00	2.08E-03	2.06E-02	2.26E-02
% of Limit	**0.00E+00**	**0.00E+00**	**4.17E-02**	**4.11E-01**	**2.26E-01**
Gaseous Effluent Dose Limit, Beta Air	10 mrad	10 mrad	10 mrad	10 mrad	20 mrad
Beta Air Dose	0.00E+00	0.00E+00	2.50E-03	7.25E-03	9.75E-03
% of Limit	**0.00E+00**	**0.00E+00**	**2.50E-02**	**7.25E-02**	**4.88E-02**
Gaseous Effluent Organ Dose Limit (Iodine, Tritium, Particulates with > 8 day half-life)	7.5 mrem	7.5 mrem	7.5 mrem	7.5 mrem	15 mrem
Gaseous Effluent Organ Dose (Iodine, Tritium, Particulates with > 8 day half-life)	1.89E-02	2.91E-02	3.62E-02	5.04E-02	1.35E-01
% of Limit	**2.52E-01**	**3.88E-01**	**4.82E-01**	**6.72E-01**	**8.97E-01**

6.2 40 CFR Part 190 Evaluation for an Individual in the Unrestricted Area

Table 25, EPA 40 CFR PART 190 Evaluation

	Whole Body	Thyroid	Any Other Organ
Dose Limit	25 mrem	75 mrem	25 mrem
Dose	1.75E+00	1.73E+00	2.39E+00
% of Limit	6.98E+00	2.30E+00	9.56E+00

Liquid dose, gaseous dose including C14, direct shine from each unit, ISFSI and any other nuclear power related facility within 5 miles of the station are considered when calculating dose compliance with 40 CFR 190.

6.3 40 CFR Part 190 Calculation

Table 26, EPA 40 CFR Part 190 Calculation

	Unit	Total Body	Thyroid	Max Organ
Routine Airborne Effluents[Note 1]	Unit 1	7.48E-05	-	1.21E-04
Routine Airborne Effluents ITP	Unit 1	6.83E-02	6.83E-02	6.83E-02
Routine Liquid Effluents	Unit 1	3.02E-03	5.71E-04	4.20E-03
Airborne Releases of C^{14}	Unit 1	3.31E-02	3.31E-02	1.77E-01
Routine Airborne Effluents[Note 1]	Unit 2	1.51E-02	-	2.54E-02
Routine Airborne Effluents ITP	Unit 2	1.35E-01	1.35E-01	1.35E-01
Routine Liquid Effluents	Unit 2	1.88E-03	9.63E-04	2.35E-03
Airborne Releases of C^{14}	Unit 2	1.19E-01	1.19E-01	6.09E-01
Ground Water & Storm Drain Totals	Site	0.00E+00	0.00E+00	0.00E+00
Direct Shine from areas such as dry cask storage, radwaste storage, Equipment Mausoleums[Note 2]	Site	1.37E+00	1.37E+00	1.37E+00
Total 40 CFR 190 Dose	Site	1.75E+00	1.73E+00	2.39E+00

[Note 1]: Routine airborne dose in this table is mrad expressed as mrem. This addition does not represent a real dose and is listed here solely to help demonstrate compliance with 40 CFR 190.

[Note 2]: Average direct radiation control location TLD was compared to the average area of an "actual" person TLD monitoring area [Reeves E. Richie Training Center (RERTC)] to determine yearly dose from direct shine to an actual member of the public. Since constant occupation is unrealistic, a value of 52 weeks a year * 40 hours in a week occupying results in a total of 2080 hours. Control Value = 25.9 mrem/yr. RERTC Value = 31.7 mrem/yr. Weighted = 2080/8784(hrs.) = 23.68%. Actual Dose = (31.7 – 25.9) * 0.237 = 1.37 mrem

7.0 METEOROLOGICAL DATA

7.1 Joint Frequency Distributions

1. Period of Record: 01/01/2021 - 12/31/2021
2. Elevation: 57 m

Table 27, Percentage of Each Wind Speed/Direction

	Wind Speed (mph)								
Wind Direction	**0 – 2**	**2 – 4**	**4 – 6**	**6- 8**	**8 – 10**	**10 – 15**	**15 - 20**	**>20**	**Total**
N	1.87	0.41	0.51	0.80	0.40	0.15	0.00	0.00	**4.14**
NNE	0.72	0.59	0.74	0.73	0.51	0.11	0.00	0.00	**3.41**
NE	1.40	1.32	1.67	1.30	0.51	0.13	0.00	0.00	**6.33**
ENE	2.59	2.24	3.59	3.07	1.13	0.29	0.03	0.00	**12.94**
E	2.27	2.14	3.02	3.23	2.54	2.14	0.18	0.00	**15.54**
ESE	1.21	1.48	2.04	1.92	1.61	1.32	0.06	0.00	**9.64**
SE	0.74	1.04	1.52	1.94	1.07	0.48	0.00	0.00	**6.79**
SSE	0.62	1.00	1.49	1.68	1.47	0.83	0.05	0.00	**7.14**
S	0.50	0.63	0.76	1.27	0.74	0.54	0.00	0.00	**4.44**
SSW	0.39	0.31	0.43	0.57	0.46	0.29	0.00	0.00	**2.44**
SW	0.49	0.32	0.33	0.25	0.30	0.34	0.02	0.00	**2.05**
WSW	0.63	0.44	0.38	0.18	0.19	0.26	0.09	0.01	**2.19**
W	0.94	0.94	1.02	0.60	0.58	0.95	0.42	0.18	**5.62**
WNW	0.75	0.81	1.05	1.19	1.24	2.05	0.99	0.29	**8.37**
NW	0.72	0.41	0.26	0.26	0.55	0.73	0.19	0.02	**3.15**
NNW	0.39	0.38	0.32	0.54	0.33	0.41	0.05	0.00	**2.41**

3. Variable
   a. Total period of calm hours: 5.17%
   b. Percentage of missing data: 3.41%

7.2 Stability Class

Table 28, Classification of Atmospheric Stability

Stability Condition	Pasquill Categories	Percentage
Extremely Unstable	A	0.55
Moderately Stable	B	0.06
Slightly Unstable	C	0.02
Neutral	D	42.58
Slightly Stable	E	40.79
Moderately Stable	F	11.46
Extremely Stable	G	1.13

ENCLOSURE, ATTACHMENT 1

0CAN042202

OFFSITE DOSE CALCULATION MANUAL

ARKANSAS NUCLEAR ONE

OFFSITE DOSE CALCULATION MANUAL

REVISION 30

Changes are indicated by beginning the affected information with a revision bar on the right side of the page which stops at the end of the change. Deletions of entire paragraphs or sections have a revision bar to the right of the page where text was deleted. The amendment number is indicated at the bottom of the affected page near the left margin and indicates the latest revision to the information contained on that page. Absence of a revision bar on a replacement page means the page was reprinted for word processing purposes only. However, general formatting changes may have been made to all pages.

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1.0 INTRODUCTION

The Offsite Dose Calculation Manual (ODCM) provides guidance for making release rate and dose calculations for radioactive liquid and gaseous effluents from Arkansas Nuclear One – Units 1 and 2 (ANO-1 and ANO-2). The methodology is drawn from NUREG-0133, Rev. 0. Parameters contained within this manual were taken from NUREG-0133 and Regulatory Guide (RG) 1.109 except as noted for site specific values. These numbers and the calculational method may be changed as provided for in the Technical Specifications (TSs).

The following references are utilized in conjunction with the limitations included in this manual concerning the indicated subjects:

Subject	ANO-1	ANO-2
Process Control Program (PCP)	EN RW-105	EN RW-105
Radioactive Effluent Controls Program	TS 5.5.4	TS 6.5.4
Annual Radiological Environmental Monitoring Report	TS 5.6.2	TS 6.6.2
Radioactive Effluent Release Report	TS 5.6.3	TS 6.6.3
ODCM	TS 5.5.1	TS 6.5.1

2.0 LIQUID EFFLUENTS

2.1 Radioactive Liquid Effluent Monitor Setpoint

ODCM Limitation L 2.1.1, "Radioactive Liquid Effluent Instrumentation," requires that the radioactive liquid effluents be monitored with the alarm/trip setpoints adjusted to ensure that the limits of the radioactive liquid effluent concentration limitations are not exceeded. These concentrations are for the site. The alarm/trip setpoint on the liquid effluent monitor is dependent upon the dilution water flow rate, radwaste tank flow rate, isotopic composition of the radioactive liquid to be discharged, a gross gamma count of the liquid to be discharged, background count rate of the monitor, and the efficiency of the monitor. Due to the fact that these are variables, an adjustable setpoint is used. The setpoint must be calculated and the monitor setpoint set prior to the release of each batch of radioactive liquid effluents. The following methodology is used for the setpoint determination for the following monitors.

ANO-1: RE-4642 – Liquid Radwaste Monitor

ANO-2: 2RE-2330 – Liquid Radwaste Monitor
2RE-4423 – Liquid Radwaste Monitor

1) A sample from each tank (batch) to be discharged is obtained and counted for gross gamma (Cs-137 equivalent) and a gamma isotopic analysis is performed.

2) A dilution factor (DF) for the tank is calculated based upon the results of the gamma isotopic analysis and the Maximum Permissible Concentration (MPC) of each detected radionuclide.

DF is calculated as follows:

$$DF = \Sigma_i(C_i/MPC_i) + C_{TNG}/MPC_{TNG}$$

where:

- DF = dilution factor;
- C_i = concentration of isotope "i", (µCi/ml);
- MPC_i = maximum permissible concentration of isotope "i", (from 10 CFR 20, Appendix B, Table II, Column 2 in µCi/ml);
- C_{TNG} = total concentration of noble gases (µCi/ml); and
- MPC_{TNG} = 2×10^{-4} (µCi/ml) per Limitation L 2.3.1.a

3) The dilution water flowrate is normally the number of ANO-1 circulating water pumps in operation at the time of release. Each circulating water pump has an approximate flowrate of 191,500 gallons per minute (gpm) (this flowrate may be reduced due to throttling of circulating water pump flow and/or circulating water bay configuration). However, under specific conditions and under strict controls, lower dilution water flowrates utilizing service water and cooling tower blowdown flowrates may be used.

4) The theoretical release rate, F_m, of the tank (batch) to be released is expressed in terms of the dilution water flowrate, such that for each volume of dilution water released, a given volume of liquid radwaste may be combined. This may be expressed as follows:

$$F_m = DV/DF$$

where:

- F_m = theoretical release rate (gpm);
- DV = Dilution volume (gpm). When ANO-1 circulating water pumps are running, DV is the number of ANO-1 circulating water pumps in operation multiplied by the approximate flowrate of an ANO-1 circulating water pump (normally 191,500 gpm) or an indicated flow rate. The minimum total flow rate shall be greater than or equal to 100,000 gpm. Otherwise DV is dilution volume provided by service water and cooling tower blowdown flowrate; and
- DF = dilution factor as calculated in Step 2 above.

Note: In the above equation, the theoretical release rate (F_m) approaches zero as the dilution factor increases. The actual flowrate (F_A) will normally be equal to the theoretical release rate for high activity releases. For low activity releases, the theoretical release rate becomes large and may exceed the capacity of the pump discharging the tank. In these cases, the actual release rate may be set to the maximum flowrate of the discharge pump.

5) The monitor setpoint is calculated by incorporating the monitor reading prior to starting the release (i.e., background countrate), and a factor which is the amount of increase in the release concentration that would be needed to exceed the radioactive liquid concentration limitation. The monitor setpoint is expressed as follows:

$$M_L = A^*(K^*F_M/F_A) + B$$

where:

M_L = monitor setpoint (counts per minute or "cpm");

A = allocation fraction for the specific unit. (Typically, these values are set at 0.45, but may be adjusted up or down as needed. However, the total site allocation can not exceed 1.0.)

K = monitor countrate (cpm) expected based on the gross activity of the release (this value is obtained from a graph of activity (µCi/ml) versus output countrate for the monitor (cpm));

F_M/F_A = number of times the activity would have to increase to exceed the radioactive liquid effluent-concentration limitation; and

B = background countrate (cpm) prior to the release.

To permit the computer to calculate the setpoint, an equation for the expected countrate (K) is expressed as follows:

$$K = \text{Offset} * S_A^{\text{Slope}}$$

where:

$$\text{Slope} = \frac{\text{Log of the detector response in cpm}}{\text{Log of activity concentration in } \mu\text{Ci/ml}}$$

S_A = Gross gamma (Cs-137 equivalent) activity for the tank (µCi/ml); and

Offset = detector response (cpm) for the minimum detectable sample activity calculated from the calibration data.

Note: I&C personnel use varying concentrations of Cs-137 to determine the response curve; therefore, a Cs-137 equivalent activity must be used to accurately predict the countrate.

Combining terms, the equation for determining the monitor setpoint may be expressed as follows:

$$M_L = A[(\text{Offset} * S_A^{\text{Slope}})F_M/F_A] + B$$

2.2 Liquid Dose Calculation

The "dose" or "dose commitment" to an individual in the unrestricted area shall be less than or equal to the limits specified in 'Radioactive Liquid Effluents – Dose' Limitations. The dose limits are on a per reactor basis. This value is calculated using the adult as the maximum exposed individual via the aquatic foods (Sport Freshwater Fish) and the potable water pathways.

2.2.1 Dose Calculations for Aquatic Foods

The concentrations of radionuclides in aquatic foods are assumed to be directly related to the concentrations in water. The equilibrium ratios between the two concentrations are called “bioaccumulation factors.”

Two different pathways are calculated for aquatic foods: sport and commercial freshwater fish.

The internal dose "d" from the consumption of aquatic foods in pathway "p" to organ "j" of individuals of age group "a" from all nuclides "I" is computed as follows (see Chapter 4 of NUREG-0133 and RG 1.109-12, equation A-3):

$$d_p\ (r,\theta,a,j) = \Sigma_i[\{(1100)(e^{-\lambda_i t_p})(B_i)\}(M)(U_a)(F)^{-1}(Q_i)(D_{aij})]$$

The total dose from both aquatic food pathways is then:

$$D(r,\theta,a,j) = \sum_{P} d_p\ (r,\theta,a,j)$$

where:

- r = user-selected distance from the release point to the receptor location, in kilometers. It may be different from the controlling distance specified for the potable water pathway (0.4 km);
- θ = user-selected sector (one of sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc.). This sector may be different from the controlling sector specified for the potable water pathway (S);
- A = user-selected age group: infant, child, teen, adult. It is the same controlling age group used in the potable water pathway (adult);
- J = user-selected organ: bone, liver, total body, thyroid, kidney, lung, GI-LLI. It is the same controlling organ used in the potable water pathway (liver);
- { } = represents the concentration factor stored in the database;

  Note: Only one concentration factor is needed to represent the two pathways since sport and commercial use the same bioaccumulation factor for a given pathway.

- 1100 = factor to convert from (Ci/yr)/(ft^3/sec) to ρCi/liter;
- λ_i = decay constant of nuclide "I" in hr^{-1};
- t_p = environmental transit time, release to receptor;

  Note: This value should be set to 0 hours (i.e., no decay correction) for the above equation in order to be consistent with the equation presented in Chapter 4 of NUREG-0133. For maximum individual dose calculations, this value is set to 24 hours, which is the minimum transit time recommended by RG 1.109, Appendix A, 2.b.

- B_i = bioaccumulation factor for nuclide "I", in ρCi/kg per ρCi/liter. Cesium has a site specific number based on carnivorous and bottom feeder sport fish of 400 ρCi/kg per ρCi/liter (0CAN048408, dated April 13, 1984); Niobium has a site specific number based upon freshwater fish of 300 ρCi/kg per ρCi/liter.
- M = dimensionless mixing ratio (reciprocal of the dilution factor) at the point of exposure;

U_a = annual usage factor that specifies the intake rate for an individual of age group "a", in kilograms/year. The program selects this usage factor in accordance with the controlling age group "a" as specified previously by the user;

F = average flow rate in ft^3/sec. This value is based on total dilution volume for the quarter divided by time into the quarter;

Q_i = number of curies of nuclide "I" released; and

D_{aij} = ingestion dose factor for age group "a", nuclide "I", and organ "j", in mrem per ρCi ingested. The program selects the ingestion dose factor according to the user-specified controlling age group "a" and controlling organ "j".

2.2.2 Dose Calculations for Potable Water

The dose “D” from ingestion of water to organ "j" of individuals of age group "a" due to all nuclides "I" is calculated as follows (See Chapter 4 of NUREG-0133 and NRC RG 1.109-12, equation A-2):

Note: The potable water pathway is used only during the time that the Russellville Water System is using the Arkansas River as a water source. The Russellville Water Works will notify ANO when they are using the Arkansas River as a water source.

$$D\,(r,\theta,a,j) \;=\; \Sigma_i\,[\{(1100)(e^{-\lambda_i t_p})\}(M)(U_a)(F^{-1})(Q_i)(D_{aij})]$$

where:

r = user-selected distance (0.4 km) from the release point to the receptor location, in kilometers. It may be different from the controlling distance selected for the aquatic food pathway;

θ = user-selected sector; (one of the sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc.). It may be different from the controlling sector for the aquatic food pathway;

a = user-selected age group (infant, child, teen, adult). The same controlling age group is used for all liquid pathways (adult);

j = user-selected organ (bone, liver, total body, thyroid, kidney, lung, GI-LLI). The same controlling organ is used for all liquid pathways (liver).

{ } = the expression in brackets represents the concentration factor stored in the database;

1100 = factor to convert from (Ci/yr)/(ft^3/sec) to ρCi/liter;

M = dimensionless mixing ratio (reciprocal of the dilution factor) at the point of exposure;

λ_i = decay constant of nuclide "I" in hr^{-1}; and

t_p = environmental transit time, release to receptor.

Note: This value is set to 0 hours (i.e., no decay correction) for the above equation to be consistent with the equation presented in Chapter 4 of NUREG-0133.

U_a = annual usage factor that specifies the intake rate for an individual of age group "a", in liters/year. The program selects this usage factor according to the user-specified controlling age group "a";

F = average flow rate in ft^3/sec; this value is based on total dilution volume for one quarter divided by time into the quarter;

Q_i = number of curies of nuclide "I" in the release; and

D_{aij} = ingestion dose factor, for age group "a", nuclide "I", and organ "j", in mrem per ρCi ingested. The program selects the ingestion dose factor according to the user-specified controlling age group "a" and controlling organ "j".

2.3 Liquid Projected Dose Calculation

The quarterly projected dose is based upon the methodology of Section 2.2 and is expressed as follows:

$$D_{QP} = 92(D_{QC} + D_{RP})/T$$

where:

D_{QP} = quarterly projected dose (mrem);

92 = number of days per quarter;

D_{QC} = cumulative dose for the quarter (mrem);

D_{RP} = dose for current release (mrem); and

T = current days into quarter;

3.0 GASEOUS EFFLUENTS

3.1 Gaseous Monitor Setpoints

Note: Sections 3.1.1 and 3.1.2 below detail two methods of calculating setpoints at ANO. These methods cover two different sets of monitors of which only one will be in-service at any one time.

3.1.1 Batch Release Setpoint Calculations

3.1.1.a This section applies to the following gaseous radiation monitors (these releases are also monitored by the SPING monitors in Section 3.1.2):

ANO-1: RE-4830 – Waste Gas Holdup System Monitor*

RX-9820 – Reactor Building Purge and Ventilation SPING

ANO-2: 2RE-8233 – Containment Building Purge Monitor*

2RE-2429 – Waste Gas Holdup System Monitor*

2RX-9820 – Containment Building Purge and Ventilation SPING

* These monitors provide automatic isolation.

The setpoints to be used during a batch type of release (i.e., Reactor Building [Containment] Purge, release from the Waste Gas Holdup System or any other non-routine release) will be calculated for each release before it occurs.

3.1.1.b The basic methodology for determining a monitor setpoint is based upon the expected concentration at the monitor (C_M). This is in turn based upon the fraction of an MPC assigned to this release point. Batch releases are maintained below the assigned MPC fraction by controlling the release rate. The calculated value of S may not exceed the equivalent of 1 MPC at site boundary. If value of S for RX (2RX) -9820 is less than SPING Channel 5 high alarm setpoint, then the high alarm setpoint may be used as a default value. If the value of S for RE-4830 and 2RE-2429 is less than 50,000 cpm, then 50,000 cpm may be used as a minimum setpoint. If the value of S for 2RE-8233 is less than 1,000 cpm, then 1,000 cpm may be used as a minimum setpoint.

$$S = 1.2(C_M)(K) + (2.0)(B)$$

where:

S = monitor setpoint (cpm or μCi/cc);

C_M = Xe-133 equivalent concentration at the monitor (μCi/cc);

K = conversion factor determined from response curve of monitor (cpm per μCi/cc). This value is 1.0 when calculating S for RX (2RX) -9820.

2.0 = factor to accommodate random count rate fluctuations;

B = background count rate at the monitor (cpm (or μCi/cc for RX-9820)).

1.2 = Safety Factor to correct for instrument uncertainties.

3.1.2 SPING (Final Effluent) Monitor Setpoint Calculations

3.1.2.a This section applies to the following gaseous radiation monitors:

ANO-1: RX-9820 – Reactor Building Purge and Ventilation SPING

RX-9825 – Auxiliary Building Ventilation SPING

RX-9830 – Spent Fuel Pool Area Ventilation SPING

RX-9835 – Emergency Penetration Room Ventilation SPING

ANO-2: 2RX-9820 – Containment Building Purge and Ventilation SPING

2RX-9825 – Auxiliary Building Ventilation SPING

2RX-9830 – Spent Fuel Pool Area Ventilation SPING

2RX-9835 – Emergency Penetration Room Ventilation SPING

2RX-9845 – Auxiliary Building Extension Ventilation SPING

2RX-9850 – Radwaste Storage Building Ventilation SPING

The determination of setpoints for the above monitors is based on an assigned fraction of the MPC of noble gas activity at the site boundary (Xe-133 equivalent) released from the above release points. The total of these fractions is always less than 1.00. The assigned fractions are based on the vent flow rates, atmospheric dilution rate, and the ventilation system(s) in operation.

Note: The fact that an effluent monitor is in alarm does not necessarily mean that radioactive gases are being released at such a rate that the MPC limit is being exceeded. The alarm would indicate that radioactive gases are being released at a rate that is exceeding the fractional allocation of an MPC allotted to that particular release point. Consideration must be given to the release rate of radioactive gases via all of the release pathways.

The initial fractions of an MPC allocated to the release points are given below. The allocations may be changed as needed, to allow for operational transients, but may not exceed a site total of 1.00.

Monitor Number	Monitor Name	Fractional Allocation
RX-9820	Reactor Building Purge and Ventilation	0.1000
RX-9825	Auxiliary Building Ventilation	0.2000
RX-9830	Spent Fuel Pool Area Ventilation	0.1500
RX-9835	Emergency Penetration Room Ventilation	0.0001

Monitor Number	Monitor Name	Fractional Allocation
2RX-9820	Containment Building Purge and Ventilation	0.1000
2RX-9825	Auxiliary Building Ventilation	0.2000
2RX-9830	Spent Fuel Pool Area Ventilation	0.1500
2RX-9835	Emergency Penetration Room Ventilation	0.0001
2RX-9845	Auxiliary Building Extension Ventilation	0.0100
2RX-9850	Radwaste Storage Building Ventilation	0.0100

Note: The setpoints to be used during a batch release (i.e., Reactor Building [Containment] Purge or Waste Gas Holdup System) will be calculated for each release before it occurs.

3.1.2.b SPING monitor setpoints may be calculated as follows:

$$\text{Setpoint } (\mu\text{Ci/cc}) = A\left(\frac{\text{Xe-133 eq } (\mu\text{Ci/cc})}{F(9.4390\text{E-}9)(\text{TMPC})}\right)$$

where:

A = allocation fraction (the fraction of an MPC at the site boundary (of noble gas Xe-133 eq activity) assigned to the particular release point);

Xe-133 eq = Xenon-133 equivalent concentration;

F = discharge flow of the particular release point in cubic feet per minute (cfm)

$$9.4390\text{E-}9 = 2.8317\text{E-}2(\text{cm/cf})\left(\frac{2.0\text{E-}5(\text{sec/m}^3)}{60(\text{sec/min})}\right)$$

where:

2.0E-5 = the annual average gaseous dispersion factor (corrected for radioactive decay) as defined in Section 2.3 of the ANO-1/ANO-2 Safety Analysis Report (SAR); and

TMPC = total MPCs at site boundary.

3.2 Airborne Release Dose Rate Effects

3.2.1 Noble Gas Release Rate

3.2.1.a To calculate the noble gas release dose rate, the average ground-level concentration of radionuclide "I" at the receptor location must first be determined from the following equation (see RG 1.109-20 equation B-4).

$$x_i\,(\theta) = (3.17 \times 10^4)(Q_i)[D1X/Q(\theta)]$$

where:

$x_i\,(\theta)$ = average ground level concentration in ρCi/m^3 of nuclide "I" at the user-specified controlling distance in sector θ (1.05 km);

(θ) = one of the sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc. (WNW);

3.17×10^4 = number of ρCi per Ci divided by the number of seconds/year;

Q_i = release rate of nuclide "I" in curies/yr and

$D1X/Q(\theta)$ = annual average gaseous dispersion factor (corrected for radioactive decay) in the sector at angle "θ" at the receptor location in sec/m^3. This value is 2.0E-5 sec/m^3 for short term releases.

The annual dose to the total body and skin due to noble gas can be calculated according to Sections 3.1.2.b and 3.2.1.c.

3.2.1.b Annual Total Body Dose Rate

The annual average total body dose rate to the maximally exposed individual is calculated as follows:

$$D^T(\theta) = (RBPF)(S_F)(\Sigma_i\,[x_i(\theta) * DFB_i]$$

where:

$D^T(\theta)$ = total body dose rate due to immersion in a semi-infinite cloud of gas at the controlling distance in sector "θ", in mrem/yr. The program computes one total body dose rate value for each sector in which the user has specified a controlling distance and reports only the maximum value;

θ = one of sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc. (WNW);

RBPF = Reactor Building (Containment) Purge Factor – This factor is used to calculate the length of time (fractional duty cycle) that the purge fans will be in operation. It is calculated by comparing the highest dose rate (DOSER) to its applicable release limit, taking into account the allocation factor for the release point (RBPF = Allocation * Limit/DOSER). This factor is calculated only for ANO-1 and ANO-2 Reactor Building (Containment) purges. For all other releases, this factor is set to 1.0;

S_F = dimensionless attenuation factor accounting for the dose reduction due to shielding by residential structures. The NRC recommended value is 0.7 (for maximum individual)

$x_i(\theta)$ = average ground-level concentration of nuclide "I" at the receptor location in the sector at angle "θ" from the release point, as defined in Section 3.2.1.a; and

DFB_i = total body dose factor for a semi-infinite cloud of radionuclide "I", which includes the attenuation of 5 g/cm² of tissue, in mrem-m³/ρCi-yr

3.2.1.c Annual Skin Dose Rate

The annual dose rate to the skin of the maximally exposed individual due to noble gases is calculated as follows (see RG 1.109-20 equation B-9):

$$D^S(\theta) = RBPF[(1.11)(S_F)(\Sigma_i(x_i(\theta))(DF_i^\gamma) + \Sigma_i(x_i(\theta))(DFS_i)]$$

where:

$D^S(\theta)$ = skin dose due to immersion in a semi-infinite cloud of gas at the user-specified controlling distance in sector "θ", in mrem;

Note: The program computes a skin dose value for each sector in which the user as specified a controlling distance, but prints out only the maximum value.

RBPF = Reactor Building [Containment] Purge Factor as defined in Section 3.2.1.b.

1.11 = average ratio of tissue to air energy absorption coefficient;

S_F = dimensionless attenuation factor accounting for the dose reduction due to shielding by residential structures. The value is 0.7 (for maximum individual);

$x_i(\theta)$ = is the average ground-level concentration of nuclide “i” at the receptor location in the sector at angle “θ“ from the release point, as defined in Section 3.2.1;

θ = one of sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc. (WNW);

DF^γ = gamma air dose factor for a semi-infinite cloud of radionuclide "I", in mrad-m³/ρCi-yr; and

DFS_i = beta skin dose factor for a semi-infinite cloud of radionuclide "I", which includes the attenuation by the outer "dead" layer of skin, in mrem-m³/ρCi-yr.

3.2.2 I-131, Tritium and Particulate Release Dose Rate Effects

The annual dose rate to the maximally exposed individual for I-131, tritium and radionuclides in particulate form with half-lives greater than eight days is calculated as follows:

$$DR^{TOT} = (RBPF)(DR^{I} + DR^{G} + DR^{M})$$

where:

- RBPF = Reactor Building (Containment) Purge Factor as defined in Section 3.2.1.b;
- DR^{I} = dose rate to the controlling age group (infant) associated with the inhalation of radioiodines and particulates, as calculated in Section 3.4.1.b;
- DR^{G} = dose rate from direct exposure to activity deposited on the ground plane, as calculated in Section 3.4.1.a; and
- DR^{M} = dose rate to the controlling age group (infant) and the controlling organ for ingestion of food (milk), as calculated in Section 3.4.1.d.

Calculation of the annual dose rate considers the infant as the most restrictive age group. The organs that are considered as contributing to the dose rate are: skin, bone, liver, total body, thyroid, kidney, lung, and GI-LLI. The food pathway for the infant is considered to be from milk only. All three pathways will contribute to the total body dose, while the skin will be affected by only the ground plane pathway. The other organs are affected only by the inhalation and food pathways.

3.3 Dose Due to Noble Gases

The air dose in unrestricted areas due to noble gases released in gaseous effluents shall be less than or equal to 5 mrad for gamma radiation and 10 mrad for beta radiation for any calendar quarter for each unit. The objective of less than or equal to 10 mrad of gamma radiation and 20 mrad of beta radiation for a calendar year per unit (2.5 mrad and 5 mrad respectively per quarter) should be used for planning releases.

Note: The following equations have been simplified from equations in NUREG-0133, Revision 0, in that there are no free-standing stacks at ANO. The equations were further simplified in that there are no long term (i.e., continuous) releases. The individual stack vents are sampled weekly, or are assigned a release period of 168 hours per sample (i.e., considered as short term (batch) releases). Individual samples are to be taken for each waste gas release and Reactor (Containment) Building purge.

3.3.1 Beta and Gamma Air Doses from Noble Gas Releases

Using the average ground level concentration of radionuclide "I" at the receptor location calculated in Section 3.2.1.a, the associated annual gamma or beta air dose may be calculated by the following equation (see RG 1.109-20 equation B-5).

$$D^{\gamma}(\theta) \ \underline{\text{or}} \ D^{\beta}(\theta) = \Sigma_i \left[(x_i(\theta))(DF_i^{\gamma} \ \underline{\text{or}} \ DF_i^{\beta})\right]$$

where:

- $D^{\gamma}(\theta)$ or $D^{\beta}(\theta)$ = the gamma or beta air dose for the controlling distance in sector "θ" (only the maximum value is reported), and
- DF_i^{γ} or DF_i^{β} = gamma or beta air dose factors for a uniform semi-annual infinite cloud of nuclide "I", in mrad-m^3/ρCi-yr.

3.4 Dose Due to I-131, Tritium, and Particulates in Gaseous Effluents

The calculational methodology for determining the dose to an individual from I-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to unrestricted areas as specified in the Limitations is in this section.

The child is the controlling age group unless stated otherwise.

The inhalation and ground plane pathways are considered to exist at all locations. The grass-cow-milk, grass-cow-meat, and vegetation pathways are used where applicable.

It is assumed that iodines are in the elemental form.

A dispersion parameter of 2.0E-5 sec/m^3 (per ANO-1/ANO-2 SAR, Section 2.3) is used for "w" in the inhalation pathway since the majority of gaseous activity released from the site is within the 8 to 24 hours time frame (i.e., Reactor Building [Containment] purges and Waste Gas Decay tanks).

The equation is:

$$D^{TOT} = D^{G} + D^{I} + D^{V} + D^{L} + D^{M} + D^{F}$$

where:

D^{TOT} = total dose;

D^{G} = dose contribution from ground plane deposition as calculated in Section 3.4.1.a;

D^{I} = dose contribution from inhalation of radioiodines, tritium, and particulates (> 8 days) as calculated in Section 3.4.1.b;

D^{V} = dose contributions from consumption of vegetation (defined as produce) for humans and stored feed for cattle. See Section 3.4.1.c for calculations;

D^{L} dose contributions from consumption of fresh leafy vegetables (defined as garden products) for humans and pasture grass for cattle. See Section 3.4.1.c for calculations;

D^{M} = dose contribution from consumption of cow's milk; and

Note: Consumption by the cow of both stored feeds and pasture grasses is taken into account when calculating this dose contribution. Concentration factors for both food sources are calculated.

D^{F} = dose contribution from consumption of meat.

Note: Consumption by the cow of both stored feeds and pasture grasses is taken into account when calculating this dose contribution. Concentration factors for both types of animal are calculated.

3.4.1 Total Dose from Atmospherically Released Radionuclide

After the calculation of the concentration factors from the applicable parts of Section 3.4.1, the maximum individual dose as calculated for controlling age group “a” and controlling organ "j", in sector θ at the controlling distance "r" is given from:

$D^G(r,\theta,j,a)$		(Section 3.4.1.a)	for ground plane deposition
$D^I(r,\theta,j,a)$		(Section 3.4.1.b)	for inhalation
$D^V(r,\theta,j,a)$	=	$\sum_i DFI_{ija} U_a^V C_i^V(r,\theta)$	for produce
$D^L(r,\theta,j,a)$	=	$\sum_i DFI_{ija} U_a^L C_i^L(r,\theta)$	for leafy vegetables
$D^M(r,\theta,j,a)$	=	$\sum_i DFI_{ija} U_a^M C_i^M(r,\theta)$	for cow's milk
$D^F(r,\theta,j,a)$	=	$\sum_i DFI_{ija} U_a^F C_i^F(r,\theta)$	for meat

where:

a = controlling age group (infant, child, teen, or adult);

j = controlling organ (bone, liver, total body, thyroid, kidney, lung, or GI-LLI);

r = user-selected distance from the release point to the receptor location in a particular sector, in kilometers (the controlling distance is the same for all airborne pathways, 1.05 km);

θ = one of sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc. (WNW);

DFI_{ija} = dose conversion factor for ingestion of nuclide "I", organ "j", and age group "a", in mrem/ρCi;

Note: Values used in these tables are taken from Tables E-11 through E-14 of RG 1.109. DFI_{ija} is selected according to the controlling organ and age group as specified in the database.

U_a^V, U_a^L, U_a^M, U_a^F = ingestion rates for produce, leafy vegetables, cow's milk, and meat, respectively, for individuals in age group "a". Values used are taken from Table E-5 of RG 1.109.);

C_i^V, D_i^L, C_i^M D_i^F = concentration of nuclide "I" for produce, leafy vegetables, cow's milk, and meat, respectively, in ρCi/kg or ρCi/liter.

The program calculates that maximum individual dose for each sector surrounding the plant in which the user has specified a controlling distance for each of the following pathways: A) ground plane deposition; B) inhalation and the ingestion of; C) produce; D) leafy vegetables; E) cow's milk; and F) meat. Only the receptor point receiving the maximum dose value is printed.

3.4.1.a Dose from Ground Plane Deposition

The dose D^G from direct exposure to activity deposited on the ground plane is calculated as follows (see RG 1.109-24, equations C-1 and C-2):

$$D^G(R,\theta,j,a) = \{(S_F)(1.0 \times 10^{12})(\Sigma_i[(\lambda_i^{-1})(1 - e^{-\lambda_i t_b})]\}(DOQ(r,\theta))(Q_i)(DFG_{ij})$$

where:

r	=	user-selected distance from the release point to the receptor location in a particular sector, in kilometers. The controlling distance is the same for all airborne pathways (1.05 km);
θ	=	one of sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc. (WNW);
a	=	user-selected age group (infant, child, teen, adult) which is the same controlling age group used for all airborne pathways (child);
j	=	user-selected organ (bone, liver, total body, thyroid, kidney, lung, GI-LLI) which is the same controlling organ used for all airborne pathways;
{ }	=	represents the concentration factor stored in the database;
S_F	=	dimensionless attenuation factor accounting for the dose reduction due to shielding by residential structures. The value is 0.7 (for maximum individual);
1.0×10^{12}	=	number of ρCi per Ci;
λ_i	=	decay constant of nuclide "I" in hr^{-1};
t_b	=	length of time over which the accumulation is evaluated (nominally 15 years which is the approximate midpoint of facility operating life or 1.31×10^5 hours);
$DOQ(r,\theta)$	=	average relative deposition of the effluent at the receptor location "r" in sector "θ", considering depletion of the plume during transport, in m^2 (1.7E-8/m^2);
Qi	=	release of nuclide "I" in curies, and
DFG_{ij}	=	open field ground plane dose conversion factor for organ "j" (total body or skin) from radionuclide "I", in mrem-m^2/ρCi-hr. The dose factor is selected according to the user-specified controlling age group "a" and controlling organ "j".

3.4.1.b Dose from Inhalation of Radionuclides in Air

The dose DI to organ "j" of age group "a" associated via inhalation of radioiodines and particulates is (see RG 1.109-25, Equations C-3 and C-4):

$$D^I(r,\theta,j,a) = (3.17 \times 10^4)(R_a)(\Sigma_i[(Q_i)(D2DPX/Q(r,\theta))(DFA_{ija})]$$

where:

r	=	user-selected distance from the release point to the receptor location in a particular sector, in kilometers. The controlling distance is the same for all airborne pathways (1.05 km);
θ	=	one of sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc. (WNW);

j = user-selected organ (bone, liver, total body, thyroid, kidney, lung, GI-LLI) and is the same controlling organ as that used for all airborne pathways;

a = user-selected age group (infant, child, teen, adult) and is the same controlling age group as that used for all airborne pathways;

3.17×10^4 = number of ρCi/Ci divided by the number of seconds/year;

R_a = annual air intake for individuals in age group "a" (in m^3/year). The air intake factor is selected in accordance with the user-specified controlling age group;

Q_i = release of nuclide "I" in curies;

$D2DPX/Q(r,\theta)$ = annual average atmospheric dispersion factor of the radionuclide at the receptor location "r" in sector "θ" (in sec/m^3) as calculated; and

Note: This includes depletion (for radioiodines and particulates) and radioactive decay of the plume.

DFA_{ija} = inhalation dose factor for radionuclide "I", organ "j", and age group "a". The inhalation dose factor is selected in accordance with the user-specified controlling age group "a" and controlling organ "j".

3.4.1.c Dose from Nuclide Concentrations in Vegetation

Note: To reduce the computational overhead of the computer, the calculations for dose resulting from nuclide concentrations in forage, produce and leafy vegetables is performed in three steps.

First, the concentration factors (CF) are computed and stored in the database. The concentration factor includes all the parameters that are considered constant for each nuclide and agricultural activity, such as the radioactive decay constant, removal rate constant, exposure time, etc.

Second, the deposition rate from the plume is multiplied by the concentration factor and the nuclide activity to produce the nuclide concentration as follows:

$$C_i^V(r,\theta) = (CF_i)(DOQ(r,\theta))(Q_i)$$

where:

$C_i^V(r,\theta)$ = concentration of nuclide "I" at the receptor location (r,θ);

CF_i = concentration factor of nuclide "I";

$DOQ(r,\theta)$ = relative deposition of nuclide "I". For the short term dispersion option, DOQ is replaced by (F x DOQ), where F is the short term dispersion correction factor;

Q_i = quantity of nuclide "I" released in curies.

For carbon-14 and tritium, the nuclide concentration is calculated from the concentration factor times the decayed and depleted X/Q for radioiodines and particulates (D2DPX/Q), times the quantity of nuclide "I" released in curies. For the short term dispersion option, D2DPX/Q is replaced by F x D2DPX/Q, where F is the short term dispersion correction factor.

V

$$C_T(r,\theta) = (CF_T)(D2DPX/Q(r,\theta))(Q_T) \text{ for tritium, and}$$

$$CF^V_{14}(r,\theta) = (CF_{14})(D2DPX/Q(r,\theta))(Q_{14}) \text{ for carbon-14}$$

Third, the nuclide concentrations for a particular pathway (produce, leafy vegetables, cow's milk, and meat) are summed and multiplied by: 1) the ingestion rate for a particular age group and 2) the dose conversion factor:

$$D(r,\theta,j,a) = \Sigma_i \, [(DFI_{ija})(U_a)(C^V_i(r,\theta))]$$

where:

- r = user-selected distance from the release point to the receptor location in a particular sector, in kilometers (1.05 km);
- θ = one of sixteen 22.5° sectors surrounding the reactor site, designated N, NNE, NE, etc. (WNW);
- j = user-selected organ (bone, liver, total body, thyroid, kidney, lung, GI-LLI), and is the same controlling organ as that used for all airborne pathways;
- a = user-selected age group (infant, child, teen, adult), and is the same controlling age group as that used for all airborne pathways;
- DFI_{ija} = dose conversion factor for ingestion of nuclide "I", organ "j", and age group "a", in mrem/ρCi, according to the controlling organ and age group;
- Ua = annual ingestion rate of food in a particular pathway (kilograms/year or liters/year) for individuals in age group "a", according to the controlling age group; and
- $C^V_i(r,\theta)$ = concentration of nuclide "I" at the receptor location (r,θ).

3.4.1.c.1 Calculating Vegetation Concentration Factors

NUREG-0133 calculations for radioiodines and particulate radionuclides (except tritium and carbon-14), the concentration factor of nuclide "I" in and on vegetation is estimated as follows:

$$CF^V_i = (CONST)\left(\frac{r}{(Y_v)(\lambda_i)}\right)(e^{-\lambda_i t_h})(f)$$

where:

- CF^V_i = concentration factor of radionuclide "I" in vegetation (forage, produce, or leafy vegetables), in m^2-hr/kg;
- CONST = 1.14 x 10^8 number of ρCi per Ci (10^{12}) divided by the number of hours per year (8760);
- r = is the fraction of deposited activity retained on crops, leafy vegetables, or pasture grass, from airborne radioiodine and particulate deposition:
  - r = 1.00 for radioiodines
  - r = 0.20 for particulates
- Y_v = agricultural productivity (yield or vegetation area density), in kg (wet weight)/m^2:
  - Y_s = 2.0 kg/m^2 for stored animal feed for grass-animal-man pathways
  - Y_ρ = 0.7 kg/m^2 for pasture grass for grass-animal-man pathways

Y_1 = 2.0 kg/m^2 for leafy vegetation (fresh) for crop/vegetation-man pathways

Y_g = 2.0 kg/m^2 for garden produce (stored vegetables) for crop/vegetation-man pathways

λ_i = is the decay constant of nuclide "I" in hr^{-1};

t_h = is a holdup time that represents the time interval between harvest and consumption of the food, in hours:

t_h = 0 hours for pasture grass consumed by animals

t_h = 2160 hours for stored feed consumed by animals

t_h = 24 hours for leafy vegetables consumed by humans

t_h = 1440 hours for produce consumed by humans

f = is the fraction of leafy vegetables or produce grown in garden of interest:

f = 0.76 for the fraction of produce ingested, grown in garden of interest (this is f_g in equation C-13 of RG 1.109)

f = 1.00 for the fraction of leafy vegetables grown in garden of interest (this is f_1 in equation C-13 of RG 1.109)

f = 1.00 for all other pathways

3.4.1.c.2 Concentration Factor for Carbon-14

For carbon-14, the concentration factor in and on vegetation is estimated as follows (see RG 1.109-26, equation C-8):

$$CF^V_{14} = (2.2 \times 10^7)(\rho)$$

where:

CF^V_{14} = concentration factor of carbon-14 in and on vegetation, in m^2-hr/kg; and

ρ = is defined as the ratio of total annual release time (for C-14 atmospheric releases) to the total annual time during which photosynthesis occurs (taken to be 4400 hours), under the condition that the value of "ρ" should never exceed unity. For continuous C-14 releases, "ρ" is taken to be unity (thus, the value of 2.2×10^7 is stored for CF^V_{14} in lieu of a site specific value for "ρ").

3.4.1.c.3 Concentration Factor for Tritium

The concentration factor for tritium in vegetation is calculated from the tritium concentration in air surrounding the vegetation (see RG 1.109-27, equation C-9):

$$CF^V_T = \frac{1.2 \times 10^7}{H}$$

where:

CF^V_T = concentration factor for tritium in vegetation (in m^2-hr/kg); and

H = absolute humidity at the location of the vegetation, in g/m^3 (the regulatory default value for "H" is 8.0 grams/m^3).

Thus, the value 1.5×10^6 is stored for CF^V_T in lieu of a site specific value for "H".

3.4.1.c.4 Nuclide Concentrations in Produce and Leafy Vegetables

The concentrations in and on produce and leafy vegetables of all radioiodine and particulate nulcides "I" (except carbon-14 and tritium) are calculated as follows:

$$C_i^V(r,\theta) = (CF_i^V)(DOQ(r,\theta))(Q_i) \text{ for produce; and}$$

$$C_i^L(r,\theta) = (CF_i^L)(DOQ(r,\theta))(Q_i) \text{ for leafy vegetables}$$

where:

CF_i^V = concentration factor of nuclide "I" in produce;

CF_i^L = concentration factor of nuclide "I" in leafy vegetables;

Note that the difference between CF_i^V and CF_i^L are the values for t_h and f_1.

$DOQ(r,\theta)$ = relative deposition of the radionuclide "I" at the receptor (r,θ); and

Q_i = release of nuclide "I" (in curies).

The C-14 and H-3 nuclide concentrations are calculated from the concentration factors times the decayed and depleted radioiodine relative deposition D2DPX/Q times the fraction grown in the garden of interest (f_g = 0.76, f_1 = 1.0):

$$C_T^V(r,\theta) = (CF_T^V)(D2DPX/Q(r,\theta))(Q_T)(f_g)$$

$$C_T^L(r,\theta) = (CF_T^L)(D2DPX/Q(r,\theta))(Q_T)(f_1) \text{ for tritium}$$

$$C_{14}^V(r,\theta) = (CF_{14}^V)(D2DPX/Q(r,\theta))(Q_{14})(f_g)$$

$$C_{14}^L(r,\theta) = (CF_{14}^L)(D2DPX/Q(r,\theta))(Q_{14})(f_1) \text{ for carbon-14}$$

3.4.1.d Nuclide Concentration in Cow's Milk

The radionuclide concentration in cow's milk is dependent upon the quantity and contamination level of feed consumed by the animal. The concentration is estimated (see RG 1.109-27, equations C-10 and C-11) as follows:

$$C_i^m(r,\theta) = \{(F_m)(Q_F)(e^{-\lambda_i t_f})[(f_p)(f_s)(CF_i^v) + (1 - f_p)(CF_i^{v1}) + (f_p)(1 - f_s)(CF_i^{v1})]\}(D(r,\theta)(Q_i)$$

where:

$C_i^m(r,\theta)$ = is the concentration of nuclide "I" in cow's milk at the receptor location (r,θ), in ρCi/liter;

{ } = the expression in brackets represents the concentration factor (note that the concentration factor for cow's milk involves two different vegetation concentration factors (see below));

F_m = average fraction of the cow's daily intake of radionuclide "I" (which appears in each liter of milk), in days/liter;

Q_F = amount of feed consumed by the cow per day, in kg/day (wet weight);

λ_i	=	decay constant of nuclide "I" in hr^{-1};
t_f	=	average transport time of the activity from the feed into the milk and to the receptor (a value of 2 days is assumed);
fp	=	fraction of the year that cows graze on pasture;
fs	=	fraction of daily feed that is pasture grass when the cow grazes on pasture;
CF_i^v	=	vegetation concentration factor of nuclide "I" on pasture grass with the holdup time t_h = 0 days, in ρCi/kg (refer to the explanation of the vegetation concentration factor calculation);
CF_i^{v1}	=	vegetation concentration factor of nuclide "i" in stored feeds with the holdup time t_h = 90 days, in ρCi/kg (refer to the explanation of the vegetation concentration factor calculations);
D(r,θ)	=	relative deposition DOQ(r,θ) of the radionuclides, except carbon-14 and tritium. For carbon-14 and tritium, the decayed and depleted dispersion factor D2DPX/Q(r,θ) for radioiodines and particulates (in sec/m^3) is used; and
Q_i	=	is the release of nuclide "I" in curies.

3.4.1.e Nuclide Concentration in Meat

The radionuclide concentration in meat is dependent upon the quantity and contamination level of feed consumed by the animal. The concentration is estimated (see RG 1.109-27, equations C-11 and C-12) as follows:

$$C_i^f(r,\theta) = \{(F_f)(Q_F)(e^{-\lambda_i t_s})[(f_p)(f_s)(CF_i^v) + (1- f_p)(CF_i^{v1}) + (f_p)(1 - f_s)(CF_i^{v1})]\}(D(r,\theta)(Q_i)$$

where:

Note: All parameters used in this pathway are for beef cattle.

$C_i^f(r,\theta)$	=	concentration of nuclide "I" in animal flesh at the receptor location (r,θ) in ρCi/liter;
{ }	=	the expression in brackets represents the concentration factor (note that the concentration factor for meat involves two different vegetation concentration factors);
F_f	=	average fraction of the animal's daily intake of radionuclide "I" which appears in each kilogram of flesh (in days/kg);
Q_f	=	amount of feed consumed by the animal per day in kg/day (wet weight);
λ_i	=	decay constant of nuclide "I" in hr^{-1};
t_s	=	average time from slaughter of the animal to consumption by humans (20 days);
f_p	=	fraction of the year that animals graze on pasture;
f_s	=	fraction of daily feed that is pasture grass when the animal grazes on pasture;
CF_i^{v1}	=	vegetation concentration factor of nuclide "I" on pasture grass with the holdup time t_h = 0 days in ρCi/kg (refer to the explanation of the vegetation concentration factor calculation);
CF_i^{v1}	=	vegetation concentration factor of nuclide "I" in stored feeds with the holdup time t_h = 90 days, in ρCi/kg (refer to the explanation of the vegetation concentration factor calculation);

$D(r,\theta)$ = relative deposition $DOQ(r,\theta)$ of the radionuclides, except carbon-14 and tritium. For carbon-14 and tritium, the decayed and depleted dispersion factor $D2DPX/Q(r,\theta)$ for radioiodines and particulates (in sec/m^3) is used;

Q_i = is the release of nuclide "I" (in curies).

3.5 Gaseous Effluent Projected Dose Calculation

3.5.1 The quarterly projected dose is based upon the methodology of Sections 3.3 and 3.4, and is expressed as follows:

$$D_{QP} = \left(\frac{D_{QC} + D_{RP}}{T}\right)(92)$$

where:

D_{QP} = Quarterly projected dose (mrem);

D_{QC} = cumulative dose for the quarter (mrem);

D_{RP} = dose for current release (mrem);

T = current days into quarter; and

92 = number of days per quarter.

3.6 Dose to the Public Inside the Site Boundary

3.6.1 Liquid Releases

Dose to the public inside the site boundary due to liquid releases will be due to ingestion of fish caught from the discharge canal and exposure to sediment along the discharge canal bank while fishing.

3.6.1.a Dose Due to Ingestion of Fish

Dose due to ingestion of fish is calculated using the methodology given in Section 2.2, Liquid Dose Calculation.

3.6.1.b Dose Due to Exposure to Shoreline Sediments

Dose from external exposure to shoreline sediments is calculated from equation A-7 of RG 1.109, Rev. 1, 10/77.

$$R_{apj} = 110{,}000\left(\frac{(U_{ap})(M_p)(W)}{F}\right)(\Sigma_i\ [(Q_i)(T_i)(D_{aipj})(e^{-\lambda_i t_p})(1-e^{-\lambda_i t_b})]$$

where:

R_{apj} = is the total annual dose to organ "j" of individuals of age group "a" from all of the nuclides "I" in pathway in mrem/yr;

U_{ap} = is the usage factor that specifies exposure time for the maximum individual of age group "a" in hours from Table E-5 of RG 1.109. Sixty-seven hours for shoreline recreation for a teen was chosen. Adult is the controlling age group for ingestion but the maximum usage factor (teen) was used rather than the adult factor to ensure a conservative dose estimate;

M_p = is the mixing ratio (reciprocal of dilution factor);

W = is the shoreline width factor from Table A-2 of RG 1.109. The discharge canal value of 0.1 was chosen;

F = is the flow rate of the liquid effluent in ft^3/sec. This was determined by:

$$F(ft^3/sec) = \text{waste volume (gal/yr)} * \frac{.134\ ft^3}{1\ gal} * \frac{1\ yr}{8760\ hr} * \frac{1\ hr}{3600\ sec}$$

where:

Q_i = is the release of nuclide "I" in Ci/yr;

T_i = is the radioactive half-life of nuclide "I", in days, from Radioactive Decay Data Tables, Technical Information Center, U. S. Dept. of Energy, 1981;

D_{aipj} = is the dose factor specific to age group "a", nuclide "I", and organ "j" from Table E-6 of RG 1.109;

λ_i = is the radioactive decay constant of nuclide "I" in hr^{-1};

t_p = is the average transit time for nuclides to reach the point of exposure. A value of 0 hours was chosen due to the proximity of the discharge canal to the plant; and

t_b = is the period of time for which sediment is exposed to the contaminated water in hours. The mid-point of plant operating life, 15 years was chosen per RG 1.109.

3.6.2 Airborne Release

3.6.2.a Dose Due to Noble Gases

Dose to fisherman at the discharge canal can be calculated by the ratio of dispersion factor for the discharge canal (1.6E-4 sec/m^3 from Table 2-45 SAR, Unit 1, 100 meters downwind in a southerly direction) and the usage factor of 67 hours of shoreline recreation to the values used in Section 3.3 of this manual.

$$\text{Dose at discharge canal} = D^T(\theta) * \frac{1.6E\text{-}4}{2.0E\text{-}5} * \frac{67\ hr}{8760\ hr}$$

where $D^T(\theta)$ is the noble gas dose calculated by Section 3.3.

3.6.2.b Dose Due to Iodine, Tritium and Particulates from Gaseous Effluents

Section 3.4 calculates total dose for iodine, tritium and particulates as the sum of:

$$D^{TOT} = D^G + D^I + D^V + D^L + D^M + D^F$$

where:

D^G = ground plane deposition;

D^I = inhalation;

D^v = consumption of vegetation;

D^L = consumption of fresh leafy vegetables;

D^m = consumption of milk; and

D^F = consumption of meat and poultry

The only contributions relevant to fishing activities at the discharge canal are ground plane deposition and inhalation. As D^G and D^I are not independently available, a conservative estimate can be obtained by using the same correction factor developed for noble gas dose to the total dose calculated in Section 3.4 for iodine, tritium and particulates. Depletion of the plume as it travels downwind can be ignored since the fraction remaining in the plume at 100 meters (discharge canal) and 1046 meters (site boundary) are both greater than 90% according to Figure 3 of RG 1.111.

The only activity inside the plant site by members of the public that might contribute a significant dose is fishing along the banks of the discharge canal. Travel along public roads would involve short exposure time and tours of the facility are conducted according to radiological control procedures enforced at the plant to control exposure. Fishing is the only uncontrolled activity.

4.0 ENVIRONMENTAL SAMPLING STATIONS - RADIOLOGICAL

Section 1.0 of the ODCM provides reference to the Radioactivty Effluent Controls Program governed by ANO-1 TS 5.5.4 and ANO-2 TS 6.5.4. However, a Radiological Environmental Monitoring Program is also necessary to meet the intent of the purpose of the ODCM.

The Radiological Environmental Monitoring Program is established to provide radiation and radionuclide monitoring in the environs surrounding the site. The program provides a method for representative measurements of radioactivity in the highest potential exposure pathways. In addition, the program provides for verification of the accuracy of the effluent monitoring program and modeling of envronmental exposure pathways.

The Radiological Environmental Monitoring Program is established by the ODCM and conforms to the guidance contained in 10 CFR 50, Appendix I. The program also provides for:

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters of the ODCM,

2. A land use census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made, if required by the results of the census, and

3. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

Environmental samples are collected as specified in the Limitations. The approximate locations of selected sample sites are shown on Figures 4-1, 4-1A, and 4-1B for illustrative purposes.

Table 4-1 lists the approximate distances and directions of the sample stations from the plant.

5.0 REPORTING REQUIREMENTS

5.1 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report is submitted by May 15 of each year and contains a summary of the Radiological Environmental Monitoring Program for the reporting period. This report meets the requirements of TS 5.6.2 (ANO-1) and TS 6.6.2 (ANO-2), and is consistent with the objectives outlined in the ODCM and 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C. The report is formatted consistent with RG 1.21, Revision 1, to the extent possible. A single submittal is normally prepared incorporating the data for both ANO units (common information is combined).

The Annual Radiological Environmental Operating Report includes the following:

1. Summarized and tabulated results of all radiological environmental samples and environmental radiation measurements required by the ODCM.

2. A summary description of the Radiological Environmental Monitoring Program.

3. A map of the sampling locations with concurrent table providing distances and directions from the Reactor (Containment) Building. Because the ODCM contains this information and the ODCM is submitted as part of the Radioactive Effluent Release Report, reference to the Radioactive Effluent Release Report submittal date and letter number may be included in the Annual Radiological Environmental Operating Report in lieu of submitting the sample location map and table.

4. A summary of the land use census results in accordance with Surveillance S 2.5.2.2.

5. A summary of the Interlaboratory Comparison Program in accordance with, Surveillance S 2.5.3.1.

As required by the Limitations, the report shall include the following for the conditions listed below:

1. A description of the condition or event and, if applicable, equipment involved.

2. The cause of the condition or event.

3. Actions taken to restore the condition and prevent/minimize recurrence.

4. The consequences of the condition or event.

Limitation	Required Action	Description
2.5.1	A.2	• Sample not taken at required location* • Sample equipment out-of-service (OOS) • Sample frequency not met • Monitoring/analysis lower limit of detection (LLD) not met • Concentration limits not met • Dose from other radionuclides exceed concentration limits
2.5.1	B.1	New sample location identified
2.5.2	A.1	New sample location identified
2.5.3	A.1	Interlaboratory Comparison Program requirements not met
NA	NA	Other harmful effects or evidence of irreversible damage detected

* The report shall include a summary of information not available for reporting at the time of submittal. Such missing information shall be submitted in a supplemental letter when data becomes available.

5.2 <u>Radioactive Effluent Release Report</u>

The Radioactive Effluent Release Report is submitted prior to May 1 of each year, but not more than 12 months from the previous year's submittal, and includes a summary of the quantities of radioactive liquid effluents, gaseous effluents, and solid waste released from the site. This report meets the requirements of TS 5.6.3 (ANO-1), TS 6.6.3 (ANO-2), 10 CFR 50.36a, and 10 CFR 50, Appendix I, Section IV.B.1. The report is formatted consistent with RG 1.21, Revision 1. A single submittal is normally prepared incorporating the data from both ANO units (common information is combined).

In general, the Radioactive Effluent Release Report includes the following:

1. A description of changes to the ODCM and PCP implemented during the reporting period. TS 5.6.3 (ANO-1) and TS 6.6.3 (ANO-2) contain a description of the ODCM change process.

2. A summary of the hourly meteorological data collected over the previous calendar year. In lieu of including this information in the report, it is permissible to retain this summary available for NRC review, if so noted in the report.

3. A summary of radiation doses due to radiological effluents during the previous calendar year, calculated in accordance with the methodology specified in the ODCM.

4. The radiation dose to members of the public while performing activities inside the site boundary. The calculated dose includes only contributions directly attributed to operation of the units.

5. A description of major changes to the radioactive waste systems (liquid/gaseous/solid) during the previous calendar year, if not included in the cycle SAR update.

As required by the Limitations, the report shall include the following for the conditions listed below:

1. A description of the condition or event and, if applicable, equipment involved.
2. The cause of the condition or event.
3. Actions taken to restore the condition and prevent/minimize recurrence.
4. The consequences of the condition or event.

Limitation	Required Action	Description
2.1.1	D.1	Liquid radioactive monitoring equipment OOS > 30 days
2.2.1	G.1	Gaseous radioactive monitoring equipment OOS > 30 days
2.3.1	A.2	Liquid radioactive release limits exceeded
2.3.1	F.1	Liquid radioactive monitor LLD exceeded
2.4.1	A.2	Gaseous radioactive release limits exceeded
2.4.1	E.1	Gaseous radioactive monitor LLD exceeded

FIGURE 4-1

RADIOLOGICAL SAMPLE STATIONS

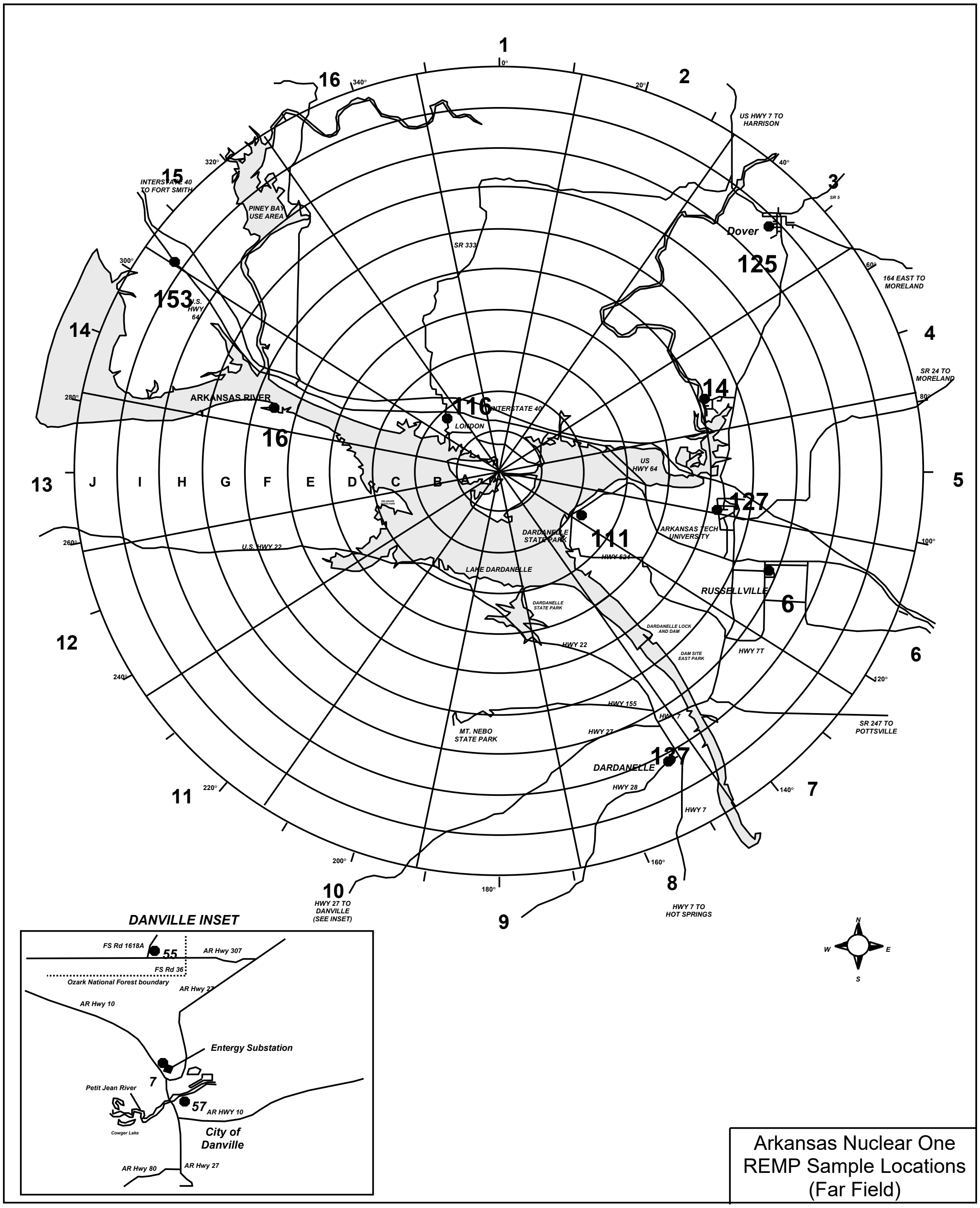

FIGURE 4-1A

RADIOLOGICAL SAMPLE STATIONS

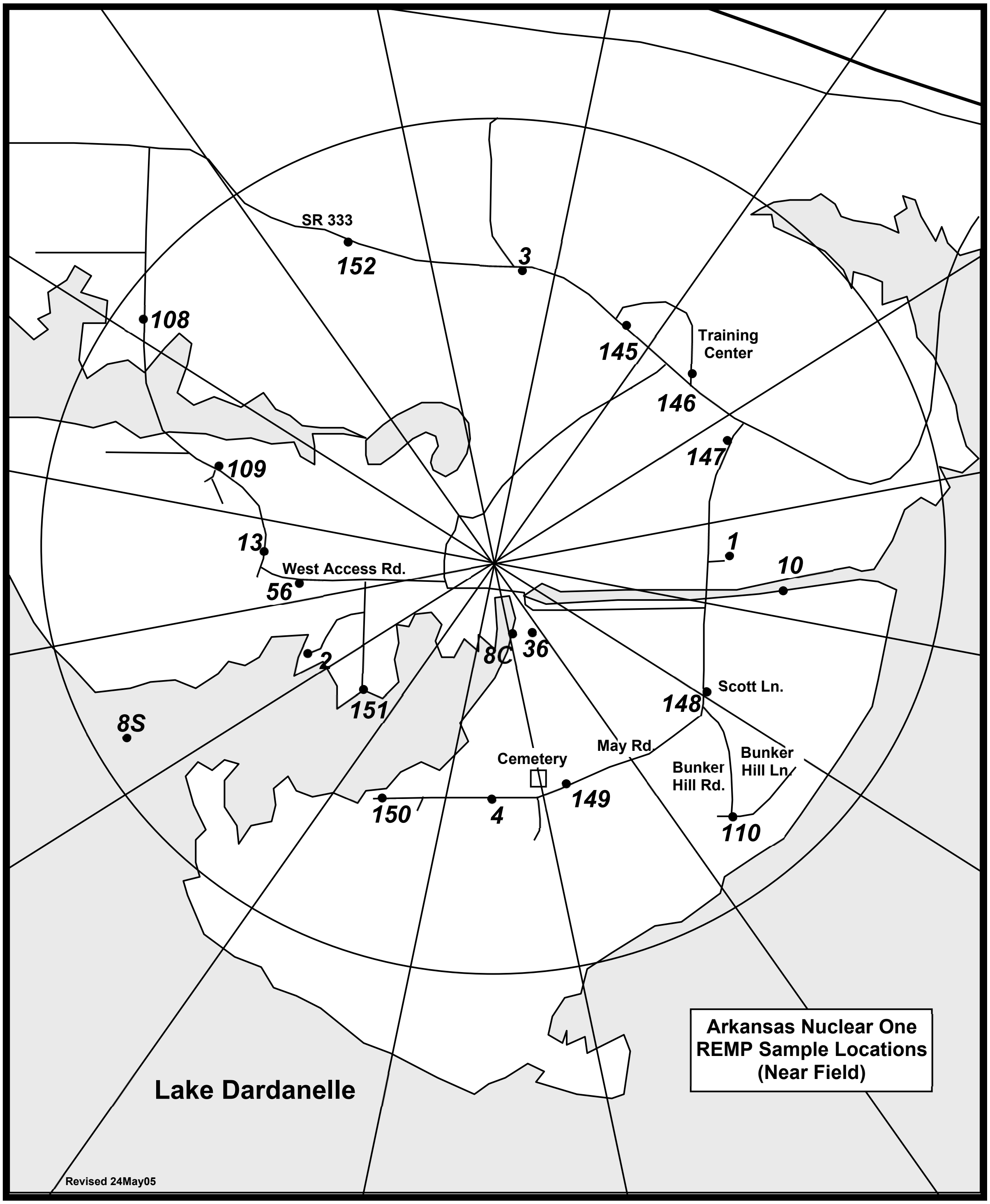

FIGURE 4-1B

RADIOLOGICAL SAMPLE STATIONS

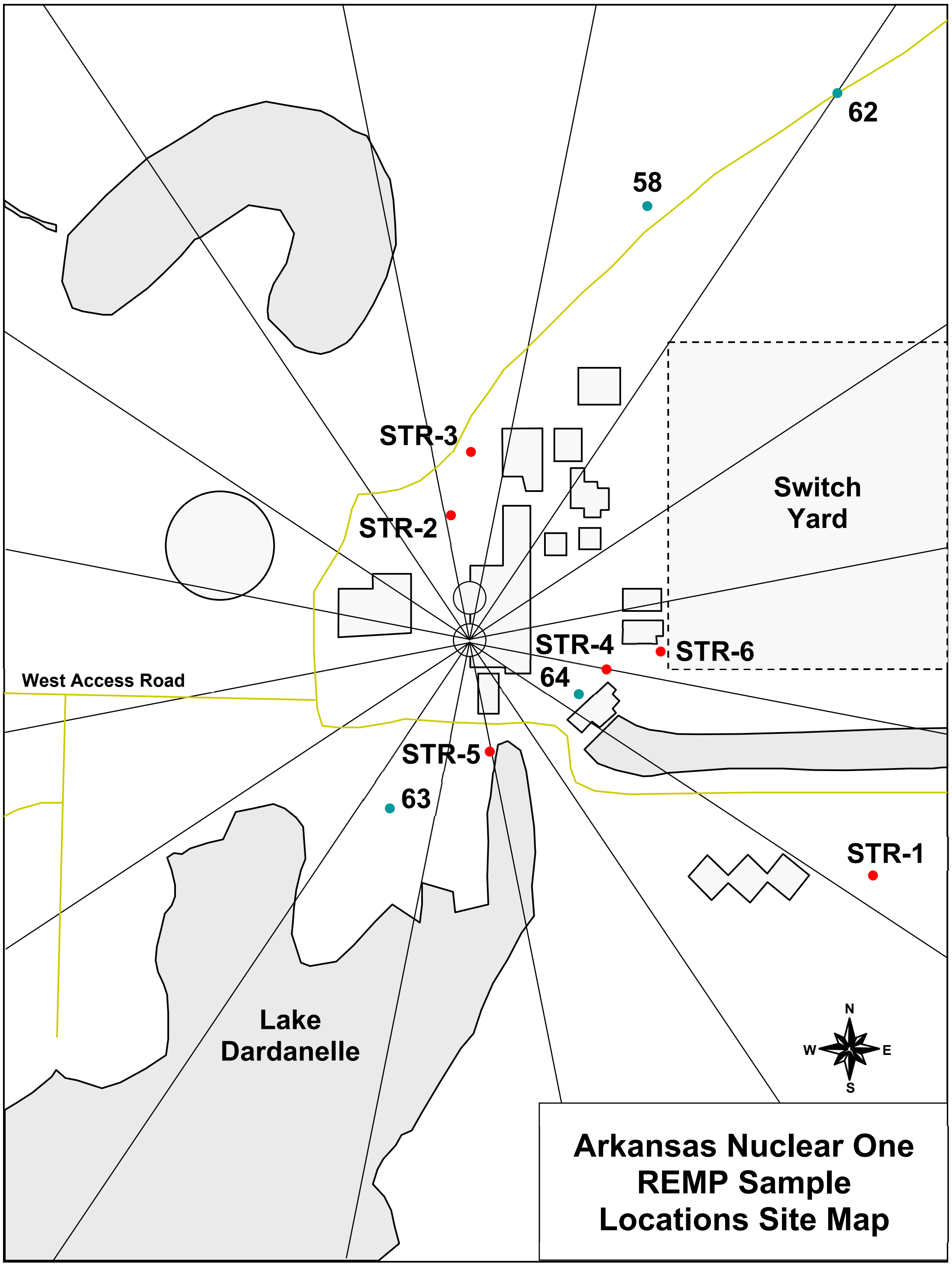

FIGURE 4-2

MAXIMUM AREA BOUNDARY FOR RADIOACTIVE RELEASE CALCULATION
(Exclusion Areas)

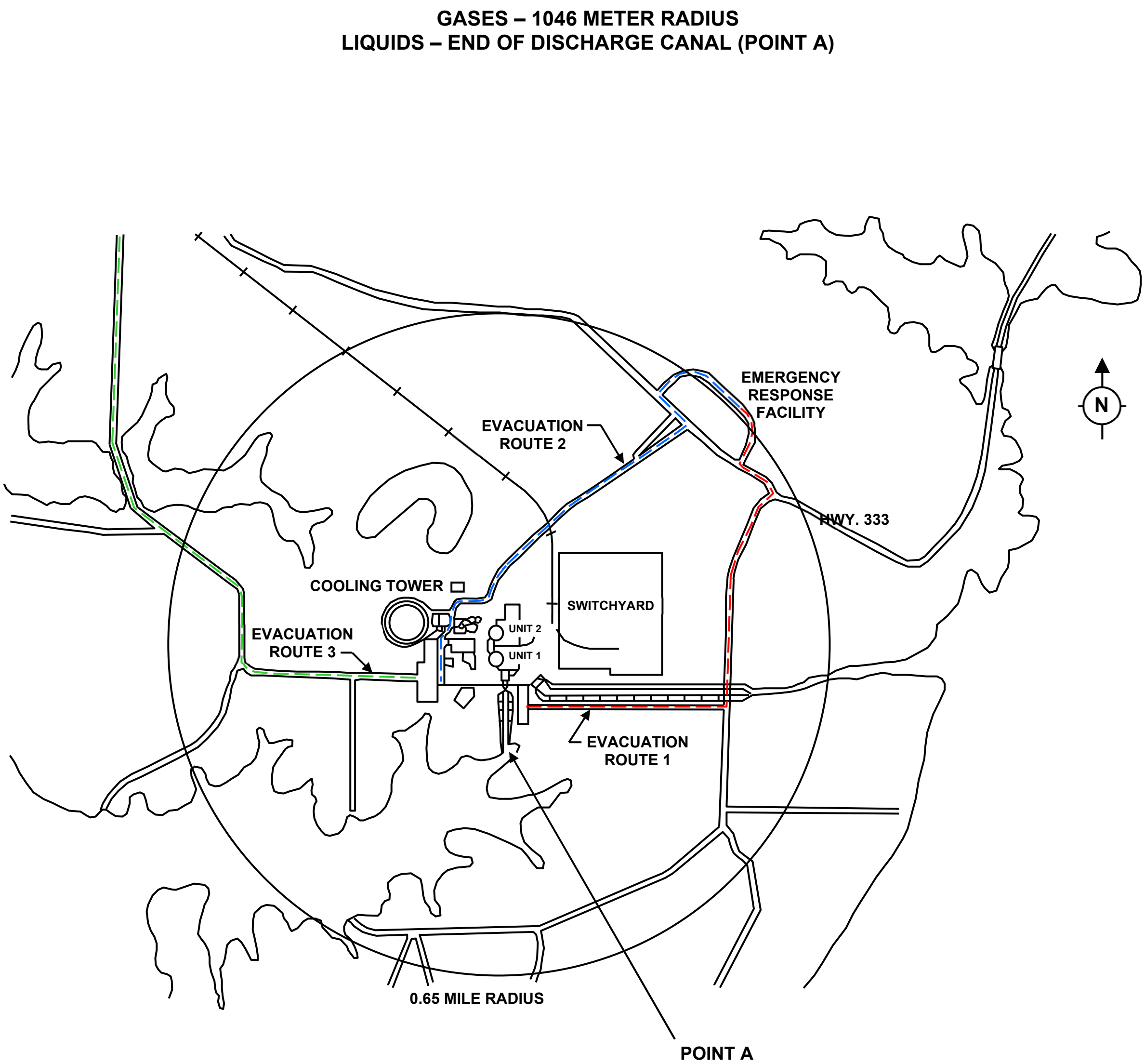

TABLE 4-1

Environmental Sampling Stations - Radiological

Sample Station #	Approximate Direction and Distance from Plant	Sample Types	Sample Station Location
1	88° - 0.5 miles	Airborne radioiodines Airborne particulates Direct radiation	The thermoluminescent dosimeter (TLD) is on a pole near the meteorology tower approx. 0.6 miles east of ANO.
2	243° - 0.5 miles	Airborne radioiodines Airborne particulates Direct radiation	Traveling from ANO, go approx. 0.2 miles west toward Gate 4. Turn left (at the east end of the sewage treatment plant) and go approx. 0.1 miles. Turn right and go approx. 0.1 miles. The sample station is on the right.
3	5° - 0.7 miles	Direct radiation	If traveling west on Highway (Hwy) 333, go approx. 0.35 miles from Gate 2 at ANO. TLD is located on utility pole on south side of Hwy 333 S. If traveling east on Highway 333, go approx. 0.9 miles from junction of Hwy 333 and Flatwood Road. TLD is located on utility pole on south side of Hwy 333 S.
4	181° - 0.5 miles	Direct radiation	Go approx. 0.25 miles south from bridge over intake canal. Turn right onto May Road. Proceed approx. 0.1 miles west of May Cemetery entrance. The TLD is located on a utility pole on the south side of May Road.
6	111° - 6.8 miles	Airborne radioiodines Airborne particulates Direct radiation	Go to the Entergy local office which is located off Hwy 7T in Russellville, Arkansas (AR) (305 South Knoxville Avenue). The sample station is against the east wall of the back lot.
7	210° - 19.0 miles	Airborne radioiodines Airborne particulates Direct radiation	Turn west at junction of Hwy 7 and Hwy 27 in Dardanelle, AR. Proceed to junction of Hwy 27 and Hwy 10 in Danville, AR. Turn right onto Hwy 10 and proceed a short distance to the Entergy supply yard, which is on the right adjacent to an Entergy substation. The sample station is in the southwest corner of the supply yard.
8	166° - 0.2 miles 243° - 0.9 miles 212° - 0.5 miles	Surface water (composite) Shoreline sediment Fish	Plant discharge canal

TABLE 4-1

Environmental Sampling Stations – Radiological (continued)

Sample Station #	Approximate Direction and Distance from Plant	Sample Types	Sample Station Location
10	95° - 0.5 miles (intake canal)	Surface water (grab)	Surface water (grab) is collected at plant intake canal.
13	273° - 0.5 miles	Broad leaf vegetation	Traveling from Hwy 333, turn south onto Flatwood Road. Go approx. 1.0 miles. The sample may be collected from either side of Flatwood Road.
14	70° - 5.1 miles	Drinking water	From junction of Hwy 7 and Water Works Road, go approx. 0.8 miles west on Water Works Road. The sample station is on the left at the intake to the Russellville city water system from the Illinois Bayou.
16	287° - 5.5 miles	Shoreline sediment Fish	Panther Bay, located on the south side of the AR River across from the mouth of Piney Creek.
36	153° - 0.02 miles	Pond water Pond sediment	The sample station is at the Wastewater Holding Pond on the ANO site east of the discharge canal.
55	217° - 13.1 miles	Broad leaf vegetation	Travel south on Hwy 27 and west on Hwy 307 to the western edge of the Ozark National Forest. Hwy 307 becomes Forest Service (FS) Rd 36; proceed ~ 3/4 mile on FS Rd 36 to its intersection with FS Rd 1618A. The sample station is located at this intersection.
56	264° - 0.4 miles	Airborne radioiodines Airborne particulates Direct radiation	Traveling west from ANO, the sample station is located at the west end of the sewage treatment plant near the facility blower building.
57	208° - 19.5 miles	Drinking water	Go to Danville and turn left on Fifth Street. Go approx. three blocks. The Danville public water supply treatment facility is located on the left.
58	22° - 0.3 miles	Groundwater	GWM – 1; North of Protected Area on owner controlled area (OCA), west of north Security Check Point, east side of access road.
62	34° - 0.5 miles	Groundwater	GWM – 101; North of Protected Area on OCA, east of outside receiving building.

TABLE 4-1

Environmental Sampling Stations – Radiological (continued)

Sample Station #	Approximate Direction and Distance from Plant	Sample Types	Sample Station Location
63	206° - 0.1 miles	Groundwater	GWM – 103; South of Protected Area on OCA, northeast of Stator Rewind Building near woodline.
64	112° - 0.1 miles	Groundwater	GWM – 13; South of Oily Water Separator, northwest corner of ANO-2 Intake Structure, inside the Protected Area.
108	306° - 0.9 miles	Direct radiation	If traveling from Hwy 333, turn south onto Flatwood Road and go approx. 0.4 miles. The TLD is on a utility pole on the right. If traveling north on Flatwood Road, go approx. 0.4 miles from sample station 109. The TLD is on a utility pole on the left.
109	291° - 0.6 miles	Direct radiation	Traveling from Hwy 333, turn south onto Flatwood Road. Go approx. 0.8 miles. The TLD is on a utility pole on the left across from the junction of Flatwood Road and Round Mountain Road.
110	138° - 0.8 miles	Direct radiation	From bridge over intake canal, go south approx. 0.25 miles. Turn left and go approx. 0.25 miles. Turn right on Bunker Hill Lane. The TLD is on the first utility pole on the left.
111	120° - 2.0 miles	Direct radiation	From junction of Hwy 64 and Hwy 326 (Marina Road), go approx. 2.1 miles on Marina Road. The TLD is on a utility pole on the left just prior to curve.
116	318° - 1.8 miles	Direct radiation	Go one block south of the west junction of Hwy 333 and Hwy 64 in London, AR. The TLD is on a utility pole north of the railroad tracks.
125	46° - 8.7 miles	Direct radiation	Traveling north on Hwy 7, turn left onto Water Street in Dover, AR. Go one block and turn left onto South Elizabeth Street. Go one block and turn right onto College Street. The TLD is on a utility pole at the southeast corner of the red brick school building, which is located on top of hill.

TABLE 4-1

Environmental Sampling Stations – Radiological (continued)

Sample Station #	Approximate Direction and Distance from Plant	Sample Types	Sample Station Location
127	100° - 5.2 miles	Direct radiation	The TLD is located on Arkansas Tech Campus on N. Glenwood Street. If traveling south on Hwy 7 from I- 40, turn right on N. Glenwood. Follow N. Glenwood for approx. 0.6 miles. The TLD is located on a utility pole (with a No Parking sign on it) across from the northeast corner of Paine Hall.
137	151° - 8.2 miles	Direct radiation	At junction of Hwy 7 and Hwy 28 in Dardanelle, AR, go approx. 0.2 miles down Hwy 28. The TLD is on a utility pole on the left in front of the Moore R. Morris Arkansas National Guard Armory.
145	28° - 0.6 miles	Direct radiation	The TLD is located near the west entrance to the Reeves E. Ritchie Training Center (RERTC) on a utility pole on the north side of Hwy 333.
146	45° - 0.6 miles	Direct radiation	The TLD is located on the south end of the east parking lot at the RERTC. The TLD is located on a utility pole.
147	61° - 0.6 miles	Direct radiation	The TLD is located on the west side of Bunker Hill Road, approx. 100 yards from the intersection with Hwy 333.
148	122° - 0.6 miles	Direct radiation	Traveling east from ANO, turn right on Bunker Hill Road. Travel south for approx. 0.25 miles to the intersection with Scott Lane. The TLD is located on the county road sign post.
149	156° - 0.5 miles	Direct radiation	Traveling south on Bunker Hill Road, turn right on May Road. Travel approx. 0.4 miles. The TLD is located on a "Notice" sign on the north side of May Road.
150	205° - 0.6 miles	Direct radiation	Traveling south on Bunker Hill Road, turn right on May Road. Travel approx. 0.8 miles. The TLD is located just past the McCurley Place turn off on the north side of May Road on a utility pole.

TABLE 4-1

Environmental Sampling Stations – Radiological (continued)

Sample Station #	Approximate Direction and Distance from Plant	Sample Types	Sample Station Location
151	225° - 0.4 miles	Direct radiation	Traveling west from ANO, turn south on plant road along the east side of the sewage treatment plant. The TLD is located at the end of this road, near the lake on a metal post.
152	338° - 0.8 miles	Direct radiation	Traveling west on Hwy 333 from the RERTC, travel approx. 0.7 miles. The TLD is located on the south side of Hwy 333 on a utility pole.
153	304° - 9.2 miles	Direct radiation	Travel Hwy 64 west to Knoxville Elementary School. The TLD is located near the school entrance gate on a utility pole.
STR - 1	120° - 0.33 miles	Storm water runoff	East side of GSB drainage ditch near lift station.
STR – 2	351° - < 0.10 miles	Storm water runoff	Inside protected area near Sally Port from drainage ditch along fence.
STR – 3	0.2° - 0.13 miles	Storm water runoff	Outside Protected Area near Sally Port from drainage ditch along fence.
STR - 4	102° - 0.10 miles	Storm water runoff	East side of Oily Water Separator from storm drain.
STR - 5	170° - < 0.10 miles	Storm water runoff	West side of discharge canal from storm drain.
STR - 6	90° - < 0.10 miles	Storm water runoff	East side of chemistry chemical storage area storm drain.

APPENDIX 1

RADIOLOGICAL EFFLUENT CONTROLS

1.0 DEFINITIONS

--NOTE--
The defined terms of this section appear in capitalized type and are applicable throughout these Limitations and Bases.

Term	Definition
ACTION(S)	ACTIONS shall be that part of a Limitation that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
BATCH RELEASE	A BATCH RELEASE is the discharge of liquid or gaseous wastes of a discrete volume.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel FUNCTIONALITY and the CHANNEL TEST. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL TEST	A CHANNEL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify FUNCTIONALITY of all devices in the channel required for channel FUNCTIONALITY. The CHANNEL TEST may be performed by means of any series of sequential, overlapping, or total steps.
CONTINUOUS RELEASE	A CONTINUOUS RELEASE is the discharge of liquid waste of a non-discrete volume, e.g. from a volume of a system that has an input flow during the continuous release.
EXCLUSION AREA	The EXCLUSION AREA is that area surrounding ANO within a minimum radius of 0.65 miles of the Reactor (Containment) Buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

1.0 DEFINITIONS (continued)

Term	Definition
FUNCTIONAL-FUNCTIONALITY	A system, subsystem, train, component, or device shall be FUNCTIONAL or have FUNCTIONALITY when it is capable of performing its specified function(s), as set forth in the current license basis (CLB) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified function(s) are also capable of performing their related support function(s).
GASEOUS RADWASTE TREATMENT SYSTEM	A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting gases from radioactive systems and providing for decay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.
LIQUID RADWASTE TREATMENT SYSTEM	A LIQUID RADWASTE TREATMENT SYSTEM is a system designed and used for holdup, filtration, and/or demineralization of radioactive liquid effluents prior to their release to the environment.
MEMBER(S) OF THE PUBLIC	MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from the category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.
MODE(S)	Refer to Definitions section of ANO-1 and ANO-2 TSs.
PURGE – PURGING	PURGE or PURGING is the controlled process of discharging air or gas from a confinement to reduce the airborne radioactivity concentration in such a manner that replacement air or gas is required to purify the confinement.
SOURCE CHECK	A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.0 DEFINITIONS (continued)

Term	Definition
VENTILATION EXHAUST TREATMENT SYSTEM	A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEMS.
UNRESTRICTED AREA	An UNRESTRICTED AREA shall be any area beyond the EXCLUSION AREA boundary.

2.0 LIMITATION (L) APPLICABILITY

L 2.0.1 Limitations shall be met during the specified conditions in the Applicability, except as provided in L 2.0.2.

L 2.0.2 Upon discovery of a failure to meet a Limitation, the applicable ACTIONS of the associated Limitation shall be met, except as provided in L 3.0.5. If the Limitation is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the ACTIONS is not required, unless otherwise stated.

L 2.0.3 When a Limitation is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, immediately initiate a condition report to document the condition and determine any limitations for continued operation of the plant.

Exceptions to this Limitation are stated in the individual Limitations.

L 2.0.4 When a Limitation is not met, entry into a MODE or other specified condition in the Applicability shall only be made when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time.

L 2.0.5 Equipment removed from service or declared non-functional to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its FUNCTIONALITY or the FUNCTIONALITY of other equipment. This is an exception to L 2.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate FUNCTIONALITY.

2.0 SURVEILLANCE (S) APPLICABILITY

S 2.0.1 Surveillances shall be met during the specified conditions in the Applicability for individual Limitations, unless otherwise stated in the Surveillance. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limitation. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the Limitation except as provided in S 2.0.3. Surveillances are not required to be performed on non-functional equipment or variables outside specified limits.

S 2.0.2 The specified Frequency for each Surveillance is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If an Action completion time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

S 2.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the Limitation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the Limitation must immediately be declared not met, and the applicable ACTIONS must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limitation must immediately be declared not met, and the applicable ACTIONS must be entered.

S 2.0.4 Entry into a specified condition in the Applicability of a Limitation shall only be made when the Limitation's Surveillances have been met within their specified Frequency, except as provided by S 2.0.3. When a Limitation is not met due to Surveillances not having been met, entry into a specified condition in the Applicability shall only be made in accordance with L 2.0.4.

L 2.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

L 2.1.1 The following Radioactive Liquid Effluent Monitoring Instrumentation shall be FUNCTIONAL:

a. Liquid Radwaste Effluent Radiation Monitor with alarm/trip function

b. Liquid Radwaste Effluent Flow Monitor

c. One Main Steam Line Radiation Monitor per Steam Generator (ANO-1 only)

APPLICABILITY: Liquid Radwaste Effluent Monitor – during releases via the associated pathway

Main Steam Line Radiation Monitors – MODES 1, 2, 3, and 4

ACTIONS

---NOTE---
Separate Condition entry is allowed for each instrument.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Liquid Radwaste Effluent Radiation Monitor non-functional.	A.1 Suspend the release of radioactive effluents monitored by the affected channel.	Immediately
	<u>AND</u>	
	A.2.1 Restore the monitor to a FUNCTIONAL status.	Prior to release of radioactive effluents monitored by the affected channel
	<u>OR</u>	
	A.2.2.1 Analyze two independent samples of the associated tank contents.	Prior to release of radioactive effluents monitored by the affected channel
	<u>AND</u>	
	A.2.2.2 Computer input data verified by two qualified individuals.	Prior to release of radioactive effluents monitored by the affected channel
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2.3 Correct discharge valve lineup independently verified by two qualified individuals.	Prior to release of radioactive effluents monitored by the affected channel
B. Required Liquid Radwaste Effluent Flow Monitor non-functional.	B.1 Estimate flow. <u>OR</u> B.2 Suspend the release of radioactive effluents monitored by the affected channel.	4 hours Immediately
C. One or more required Main Steam Line Radiation Monitor non-functional.	C.1 Establish pre-planned alternate monitoring method of monitoring. <u>AND</u> C.2 Restore the affected Main Steam Line Radiation Monitor(s) to a FUNCTIONAL status.	72 hours 7 days
D. Required Action(s) and/or Completion Time(s) of Conditions A, B, and/or C not met.	D.1 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately
E. Required Radioactive Liquid Effluent Monitoring Instrument non-functional for > 30 days.	E.1 Initiate a condition report to document and track the condition for inclusion in the Radioactive Effluent Release Report pursuant to TS 5.6.3 (ANO-1) / TS 6.6.3 (ANO-2).	Immediately

SURVEILLANCES

SURVEILLANCE		FREQUENCY
S 2.1.1.1	Perform a CHANNEL CHECK of required instrumentation.	24 hours
S 2.1.1.2	----------------------------------NOTE---------------------------------- Not applicable to Liquid Radwaste Effluent Flow Monitor. -- Perform a CHANNEL TEST of the required instrumentation.	92 days
S 2.1.1.3	Perform a CHANNEL CALIBRATION on the required instrumentation.	18 months
S 2.1.1.4	----------------------------------NOTES---------------------------------- 1. SOURCE CHECK not required when background radioactivity is greater than the check source. 2. Not applicable to Liquid Radwaste Effluent Flow Monitor or Main Steam Line Radiation Monitors. -- Perform a SOURCE CHECK on the required instrumentation.	Within 8 hours prior to release of radioactive effluents monitored by the channel

L 2.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

L 2.2.1 The following Radioactive Gaseous Effluent Monitoring Instrumentation shall be FUNCTIONAL:

---NOTE---
Refer to ANO-2 Technical Specification (TS) 3.3.3.1 for ANO-2 Containment Building Purge System Process Monitor operability requirements and associated ACTIONS.
----

a. Waste Gas Holdup Systems

1. Gas Activity Process Monitor with alarm/trip function
2. Effluent Flow Process Monitor

b. Reactor (Containment) Building Purge and Ventilation, Auxiliary Building Ventilation, Spent Fuel Pool Area Ventilation, Emergency Penetration Room Ventilation, Low Level Radwaste Building Ventilation, and ANO-2 Auxiliary Building Extension Ventilation SPING Monitors

1. Noble Gas Activity Monitor
2. Iodine Sampler
3. Particulate Sampler
4. Effluent Flow Monitor
5. Sampler Flow Monitor

APPLICABILITY:

1. SPINGS 4 and 8 – when Emergency Penetration Room Ventilation is capable of auto-start
2. All Radioactive Gaseous Effluent Monitoring Instrumentation – during releases via the associated pathway

ACTIONS

--NOTE--
Separate Condition entry is allowed for each instrument.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Applicable to releases associated with Waste Gas Holdup Systems and PURGE of the ANO-1 Reactor Building. ----------------------------------- Required Waste Gas Holdup and/or Reactor Building Purge System Gas Activity Process and/or Noble Gas Activity Monitor non-functional.	A.1 Suspend the release of radioactive effluents monitored by the affected channel.	Immediately
	<u>AND</u>	
	A.2.1 Restore the monitor to a FUNCTIONAL status.	Prior to release of radioactive effluents monitored by the affected channel
	<u>OR</u>	
	A.2.2.1 Analyze two independent samples of the Waste Gas Holdup Tank and/or Reactor Building contents.	Prior to release of radioactive effluents monitored by the affected channel
	<u>AND</u>	
	A.2.2.2 Computer input data verified by two qualified individuals.	Prior to release of radioactive effluents monitored by the affected channel
	<u>AND</u>	
	A.2.2.3 -------------NOTE------------- Not applicable to Reactor Building Purge System. ---------------------------------- Correct discharge valve lineup independently verified by two qualified individuals.	Prior to release of radioactive effluents monitored by the affected channel

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Effluent or Sampler Flow Monitor non-functional.	B.1 Estimate flow. OR B.2 Suspend the release of radioactive effluents monitored by the affected channel.	Once per 4 hours Immediately
C. --------------NOTE-------------- 1. Applicable to releases other than those described in Condition A above. 2. Applicable to SPINGS 4 and 8 only when pathway is in service. ------------------------------------ Required Noble Gas Activity Monitor non-functional.	--------------------NOTE------------------ If ANO-1 Reactor Building Purge and Ventilation required Noble Gas Activity Monitor inoperable and moving irradiated fuel within the ANO-1 Reactor Building, refer to ANO-1 TS 3.9.3. --- C.1 Obtain sample of effluent. AND C.2 Analyze sample of effluent.	 12 hours Within 24 hours following completion of Required Action C.1
D. --------------NOTE-------------- Applicable to SPINGS 4 and 8 only when pathway is in service. ------------------------------------ Required Iodine and/or Particulate Sampler non-functional.	D.1 Verify effluent samples are continuously collected by auxiliary sampling equipment. AND D.2 Replace Iodine and/or Particulate cartridges (as applicable). AND D.3 Analyze Iodine and/or Particulate cartridges (as applicable).	4 hours 7 days Within 48 hours following replacement

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action(s) and/or Completion Time(s) of Condition C and/or Condition D not met.	E.1 Suspend the release of radioactive effluents monitored by the affected channel.	Immediately
F. Required Action(s) and/or Completion Time(s) Condition A, B, and/or E not met.	F.1 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately
G. Required Radioactive Gaseous Effluent Monitoring Instrument non-functional for > 30 days.	G.1 Initiate a condition report to document and track the condition for inclusion in the Radioactive Effluent Release Report pursuant to TS 5.6.3 (ANO-1) / TS 6.6.3 (ANO-2).	Immediately

SURVEILLANCES

SURVEILLANCE		FREQUENCY
S 2.2.1.1	----------NOTE---------- Not applicable to Iodine and Particulate Samplers ------------------------------ Perform a CHANNEL CHECK of required instrumentation.	24 hours
S 2.2.1.2	Verify presence of required Iodine Sampler Cartridge and required Particulate Sample Filter.	7 days
S 2.2.1.3	Perform a CHANNEL TEST of the required Reactor Building Purge and Ventilation System Gas Activity Process and Noble Gas Activity Monitors.	31 days prior to initiating Reactor Building Purge and/or Ventilation activities

SURVEILLANCES (continued)

SURVEILLANCE		FREQUENCY
S 2.2.1.4	---NOTES--- SOURCE CHECK not required when background radioactivity is greater than the check source. --- Perform a SOURCE CHECK on the required Noble Gas Activity Monitors.	31 days
S 2.2.1.5	---NOTES--- 1. SOURCE CHECK not required when background radioactivity is greater than the check source. 2. Only applicable to Waste Gas Holdup and Reactor Building Purge Systems. --- Perform a SOURCE CHECK on the required Gas Activity Process and Noble Gas Activity Monitors.	Within 14 days prior to release of radioactive effluents monitored by the channel
S 2.2.1.6	Perform a CHANNEL TEST of the required Noble Gas Activity Monitors.	92 days
S 2.2.1.7	---NOTE--- Not applicable to Iodine and Particulate Samplers --- Perform a CHANNEL CALIBRATION on the required instrumentation.	18 months

L 2.3 RADIOACTIVE LIQUID EFFLUENTS

L 2.3.1 Radioactive material released to the discharge canal shall:

a. For dissolved or entrained noble gases, be limited to a total concentration of $\leq 2 \times 10^{-4}$ µCi/ml.

b. For radioactive nuclides other than dissolved or entrained noble gases, be limited to the concentration specified in 10 CFR 20, Appendix B, Table II, Column 2.

c. During any calendar quarter, result in a dose commitment to a MEMBER OF THE PUBLIC of ≤ 1.5 mrem to the total body and ≤ 5 mrem to any organ.

d. During any calendar year, result in a dose commitment to a MEMBER OF THE PUBLIC of ≤ 3 mrem to the total body and ≤ 10 mrem to any organ.

e. Be processed by a LIQUID RADWASTE TREATMENT SYSTEM when accumulative dose during a calendar quarter is projected to exceed 0.18 mrem to the total body and/or 0.625 mrem to any organ.

APPLICABILITY: At all times.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each Limitation L 2.3.1.a through L 2.3.1.e above and for each BATCH RELEASE and CONTINUOUS RELEASE Surveillance requirement not met.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any limit listed in L 2.3.1.a through L 2.3.1.e not met.	A.1 Initiate action to restore to within limit.	Immediately
	<u>AND</u>	
	A.2 Initiate a condition report to document the condition, determine any limitations for the continued effluent release operations, and track the condition for inclusion in the Radioactive Effluent Release Report pursuant to TS 5.6.3 (ANO-1) / TS 6.6.3 (ANO-2).	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. --------------NOTE-------------- Only applicable to BATCH RELEASE. ------------------------------------ Sampling and/or analysis requirements not met.	B.1 Verify associated effluent release suspended. AND B.2 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately Immediately
C. --------------NOTE-------------- Only applicable to CONTINUOUS RELEASE of secondary coolant. ------------------------------------ Secondary coolant dose equivalent I-131 (DEI) > 0.01 µCi/ml.	C.1 Obtain a grab sample of the associated secondary coolant. AND C.2 Perform gamma isotopic and I-131 analysis of sample.	12 hours 12 hours following sample acquisition
D. Annual dose limits of L 2.3.1.d projected to exceed 40 CFR 190 limits.	D.1 Apply for a variance from the NRC to permit releases in excess of 40 CFR 190 limits.	Prior to exceed 40 CFR 190 limits
E. Required Action(s) and/or Completion Time(s) of Conditions C and/or D not met. OR Sampling and/or analysis requirements not met.	E.1 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Lower Limit(s) of Detection (LLD) not met.	F.1 Initiate a condition report to document and track the condition for inclusion in the Radioactive Effluent Release Report pursuant to TS 5.6.3 (ANO-1) / TS 6.6.3 (ANO-2).	Immediately

SURVEILLANCES

SURVEILLANCE		FREQUENCY
S 2.3.1.1	----------------------------------NOTE---------------------------------- Only applicable to BATCH RELEASE. ---	
	Obtain representative sample of each batch.	Prior to release
	AND	
	Perform gamma isotopic and I-131 analysis of sample.	Prior to release
	AND	
	Perform dissolved and entrained gas analysis of sample.	31 days following sample acquisition
	AND	
	Perform gross alpha composite and H-3 analysis of sample.	31 days following sample acquisition
	AND	
	Perform Sr-89, Sr-90, and Fe-55 composite analysis of sample.	92 days following sample acquisition

SURVEILLANCES (continued)

	SURVEILLANCE	FREQUENCY
S 2.3.1.2	----------------------------------NOTE---------------------------------- Only applicable to CONTINUOUS RELEASE. --	
	Obtain representative sample of effluent.	24 hours
	AND	
	Perform gamma isotopic and I-131 analysis.	24 hours following sample acquisition
	AND	
	Perform dissolved and entrained gas analysis.	31 days following sample acquisition
	AND	
	Perform gross alpha composite and H-3 analysis.	31 days following sample acquisition
	AND	
	Perform Sr-89, Sr-90, and Fe-55 composite analysis.	92 days following sample acquisition
S 2.3.1.3	Using data acquired by performance of S 2.3.1.1 and S.2.3.1.2, verify Limitations L 2.3.1.a through L 2.3.1.e continue to be met.	Within 7 days following completion of each required analysis
S 2.3.1.4	Using data acquired by performance of S 2.3.1.1 and S.2.3.1.2, verify the limits of 40 CFR 190 are not projected to be exceeded.	31 days
S 2.3.1.5	Verify the following LLDs are met: Gamma isotopic 5×10^{-7} µCi/ml I-131 and Fe-55 1×10^{-6} µCi/ml Dissolved/entrained gases (gamma emitters) 1×10^{-5} µCi/ml H-3 1×10^{-5} µCi/ml Gross alpha 1×10^{-7} µCi/ml Sr-89 and Sr-90 5×10^{-8} µCi/ml	12 months

L 2.4 RADIOACTIVE GASEOUS EFFLUENTS

L 2.4.1 Radioactive Gaseous Effluent releases to unrestricted areas shall:

---NOTE---
Dose rates associated with Reactor (Containment) Building Purge operations may be averaged over a one-hour interval.
----

a. For noble gases, be limited to:

   1. A total body dose rate of ≤ 500 mrem/yr.

   2. A skin dose rate of ≤ 3000 mrem/yr.

   3. A dose commitment to a MEMBER OF THE PUBLIC in any calendar quarter of ≤ 5 mrads gamma and ≤ 10 mrads beta radiation.

   4. A dose commitment to a MEMBER OF THE PUBLIC in any calendar year of ≤ 10 mrads gamma and ≤ 20 mrads beta radiation.

b. For I-131, H-3, and for all radionuclides in particulate form having a half-life of > 8 days, be limited to:

   1. An organ dose rate of ≤ 1500 mrem/yr.

   2. A dose commitment to a MEMBER OF THE PUBLIC in any calendar quarter of ≤ 7.5 mrem to any organ.

   3. A dose commitment to a MEMBER OF THE PUBLIC in any calendar year of ≤ 15 mrem to any organ.

c. Be processed by a VENTILATION EXHAUST TREATMENT SYSTEM when:

   1. For noble gases, the dose over a calendar quarter is project to exceed 0.625 mrads gamma and/or 1.25 mrads beta radiation.

   2. For I-131, H-3, and for all radionuclides in particulate form having a half-life of > 8 days, the dose over a calendar quarter is project to exceed 1.0 mrem to any organ.

d. Be processed by the GASEOUS RADWASTE TREATMENT SYSTEM when degasifying the Reactor Coolant System (RCS), if projected dose would exceed 0.625 mrads gamma and/or 1.25 mrads beta radiation over a calendar quarter.

APPLICABILITY: At all times.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Limitation L 2.4.1.a through L 2.4.1.d above and for each Surveillance requirement not met.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any limit listed in L 2.4.1.a through L 2.4.1.d not met.	A.1 Initiate action to restore to within limit.	Immediately
	<u>AND</u>	
	A.2 Initiate a condition report to document the condition, determine any limitations for the continued effluent release operations, and track the condition for inclusion in the Radioactive Effluent Release Report pursuant to TS 5.6.3 (ANO-1) / TS 6.6.3 (ANO-2).	Immediately
B. Sampling and/or analysis requirements of S 2.4.1.1 not met.	B.1 Verify associated effluent release suspended.	Immediately
	<u>AND</u>	
	B.2 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately
C. Annual dose limits of L 2.4.1.a.4 and/or L 2.4.1.b.3 projected to exceed 40 CFR 190 limits.	C.1 Apply for a variance from the NRC to permit releases in excess of 40 CFR 190 limits.	Prior to exceed 40 CFR 190 limits

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action(s) and/or Completion Time(s) of Condition C not met. <u>OR</u> Sampling and/or analysis requirements of S 2.4.1.2 not met.	D.1 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately
E. Lower Limit(s) of Detection (LLD) not met.	E.1 Initiate a condition report to document and track the condition for inclusion in the Radioactive Effluent Release Report pursuant to TS 5.6.3 (ANO-1) / TS 6.6.3 (ANO-2).	Immediately

SURVEILLANCES

SURVEILLANCE	FREQUENCY
S 2.4.1.1 -----------------------------------NOTE----------------------------------- Only applicable to Waste Gas Storage Tank and Reactor Building Purge release. ---	
Obtain representative sample of gas to be released.	Prior to release
<u>AND</u>	
Analyze sample for principal gamma emitters.	Prior to release
<u>AND</u>	
-----------------------------------NOTE----------------------------------- Only applicable to Reactor Building Purge release. ---	
Perform H-3 analysis of sample.	Prior to release

SURVEILLANCES (continued)

SURVEILLANCE		FREQUENCY
S 2.4.1.2	-----------------------------------NOTE----------------------------------- Only applicable to Auxiliary Building, Spent Fuel Pool Area, Auxiliary Building Extension Area (ANO-2), Low Level Radwaste Building, Emergency Penetration Room, and Reactor (Containment) Building Ventilation systems. ---	
	The following effluent samples shall be obtained to support the radioactive analysis specified:	
	a. -----------------------------------NOTE------------------------------ Only applicable to Reactor Building Ventilation when Reactor Vessel Head is removed. --	
	Representative sample for H-3 analysis.	24 hours
	b. -----------------------------------NOTE------------------------------ Only applicable to Spent Fuel Pool Area Ventilation. --	
	Representative sample for H-3 analysis.	7 days
	c. Charcoal sample for I-131 analysis.	7 days
	d. Particulate sample for principal gamma emmiters analysis.	7 days
	e. Particulate sample for composite gross alpha analysis.	31 days
	f. Representative sample for principal gamma emmiters analysis.	31 days
	g. Representative sample for H-3 analysis.	31 days
	h. Particulate sample of for Sr-89 and Sr-90 composite analysis.	92 days
	AND	
	(continued)	*(continued)*

SURVEILLANCES (continued)

SURVEILLANCE		FREQUENCY
S 2.4.1.2	*(continued)* Complete analysis of above samples: i. Samples a, b, c, and d	 48 hours following sample acquisition
	j. Samples e, f, and g	31 days following sample acquisition
	k. Sample h	60 days following sample acquisition
S 2.4.1.3	Record SPING Noble Gas activity.	24 hours
S 2.4.1.4	Using data acquired by performance of S 2.4.1.1 and S.2.4.1.2, verify Limitations L 2.4.1.a through L 2.4.1.d continue to be met.	31 days
S 2.4.1.5	Using data acquired by performance of S 2.4.1.1 and S.2.4.1.2, verify the limits of 40 CFR 190 are not projected to be exceeded.	31 days
S 2.4.1.6	Verify the following LLDs are met: Principal gamma emitters (gaseous) 1×10^{-4} µCi/ml Principal gamma emitters (particulate) 1×10^{-11} µCi/ml I-131 1×10^{-12} µCi/ml H-3 1×10^{-6} µCi/ml Gross alpha 1×10^{-11} µCi/ml Sr-89 and Sr-90 1×10^{-11} µCi/ml Noble gas (dose equivalent Xe-133) 1×10^{-6} µCi/ml	12 months

L 2.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

L 2.5.1 The following environmental sample locations shall be designated and maintained:

---NOTE---
Other instruments may be used in place of, or in addition to, integrating dosimeters.

Pathway / Sample Type		*#*	*Location*
Airborne Radionuclide and Particulate		3	Samples close to site boundary in or near different sectors having the highest calculated annual average ground-level D/Q
		1	Sample from the vicinity of a community having the highest calculated annual average ground-level D/Q
		1	Background information sample from a control location 10-20 miles from one reactor building
Direct Radiation		16	Inner ring stations with 2 or more dosimeters in each meteorological sector in the general area of the site boundary
		8	Stations with 2 or more dosimeters in special interest areas such as population centers, nearby residences, schools, and in 1-2 areas to serve as control locations.
Waterborne	Surface Water	1	Indicator location influenced by plant discharge
		1	Control location uninfluenced by plant discharge
	Drinking Water	1	Indicator location influenced by plant discharge
		1	Control location uninfluenced by plant discharge
	Shoreline Sediment	1	Indicator location influenced by plant discharge
		1	Control location uninfluenced by plant discharge
	Ground Water	1	Indicator location influenced by plant discharge
		1	Control location uninfluenced by plant discharge
Ingestion	Milk	1	Indicator location within 5 miles of one reactor, if commercially available
		1	Control location > 5 miles from one reactor when an indicator exists
	Fish	1	Sample of commercially and/or recreationally important species in vicinity of plant discharge
		1	Sample of same species in area not influenced by plant discharge
	Food Products	1	Sample of broadleaf (edible or inedible) near the site boundary from one of the highest anticipated annual average ground-level D/Q sectors
		1	Sample location of broadleaf vegetation (edible or inedible) from a control location 10-20 miles from one reactor

L 2.5.1

APPLICABILITY: At all times.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each sample location and Surveillance requirement.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Sample location requirement not met. <u>OR</u> Required sample equipment non-functional. <u>OR</u> Sample Frequency not met. <u>OR</u> Sample analysis Frequency not met. <u>OR</u> One or more Lower Limit(s) of Detection (LLD) listed in Table 2.5-1 not met. <u>OR</u> One or more limits listed in Table 2.5-2 not met. <u>OR</u> Dose to a MEMBER OF THE PUBLIC from radionuclides other than those listed in Table 2.5-2 projected to exceed calendar year limits of L 2.3.1 and/or L 2.4.1.	A.1 Initiate action to restore to within limits. <u>AND</u> A.2 Initiate a condition report to document and track the condition for inclusion in the Annual Radiological Environmental Operating Report pursuant to TS 5.6.2 (ANO-1) / TS 6.6.2 (ANO-2).	Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Sample(s) from required sample location(s) unavailable.	B.1 Identify and add to the Radiological Environment Monitoring Program, locations for obtaining replacement samples.	30 days

SURVEILLANCES

SURVEILLANCE		FREQUENCY
S 2.5.1.1	----------NOTE---------- Only applicable to Airborne Radionuclide and Particulate. ---------------------------- Collect sample from continuous sampler. <u>AND</u> Perform I-131 analysis of radioiodine canister. <u>AND</u> Perform gross beta analysis of particulate sampler.	 14 days 14 days following sample acquisition ≥ 24 hours and ≤ 14 days following filter change
S 2.5.1.2	----------NOTE---------- Only applicable to Direct Radiation locations. ---------------------------- Collect sample from required location. <u>AND</u> Perform gamma dose analysis of sample.	 92 days 60 days following sample acquisition

SURVEILLANCES (continued)

SURVEILLANCE		FREQUENCY
S 2.5.1.3	----------------------------------NOTE---------------------------------- Only applicable to Surface Water samples. --	
	Collect sample from required location.	92 days
	AND	
	Perform gamma isotopic analysis of sample.	21 days following sample acquisition
	AND	
	Perform H-3 analysis of sample.	31 days following sample acquisition
S 2.5.1.4	----------------------------------NOTE---------------------------------- Only applicable to Drinking and Ground Water samples. --	
	Collect sample from required location.	92 days
	AND	
	Perform gamma isotopic analysis of sample.	21 days following sample acquisition
	AND	
	Perform H-3 analysis of sample.	31 days following sample acquisition
	AND	
	Perform I-131 analysis of sample.	21 days following sample acquisition
	AND	
	Perform gross beta analysis of sample.	31 days following sample acquisition

SURVEILLANCES (continued)

	SURVEILLANCE	FREQUENCY
S 2.5.1.5	----------------------------------NOTE---------------------------------- Only applicable to Waterborne Shoreline Sediment samples. -- Collect sample from required location. <u>AND</u> Perform gamma isotopic analysis of sample.	 12 months 60 days following sample acquisition
S 2.5.1.6	----------------------------------NOTE---------------------------------- Only applicable to Milk samples. -- Collect sample from required location. <u>AND</u> Perform gamma isotopic analysis of sample. <u>AND</u> Perform I-131 analysis of sample.	 92 days 21 days following sample acquisition 21 days following sample acquisition
S 2.5.1.7	----------------------------------NOTE---------------------------------- Only applicable to edible portions of Fish samples. -- Collect sample from required location. <u>AND</u> Perform gamma isotopic analysis of sample.	 12 months 60 days following sample acquisition

SURVEILLANCES (continued)

SURVEILLANCE		FREQUENCY
S 2.5.1.8	---------------------------------NOTES---------------------------------- 1. Only applicable to Food Product samples. 2. Only applicable if Milk sampling not performed. ---	
	Collect sample from required location.	12 months
	AND	
	Perform gamma isotopic analysis of sample.	21 days following sample acquisition
	AND	
	Perform I-131 analysis of sample.	21 days following sample acquisition
S 2.5.1.9	Verify the LLDs listed in Table 2.5-1 are met.	12 months
S 2.5.1.10	Verify radioactivity concentrations are less than or equal to the limits listed in Table 2.5-2, when averaged over a calendar quarter.	92 days

TABLE 2.5-1

MAXIMUM VALUES OF THE LOWER LIMITS OF DETECTION (LLD)

Analyses	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[a]	1×10^{-2}[b]				
H-3	2000[c]					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	1[d]	7×10^{-2}[e]		1	60	
Cs-134	15	5×10^{-2}[f]	130	15	60	150
Cs-137	18	6×10^{-2}[f]	150	18	80	180
Ba-140	60			60		
La-140	15			15		

(a) LLD for drinking water.

(b) Only applicable to particulate.

(c) LLD for drinking water. When no drinking water pathway exists, a value of 3000 pCi/l may be used.

(d) LLD for drinking water. When no drinking water pathway exists, a gamma isotopic analysis LLD value of 15 pCi/l may be used.

(e) Only applicable to gas.

(f) Only applicable to particulate gamma isotopic analysis.

L 2.5.1

TABLE 2.5-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Analyses	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	2×10^4 [(a)]				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-95, Nb-95	4×10^2 [(b)]				
I-131	2 [(c)]	0.9		3	1×10^2
Cs-134	30	10 [(d)]	1×10^3	60	1×10^3
Cs-137	50	20 [(d)]	2×10^3	70	2×10^3
Ba-140, La-140	2×10^2 [(b)]			3×10^2 [(b)]	

[(a)] Drinking water samples.

[(b)] Total for parent and daughter.

[(c)] LLD for drinking water. When no drinking water pathway exists, a value of 20 pCi/l may be used.

[(d)] Applicable when performing a gamma isotopic analysis of individual particulate samples with gross beta activity more than 10 times the yearly mean of control samples.

L 2.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

L 2.5.2 --NOTE--
Broad leaf vegetation sampling may be performed at the site boundary in the directional sector with the highest D/Q in lieu of the garden census.

The location of the nearest milk animal, the nearest residence, and the nearest garden of greater than 500 ft^2 producing fresh leafy vegetables in each of the 16 meteorological sectors within a 5-mile distance from one reactor (containment) building shall be identified.

APPLICABILITY: At all times.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each sample location.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. New sample location identified which yields a calculated dose due to I-131, H-3, and/or particulates projected to exceed 40 CFR 190 limits. <u>OR</u> New sample location identified which yields a calculated dose via the same exposure pathway in excess of values calculated at sample locations of Limitation L 2.5.1.	A.1 Initiate a condition report to document and track the condition for inclusion in the Annual Radiological Environmental Operating Report pursuant to TS 5.6.2 (ANO-1) / TS 6.6.2 (ANO-2).	Immediately
	<u>AND</u>	
	A.2.1 Identify and add the new sample location to the Radiological Environment Monitoring Program.	30 days
	<u>AND</u>	
	A.2.2 Delete the previous sample location via the associated exposure pathway from the Radiological Environment Monitoring Program.	Within 90 days following October 31 of the year in which the new sample location was identified.

SURVEILLANCES

---NOTE---
S 2.0.2 is not applicable to the Surveillances of this Limitation.

SURVEILLANCE		FREQUENCY
S 2.5.2.1	A land use census to identify the locations described in Limitation L 2.5.2 shall be performed by door-to-door survey, aerial survey, or by consulting local agricultural authorities.	24 months between June 1 and October 1
S 2.5.2.2	Include the results of S 2.5.2.1 in the Annual Radiological Environmental Operating Report pursuant to TS 5.6.2 (ANO-1) / TS 6.6.2 (ANO-2).	12 months

L 2.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

L 2.5.3 Radioactive materials supplied as part of the Interlaboratory Comparison Program shall be analyzed.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Limitation not met.	A.1 Initiate a condition report to document and track the condition for inclusion in the Annual Radiological Environmental Operating Report pursuant to TS 5.6.2 (ANO-1) / TS 6.6.2 (ANO-2).	Immediately

SURVEILLANCES

---NOTE---
S 2.0.2 is not applicable to the Surveillances of this Limitation.

SURVEILLANCE	FREQUENCY
S 2.5.3.1 Include the results of analyses performed as part of the Interlaboratory Comparison Program in the next Annual Radiological Environmental Operating Report pursuant to TS 5.6.2 (ANO-1) / TS 6.6.2 (ANO-2).	12 months

B 2.0 LIMITITATION (L) APPLICABILITY

BASES

Limitations	L 2.0.1 through L 2.0.5 establish the general requirements applicable to all Limitations and apply at all times, unless otherwise stated.
B 2.0.1	L 2.0.1 establishes the Applicability statement within each individual Limitation as the requirement for when the Limitation is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Limitation).
B 2.0.2	L 2.0.2 establishes that upon discovery of a failure to meet a Limitation, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of a Limitation are not met. This Limitation establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Limitation; and b. Completion of the Required Actions is not required when a Limitation is met within the specified Completion Time, unless otherwise specified. Completing the Required Actions is not required when a Limitation is no longer applicable, unless otherwise stated in the individual Specification.
B 2.0.3	L 2.0.3 establishes the Required Actions that must be implemented when a Limitation is not met and the condition is not specifically addressed by the associated Conditions. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. This requirement is intended to provide assurance that plant management is aware of the condition and to ensure that the condition is evaluated for its affect on continued operation of the plant.
B 2.0.4	L 2.0.4 establishes Limitations on changes in MODES or other specified conditions in the Applicability when a Limitation is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the Limitation would not be met, in accordance with Limitation L 2.0.4.a, L 2.0.4.b, or L 2.0.4.c.

BASES

LIMITATION APPLICABILITY (continued)

B 2.0.4 (continued)

L 2.0.4 allows entry into a MODE or other specified condition in the Applicability with the Limitation not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Limitation should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to FUNCTIONAL status before entering an associated MODE or other specified condition in the Applicability.

Upon entry into a MODE or other specified condition in the Applicability with the Limitation not met, L 2.0.1 and L 2.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the Limitation is met, or until the unit is not within the Applicability of the Limitation.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by S 2.0.1. Therefore, utilizing L 2.0.4 is not a violation of S 2.0.1 or S 2.0.4 for any Surveillances that have not been performed on equipment. However, Surveillances must be met to ensure FUNCTIONALITY prior to declaring the associated equipment FUNCTIONAL (or variable within limits) and restoring compliance with the affected Limitation.

B 2.0.5

L 2.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared non- functional to comply with ACTIONS. The sole purpose of this Limitation is to provide an exception to L 2.0.2 (e.g., to not comply with the applicable Required Actions) to allow the performance of required testing to demonstrate:

a. The FUNCTIONALITY of the equipment being returned to service; or

b. The FUNCTIONALITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate FUNCTIONALITY. This Limitation does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the FUNCTIONALITY of the equipment being returned to service is restarting a ventilation system that has been secured to comply with Required Actions and must be restarted to perform the required testing.

B 2.0 SURVEILLANCE (S) APPLICABILITY

BASES

S 2.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual Limitations, unless otherwise stated in the individual Surveillance. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limitation. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the Limitation except as provided in S 2.0.3. Surveillances are not required to be performed on non-functional equipment or variables outside specified limits.
S 2.0.2	The specified Frequency for each Surveillance is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Limitation are stated in the individual Limitations.
S 2.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the Limitation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the Limitation must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the Limitation must immediately be declared not met, and the applicable Condition(s) must be entered.
S 2.0.4	Entry into a MODE or other specified condition in the Applicability of a Limitation shall only be made when the Limitation's Surveillances have been met within their specified Frequency, except as provided by S 2.0.3. When a Limitation is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with L 2.0.4.

B 2.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

BASES

BACKGROUND

The Radioactive Liquid Effluent Monitoring Instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases.

LIMITATION

The following Radioactive Liquid Effluent Monitoring Instrumentation is required to be FUNCTIONAL:

ANO-1: RE-4642 – Liquid Radwaste Monitor
RE-2682 – “A” Main Steam Line Radiation Monitor
RE-2681 – “B” Main Steam Line Radiation Monitor

ANO-2: 2RE-2330 – Liquid Radwaste Monitor
2RE-4423 – Liquid Radwaste Monitor

Both radiation monitoring and flow monitoring capability are required to be FUNCTIONAL for each Liquid Radwaste Monitor. With regard to Liquid Radwaste radiation monitoring, the alarm/trip function must also be FUNCTIONAL. The alarm/trip setpoints for these instruments are calculated in accordance with the methods contained in ODCM Section 2.1 to ensure that the alarm/trip will occur prior to potentially exceeding the limits of 10 CFR Part 20.

With regard to the Main Steam Line Radiation Monitors, these monitors must have a measurement range capability from 10^{-1} mR/hr to 10^{4} mR/hr.

APPLICABILITY

The Liquid Radwaste Monitors are required to be FUNCTIONAL during any release via the pathway in which the monitor is installed. The Main Steam Line Radiation Monitors are required to be FUNCTIONAL in MODES 1, 2, 3, and 4.

ACTIONS

The following ACTIONS are generally applicable to the pathway in which a radioactive liquid release is in progress. Because more than one release could occur simultaneously, the ACTIONS are modified by a Note that permits separate Condition entry for each non-functional Radioactive Liquid Effluent Monitoring Instrument.

ACTIONS (continued)

A.1

If the radiation monitoring feature of the Radioactive Liquid Effluent Monitoring Instrument is non-functional, any release via the associated pathway must be suspended immediately. This prevents the release of unmonitored effluents to the environment.

A.2.1

In addition to Required Action A.1, a non-functional radiation monitoring feature of a Radioactive Liquid Effluent Monitoring Instrument must be returned to a FUNCTIONAL status prior to the restart or subsequent release of effluents via the associated pathway. This prevents the release of unmonitored effluents to the environment. Exceptions to this requirement are included in Required Actions A.2.2.1 through A.2.2.3 below.

A.2.2.1 through A.2.2.3

In lieu of performing Required Action A.2.1 above, grab samples may be obtained and analyzed to provide a backup monitoring method for the effluent release. Because of the importance of monitoring radioactive liquid releases, two independent samples of the effluent must be obtained and analyzed. The independency required is with regard to obtaining and analyzing each sample separately. Two independent personnel are not required to obtain and analyze the two samples.

Notwithstanding the above, computer input data and the discharge valve lineup associated with the effluent release path must be verified by two independent, qualified individuals. Integrity of independence is maintained by preventing interaction between personnel during the verification process. With regard to valve lineups, independent verification is conducted such that each check constitutes actual identification of the valve and a determination of both “required” and “actual” valve position.

B.1 and B.2

If the flow monitoring feature of the Radioactive Liquid Effluent Monitoring Instrument is non-functional, the flow rate may be estimated within 4 hours of initial loss of the instrument and every 4 hours thereafter, for the duration of the effluent release. Flow rate data is necessary to calculate the amount of radioactive released via the effluent discharge. The 4-hour Completion Time is reasonable because a significant change in flow rate over the course of an effluent release is unlikely.

S 2.0.2 is not applicable to the initial flow estimation, but may be applied to the flow estimations thereafter. Pump curves may be used to estimate flow.

ACTIONS (continued)

C.1

If one or more Main Steam Line Radiation Monitors is non-functional, the pre-planned alternate monitoring method of monitoring must be established within 72 hours. The alternate method chosen should ensure continued monitoring of the Main Steam system for radiation while operating in MODES 1, 2, 3, or 4. In addition, the affected monitor(s) must be restored to a FUNCTIONAL status within 7 days.

D.1

If the Required Actions and associated Completion Times of Conditions A, B, and/or C cannot be met, then additional measures may be necessary to ensure continued safe operation or to reduce overall station risk. Therefore, a condition report must be initiated immediately to assess the impact on continued effluent release operations given the degraded condition.

E.1

Instrumentation installed to ensure radiological monitoring of effluent releases is expected to be normally available in accordance with the design function or purpose of the equipment. During releases via a respective pathway, instrumentation that remains non-functional for greater than 30 days may indicate inappropriate importance placed on the equipment or over-reliance on the backup sampling method for effluent release monitoring. As an incentive to avoid either of these conditions, Radioactive Liquid Effluent Monitoring Instrumentation that remains non-functional for more than 30 days must be included in the Radioactive Effluent Release Report submitted pursuant to TS 5.6.3 (ANO-1) or TS 6.6.3 (ANO-2). In order to ensure inclusion, Required Action E.1 requires the condition to be tracked via a condition report.

Information to be provided in the respective Radioactive Effluent Release Report should include 1) the component number and noun name, 2) the failure mode, 3) the reason for continued inoperability, and 4) the expected return to service date.

SURVEILLANCES

S 2.1.1.1

Performance of the CHANNEL CHECK every 24 hours provides reasonable assurance for prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. Where parameter comparison is not possible, the CHANNEL CHECK will continue to identify gross instrument failure such as loss of power, unexpected upscale readings, failed-low indications, etc. The CHANNEL CHECK is key in verifying that the instrumentation continues to operate properly between CHANNEL CALIBRATIONs. The Frequency is based on unit operating experience that demonstrates channel failure is rare.

SURVEILLANCES (continued)

S 2.1.1.2

A CHANNEL TEST is performed on the radiation monitoring portion of each required instrument channel to ensure the entire channel will perform the intended functions. The CHANNEL TEST demonstrates that automatic isolation of the associated pathway and Control Room alarm occur should the instrument indicate measured levels above the trip setpoint. The channel test also demonstrates that alarm occurs when any of the following conditions exist:

A. Power to the detector is lost.

B. The instrument indicates a downscale failure.

C. Instrument controls are not set in the operate mode.

Any setpoint adjustment shall be consistent with Section 2.1 of the ODCM.

The Surveillance is modified by a Note clarifying that the CHANNEL TEST is applicable only to the radiation detection portion of the monitor function and is not applicable to the flow monitoring function. The Frequency of 92 days is based on unit operating experience, with regard to channel FUNCTIONALITY and drift, which demonstrates that failure of a channel in any 92-day interval is a rare event, especially in light of the infrequency of radioactive liquid releases.

S 2.1.1.3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift (as required) to ensure that the instrument channel remains FUNCTIONAL between successive tests. CHANNEL CALIBRATION shall find that measurement errors and setpoint errors are within the assumptions of the setpoint calculations. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint calculations. This Frequency is justified by the assumption of at least an 18-month calibration interval to determine the magnitude of equipment drift or deviation in the setpoint calculations.

Initial CHANNEL CALIBRATION is performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration are used.

SURVEILLANCES (continued)

S 2.1.1.4

A SOURCE CHECK provides a qualitative assessment of channel response when the channel sensor is exposed to the radioactive source. This check is performed within 8 hours prior to release of effluent via the associated flow path. When a SOURCE CHECK can be performed, it provides verification that the sensor will respond to an increase in radiation level. Note 1, however, does not require a SOURCE CHECK when the background radiation at the sensor is greater than the check source. This is acceptable because of the other required tests above (CHANNEL CHECK, CHANNEL TEST, CHANNEL CALIBRATION). The 8-hour restriction is reasonable because it is unlikely that the sensor will unexpectedly fail in any 8-hour period.

Note 2 provides clarification that the SOURCE CHECK applies only to the radiation detection portion of the Liquid Radwaste Monitor and is not applicable to the flow monitor portion or to the Main Steam Line Radiation Monitors.

B 2.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

BASES

BACKGROUND

The Radioactive Gaseous Effluent Monitoring Instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases.

LIMITATION

The following Radioactive Gaseous Effluent Monitoring Instrumentation is required to be FUNCTIONAL:

--NOTE--
Refer to ANO-2 Technical Specification (TS) 3.3.3.1 for ANO-2 Containment Building Purge System Process Monitor (2RE-8233) operability requirements and associated ACTIONS.

ANO-1: RE-4830 – Waste Gas Holdup System Process Monitor*
RX-9820 – Reactor Building Purge and Ventilation SPING
RX-9825 – Auxiliary Building Ventilation SPING
RX-9830 – Spent Fuel Pool Area Ventilation SPING
RX-9835 – Emergency Penetration Room Ventilation SPING

ANO-2: 2RE-2429 – Waste Gas Holdup System Process Monitor*
2RX-9820 – Containment Building Purge and Ventilation SPING
2RX-9825 – Auxiliary Building Ventilation SPING
2RX-9830 – Spent Fuel Pool Area Ventilation SPING
2RX-9835 – Emergency Penetration Room Ventilation SPING
2RX-9845 – Auxiliary Building Extension Ventilation SPING
2RX-9850 – Radwaste Storage Building Ventilation SPING

* These monitors provide automatic isolation.

The radiation monitoring (process gas and SPING noble gas) and effluent flow monitoring capability are required to be FUNCTIONAL for each monitor. For SPING monitors the sample flow monitoring, the iodine sample, and the particulate sampler must also be FUNCTIONAL. With regard to Waste Gas Holdup System radiation monitoring, the alarm/trip function must also be FUNCTIONAL. The alarm/trip setpoints for specified instruments are calculated in accordance with the methods contained in ODCM Section 3.1 to ensure that the alarm/trip will occur prior to potentially exceeding the limits of 10 CFR Part 20.1301. Note that the PURGE function of the ANO-1 and ANO-2 Reactor (Containment) Building is treated separately from the ventilation function.

Performance of a SOURCE CHECK on a given radiation monitor does not require the monitor to be declared non-functional due to the short period of time required to perform this test.

APPLICABILITY

The above monitors are required to be FUNCTIONAL during any release via the pathway in which the monitor is installed. Because SPINGs 4 and 8 monitor the Emergency Penetration Room Ventilation of ANO-1 and ANO-2, respectively, and because these ventilation systems are normally aligned for auto-start capability to aid in accident mitigation, these SPINGs must be FUNCTIONAL whenever the associated ventilation system is available for auto-start.

ACTIONS

The following ACTIONS are applicable to the pathway in which a radioactive gaseous release is in progress. Because more than one release could occur simultaneously, the ACTIONS are modified by a Note that permits separate Condition entry for each non-functional Radioactive Gaseous Effluent Monitoring Instrument.

A.1

If the radiation monitoring feature, including the alarm/trip function for monitors having an automatic isolation feature, of the Waste Gas Holdup or ANO-1 Reactor Building Purge and Ventilation System Gas Activity Process or Noble Gas Activity Monitor(s) is non-functional, any release via the associated pathway must be suspended immediately. This prevents the release of unmonitored effluents to the environment.

A.2.1

In addition to Required Action A.1, a non-functional Waste Gas Holdup or ANO-1 Reactor Building Purge and Ventilation System Gas Activity Process or Noble Gas Activity Monitor, including the alarm/trip function for monitors having an automatic isolation feature, must be returned to a FUNCTIONAL status prior to the restart or subsequent release of effluents via the associated pathway. This prevents the release of unmonitored effluents to the environment. Exceptions to this requirement are included in Required Actions A.2.2.1 through A.2.2.3 below.

A.2.2.1 through A.2.2.3

In lieu of performing Required Action A.2.1 above, grab samples may be obtained and analyzed to provide a backup monitoring method for the effluent release. Because of the importance of monitoring radioactive gaseous releases, two independent samples of the effluent must be obtained and analyzed. The independency required is with regard to obtaining and analyzing each sample separately. Two independent personnel are not required to obtain and analyze the two samples.

ACTIONS (continued)

A.2.2.1 through A.2.2.3 (continued)

Notwithstanding the above, computer input data and the discharge valve lineup associated with the effluent release path must be verified by two independent, qualified individuals. Integrity of independence is maintained by preventing interaction between personnel during the verification process. With regard to valve lineups, independent verification is conducted such that each check constitutes actual identification of the valve and a determination of both "required" and "actual" valve position. Required Action A.2.2.3 is modified by a Note that excepts the valve lineup requirement from the ANO-1 Reactor Building Purge and Ventilation System since no manual valves are manipulated for this release path.

B.1 and B.2

If the flow monitoring features of the Radioactive Gaseous Effluent Monitoring Instrumentation is non-functional, the flow rate may be estimated within 4 hours of initial loss of the instrument and every 4 hours thereafter, for the duration of the effluent release. Flow rate data is necessary to calculate the amount of radioactive released via the effluent discharge. Therefore, if flow cannot be estimated, it is necessary to suspend the release of radioactive effluents monitored by the affected channel. The 4-hour Completion Time is reasonable because a significant change in flow rate over the course of an effluent release is unlikely.

A Control Room RDACS trouble alarm is received when sample flows are not within predetermined limits (among other SPING conditions). With regard to SPINGs 4 or 8, procedures require a temporary sample pump to be installed when the sample flow channel is non-functional, which may be used to meet Required Action B.1, even if the flow path is in auto-standby status. With the temporary sample pump installed, Required Action D.1 will be met should the flow path auto start. Therefore, as indicated below, Condition D is not required to be considered while the SPING 4 and 8 flow paths are idle.

S 2.0.2 is not applicable to the initial flow estimation, but may be applied to the flow estimations thereafter. Pump curves may be used to estimate flow.

C.1 and C.2

Condition C is modified by two notes. Note 1 omits this Condition from being applicable to the Waste Gas Holdup or ANO-1 Reactor Building Purge and Ventilation System Gas Activity Process or Noble Gas Activity Monitors. These monitors are addressed in Condition A. Note 2 requires the associated Required Actions and Completion Times of Condition C be applied to SPINGS 4 and 8 (Emergency Penetration Room Ventilation of ANO-1 and ANO-2, respectively) only when the pathway is in service, since noble gas activity sampling and analysis cannot be performed when the pathway is idle.

ACTIONS (continued)

C.1 and C.2 (continued)

With the exception of Waste Gas Holdup System releases or during a PURGE of the ANO-1 Reactor Building, releases may continue via an associated pathway when the Noble Gas Activity Monitor(s) is non-functional, provided a sample of the effluent is obtained once every 12 hours and analyzed within the following 24 hours. This prevents the release of unmonitored effluents to the environment. ACTIONS C.1 and C.2 are modified by a note, referring to ANO-1 TS 3.9.3 for additional ACTIONS that may be necessary if the required ANO-1 Reactor Building Purge and Ventilation System Noble Gas Activity Monitor is inoperable.

S 2.0.2 is not applicable to the initial sample and analysis, but may be applied to the sample and analysis thereafter.

D.1, D.2, and D.2

Condition D is modified by a Note which requires the associated Required Actions and Completion Times of Condition D be applied to SPINGS 4 and 8 (Emergency Penetration Room Ventilation of ANO-1 and ANO-2, respectively) only when the pathway is in service, since iodine and particulate sampling and analysis cannot be performed when the pathway is idle.

If one or more required Iodine and/or Particulate Samplers are non-functional, auxiliary sampling equipment must be established within 4 hours. The backup Iodine and Particulate cartridges must be replaced every 7 days. Following replacement, the respective cartridge must be analyzed within 48 hours. This prevents the release of unmonitored effluents to the environment.

E.1

If the Required Actions and associated Completion Times of Condition C and/or D cannot be met, then releases via the associated pathway must be suspended. This prevents the release of unmonitored effluents to the environment.

F.1

If the Required Actions and associated Completion Times of Condition A, B, and/or E cannot be met, then additional measures may be necessary to ensure continued safe operation or to reduce overall station risk. Therefore, a condition report must be initiated immediately to assess the impact on continued effluent release operations given the degraded condition.

ACTIONS (continued)

G.1

Instrumentation installed to ensure radiological monitoring of effluent releases is expected to be normally available in accordance with the design function or purpose of the equipment. Instrumentation that remains non-functional for greater than 30 days may indicate inappropriate importance placed on the equipment or over-reliance on the backup sampling method for effluent release monitoring. As an incentive to avoid either of these conditions, Radioactive Gaseous Effluent Monitoring Instrumentation that remains non-functional for more than 30 days must be included in the Radioactive Effluent Release Report submitted pursuant to TS 5.6.3 (ANO-1) or TS 6.6.3 (ANO-2). In order to ensure inclusion, Required Action G.1 requires the condition to be tracked via a condition report.

Information to be provided in the respective Radioactive Effluent Release Report should include 1) the component number and noun name, 2) the failure mode, 3) the reason for continued inoperability, and 4) the expected return to service date.

SURVEILLANCES

S 2.2.1.1

Performance of the CHANNEL CHECK every 24 hours provides reasonable assurance for prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. Where parameter comparison is not possible, the CHANNEL CHECK will continue to identify gross instrument failure such as loss of power, unexpected upscale readings, failed-low indications, etc. The CHANNEL CHECK is key in verifying that the instrumentation continues to operate properly between CHANNEL CALIBRATIONs. The Frequency is based on unit operating experience that demonstrates channel failure is rare.

This Surveillance is modified by a Note the exempts the Iodine and Particulate Samplers from a CHANNEL CHECK since these components do not have electronic features or indications.

S 2.2.1.2

A local check must be made every 7 days to verify that required Iodine Sampler cartridges and Particulate Sample filters are in place. The 7-day Frequency is reasonable because it is unlikely a cartridge or filter could be inadvertently removed from the system.

SURVEILLANCES (continued)

S 2.2.1.3 and S 2.2.1.6

A CHANNEL TEST is performed on required Gas Activity Process and Noble Gas Activity Monitors to ensure the entire channel will perform the intended functions. For the Waste Gas Holdup and ANO-2 Containment Building Purge Systems, the CHANNEL TEST demonstrates that automatic isolation of the associated pathway and Control Room alarm occur should the instrument indicate measured levels above the trip setpoint. The channel test also demonstrates that alarm occurs when any of the following conditions exist:

A. Power to the detector is lost.

B. The instrument indicates a downscale failure.

C. Instrument controls are not set in the operate mode.

Any setpoint adjustment shall be consistent with Section 3.1 of the ODCM.

Because the alarm/trip function and/or the importance of the release path, a CHANNEL TEST of the associated Gas Activity Process and Noble Gas Activity Monitors is required within 31 days prior to release via the Waste Gas Holdup or ANO-1 Reactor Building Purge and Ventilation Systems. This ensures the monitors are FUNCTIONAL within a reasonable period of time before such a release is commenced. All active pathway Gas Activity Process and Noble Gas Activity Monitors undergo a CHANNEL TEST once every 92 days. This Frequency is reasonable because each has a Control Room alarm function.

S 2.2.1.4 and S 2.2.1.5

A SOURCE CHECK provides a qualitative assessment of channel response when the channel sensor is exposed to the radioactive source. This check is performed within 14 days prior to release of effluent via the Waste Gas Holdup or ANO-1 Reactor Building Purge Systems. The 14-day restriction is reasonable because it is unlikely that the sensor will unexpectedly fail in any 14-day period. All active pathway Gas Activity Process and Noble Gas Activity Monitors must undergo a SOURCE CHECK every 31 days. This Frequency is reasonable because each has a Control Room alarm function.

When a SOURCE CHECK can be performed, it provides verification that the sensor will respond to an increase in radiation level. Note 1 of S 2.2.1.5 and the Note associated with S 2.2.1.4 does not require a SOURCE CHECK when the background radiation at the sensor is greater than the check source. This is acceptable because of the other required tests above (CHANNEL CHECK, CHANNEL TEST, and CHANNEL CALIBRATION).

SURVEILLANCES (continued)

S 2.2.1.7

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift (as required) to ensure that the instrument channel remains FUNCTIONAL between successive tests. CHANNEL CALIBRATION shall find that measurement errors and setpoint errors are within the assumptions of the setpoint calculations. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint calculations. This Frequency is justified by the assumption of at least an 18-month calibration interval to determine the magnitude of equipment drift or deviation in the setpoint calculations.

Initial CHANNEL CALIBRATION is performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration are used.

This Surveillance is modified by a Note the exempts the Iodine and Particulate Samplers from a CHANNEL CALIBRATION since these components do not have electronic features or indications.

B 2.3 RADIOACTIVE LIQUID EFFLUENTS

BASES

BACKGROUND

This Limitation 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, Appendix B, Table II. This limit provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures greater than the Section II.A design objectives of 10 CFR 50, Appendix I, to a MEMBER OF THE PUBLIC.

LIMITATION

The concentration limit for noble gases is based upon the assumption that Xe-133 is the controlling radioisotope and its maximum permissible concentration (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.

Radioactive nuclides other than dissolved or entrained noble gases must be maintained within the limits of 10 CFR 20, Appendix B, Table II, Column 2 values. The various dose limitations are conservative with regard to 10 CFR 20 requirements in order to provide a margin of safety through the use of “as low as reasonably achievable” (ALARA) practices.

Necessary portions of the LIQUID RADWASTE TREATMENT SYSTEM shall be used to reduce the radioactive materials in liquid waste prior to discharge when it is projected that the cumulative dose during a calendar quarter due to liquid effluent releases would exceed 0.18 mrem to the total body or 0.625 mrem to any organ. The provisions of this Limitation do not apply to the laundry tanks due to their incompatibility with the radwaste system.

The specified limits governing the use of appropriate portions of the LIQUID RADWASTE TREATMENT SYSTEM are a suitable fraction of the guide set forth in Section II.A of 10 CFR 50, Appendix I, for liquid effluents. The values of 0.18 mrem and 0.625 mrem are approximately 25% of the yearly design objectives on a quarterly basis. The yearly design objectives are provided in 10 CFR 50, Appendix I, Section II.

APPLICABILITY

The Limitations are required to be met at all times.

ACTIONS

Because more than one Limitation or Surveillance requirement may not be met at a given time, the ACTIONS are modified by a Note that permits separate Condition entry for each Limitation and/or Surveillance requirement that is not met.

ACTIONS (continued)

A.1 and A.2

If any Limitation L 2.3.1.a through L 2.3.1.e is not met, action must be initiated immediately to restore the parameter within limits. This could require a reduction in offsite releases scheduled for the near future or further processing of effluents prior to release. In any event, a condition report must be initiated to determine whether additional actions are necessary to permit continued operations involving radioactive liquid effluent releases given the current circumstances. In addition, corrective action must be issued to identify and track the Limitation that was exceeded for inclusion in the annual Radioactive Effluent Release Report. However, the condition need not be reported in the annual Radioactive Effluent Release Report if reported otherwise (i.e., in accordance with reporting requirements of 10 CFR 20, 10 CFR 50.72, 10 CFR 50.73, or 40 CFR 190).

B.1 and B.2

If the sampling and/or analysis requirements of S 2.3.1.1 are not met, the release must be terminated. This action prevents or minimizes the potential for an unmonitored offsite radioactive liquid release. Such release may commence or be re-initiated once the sampling and analysis requirements of S 2.3.1.1 are met. Regardless, a condition report must be initiated to determine whether additional actions are necessary to permit continued operations involving radioactive liquid effluent releases given the current circumstances. If a condition report has already been initiated relevant to this Condition, then this assessment may be performed in conjunction with that condition report; a second condition report is not required.

C.1 and C.2

This ACTION is modified a Note, limiting its applicability to only a CONTINUOUS RELEASE of secondary coolant.

With elevated dose equivalent I-131 (DEI) activity in the secondary coolant, it is prudent to modify the frequencies for obtaining and analyzing grab samples. Therefore, with secondary coolant DEI > 0.01 μCi/ml, sample frequency is modified from once every 24 hours to once every 12 hours. The analysis of the sample must be completed with 12 hours of sample acquisition. More frequent monitoring of the secondary coolant will assist in detecting further increases in activity and provide personnel better opportunity for in developing corrective action plans, as necessary.

ACTIONS (continued)

D.1

In accordance with 40 CFR 190, a variance must be received from the regulatory authority (NRC) if offsite dose to a member of the public will, or has exceeded, limits established in 40 CFR 190. Because Surveillance S 2.3.1.3 tracks the accumulated dose to members of the public over specified time periods (calendar quarter or calendar year), the dose may be projected and a determination made with regard to whether it is likely 40 CFR 190 limits will be exceeded. If 40 CFR 190 limits are projected to be exceeded, an application for a variance from the NRC must be submitted prior to the estimated date in which any 40 CFR 190 limit will be exceeded. The variance will allow continued offsite liquid and gaseous releases in excess of 40 CFR 190 limits. Note that the variance is normally expected to remain in effect until the end of the current calendar year since 40 CFR 190 limits only apply to the calculated annual dose to members of the public.

If application for variance cannot be made prior to exceeding any 40 CFR 190 limit, it may be prudent to notify the NRC by phone as soon as possible of the need for a variance, providing the expected date in which the application will be submitted. Note that the NRC may provide verbal approval for variance in situations where time is a factor.

E.1

If the Required Actions and associated Completion Times of Conditions C and/or D cannot be met or if the sampling and/or analysis requirements denoted in Surveillances S 2.3.1.1 and/or S 2.3.1.2 are not met, then additional measures may be necessary to ensure continued safe operation or to reduce overall station risk. Therefore, a condition report must be initiated immediately to assess the impact on continued effluent release operations given the requirements that are not being met.

F.1

Surveillance S 2.3.1.5 establishes required capability of various sample analyses. A given analysis must be capable of detecting respective radioactivity at a reasonably low threshold in order to ensure radioactive liquid releases to the public are carefully and accurately monitored. If the stated thresholds cannot be met, a condition report must be initiated and corrective action issued to ensure the condition is included and described in the annual Radioactive Effluent Release Report.

SURVEILLANCES

S 2.3.1.1 and S 2.3.1.2

All radioactive liquid effluent releases are required to be monitored. Because a BATCH RELEASE is of a known quantity and of finite duration, sampling of batch effluents must be performed prior to release. In addition, the sample must undergo a gamma isotopic and DEI analysis prior to the release to provide high confidence that radioactive release limits will not be exceeded. Remaining analyses may then be completed at the designated Frequency during or following the release.

For a BATCH RELEASE, a composite sample, 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, is performed. In order to ensure a representative sample, the batch shall be thoroughly mixed before the sample is obtained.

Unlike the BATCH RELEASE, a CONTINUOUS RELEASE must be monitored at a set Frequency. While gross activity monitoring is available for various release paths as is recommended by Regulatory Guide (RG) 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," such monitoring does not provide the necessary breakdown and quantification of radioactivities being discharged. Therefore, the ODCM requires grab samples and analyses of these effluents at a specified Frequency.

To be representative of the quantities and concentrations of radioactive materials in liquid effluents, a CONTINUOUS RELEASE sample must be proportional to the rate of flow of the effluent stream.

S 2.3.1.3

Limitation L 2.3.1 establishes limits on radioactive liquid concentrations discharged from the plant and the accumulative dose that may be received by a MEMBER OF THE PUBLIC as a result of such releases. In order to determine that these limits are met and being maintained, the results of analyses required by Surveillances S 2.3.1.1 and S 2.3.1.2 must be compared to the Limitation requirements on a specified Frequency. Therefore, analysis results obtained within a given 7-day period must be considered, in some cases along with previous analysis results of all liquid release over a specified period of time (calendar quarter or calendar year), to ensure limits are not exceeded.

S 2.3.1.4

In accordance with 40 CFR 190, a variance must be received from the regulatory authority (NRC) is offsite dose to a member of the public will, or has exceeded, limits established in 40 CFR 190. Because Surveillance S 2.3.1.3 tracks the accumulated dose to members of the public over specified time periods (calendar quarter or calendar year), the dose may be projected and a determination made with regard to whether it is likely 40 CFR 190 limits will be exceeded. The 31-day Frequency is acceptable because associated ODCM limits for these releases are significantly less than those described in 40 CFR 190 and, therefore, it is unlikely any 40 CFR 190 limit would be exceeded in any 31-day period.

SURVEILLANCES (continued)

S 2.3.1.5

The Lower Limit of Detection (LLD) is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal. This Surveillance contains a list of isotopes and required LLD for each. Sample analysis sensitivity must be such that radioactivities can be detected and measured at the LLD value.

It should be recognized that the LLD is an "a Priori" (before the fact) limit representing the capability of measurement system and not an "a Posteriori" (after the fact) limit for a particular measurement.

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

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

where:

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

S_b = standard deviation of the background or blank sample counts
= square root of either the background or the blank sample counts

E = counting efficiency (as counts per transformation)

V = sample size (in units of mass or volume)

T = elapsed count time

2.22 = number of transformations per minute per picocurie

Y = fractional radiochemical yield (when applicable)

λ = radioactive decay constant for the particular radionuclide

Δt = elapsed time between sample collection (or end of the sample collection period) and time of counting

Typical values of E, V, Y and Δt should be used in the calculation.

For certain mixtures of gamma emitters, it may not be possible to measure radionuclides in concentrations near their sensitivity limits when other nuclides are present in the sample in much greater concentrations. Under these circumstances, it will be more appropriate to calculate the concentration of such radionuclides using observed ratios with those radionuclides which are measurable.

SURVEILLANCES (continued)

S 2.3.1.5 (continued)

The principal gamma emitters for which the LLD limitation 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 analyses should not be reported as being present at the LLD level. When unusual circumstances result in LLD requirements not being met, the reasons shall be documented in the Radioactive Effluent Release Report as stated in Required Action F.1 of this Limitation, or the Annual Radiological Environmental Operating Report as stated in L 2.5.1, Required Action A.2.

B 2.4 RADIOACTIVE GASEOUS EFFLUENTS

BASES

BACKGROUND

This Limitation is provided to ensure that radioactive materials released in gaseous effluents from the site to unrestricted areas will be less than the limits specified in 10 CFR Part 20. This Limitation also implements the requirements of Sections II.C, III.A, and IV.A of 10 CFR 50, Appendix I.

Figure 4-2 illustrates the maximum area boundary for radioactive release calculations. For individuals who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary.

LIMITATION

Radioactive nuclides must be maintained within the limits of 10 CFR 20. The various dose rate and dose limitations are conservative with regard to 10 CFR 20 requirements in order to provide a margin of safety through the use of "as low as reasonably achievable" (ALARA) practices.

The necessary VENTILATION EXHAUST TREATMENT SYSTEMs shall be used to reduce the radioactive materials in gases prior to discharge when it is projected that the cumulative dose during a calendar quarter due to gaseous effluent releases would exceed values specified in this Limitation. The specified limits governing the use of the VENTILATION EXHAUST TREATMENT SYSTEMs are a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of 10 CFR Part 50, Appendix I, for gaseous effluents.

APPLICABILITY

The Limitations are required to be met at all times.

ACTIONS

Because more than one Limitation or Surveillance requirement may not be met at a given time, the ACTIONS are modified by a Note that permits separate Condition entry for each Limitation and/or Surveillance requirement that is not met.

ACTIONS (continued)

A.1 and A.2

If any Limitation L 2.4.1.a through L 2.4.1.d is not met, action must be initiated immediately to restore the parameter within limits. This could require a reduction in offsite releases scheduled for the near future or further processing of effluents prior to release. In any event, a condition report must be initiated to determine whether additional actions are necessary to permit continued operations involving radioactive gaseous effluent releases given the current circumstances. In addition, corrective action must be issued to identify and track the Limitation that was exceeded for inclusion in the annual Radioactive Effluent Release Report. However, the condition need not be reported in the annual Radioactive Effluent Release Report if reported otherwise (i.e., in accordance with reporting requirements of 10 CFR 20, 10 CFR 50.72, 10 CFR 50.73, or 40 CFR 190).

B.1 and B.2

If the sampling and/or analysis requirements of S 2.4.1.1 are not met, the release must be terminated. This action prevents or minimizes the potential for an unmonitored offsite radioactive liquid release. Such release may commence or be re-initiated once the sampling and analysis requirements of S 2.4.1.1 are met. Regardless, a condition report must be initiated to determine whether additional actions are necessary to permit continued operations involving radioactive liquid effluent releases given the current circumstances. If a condition report has already been initiated relevant to this Condition, then this assessment may be performed in conjunction with that condition report; a second Condition Report is not required.

C.1

In accordance with 40 CFR 190, a variance must be received from the regulatory authority (NRC) if offsite dose to a member of the public will, or has exceeded, limits established in 40 CFR 190. Because Surveillance S 2.4.1.3 tracks the accumulated dose to members of the public over specified time periods (calendar quarter or calendar year), the dose may be projected and a determination made with regard to whether it is likely 40 CFR 190 limits will be exceeded. If 40 CFR 190 limits are projected to be exceeded, an application for a variance from the NRC must be submitted prior to the estimated date in which any 40 CFR 190 limit will be exceeded. The variance will allow continued offsite liquid and gaseous releases in excess of 40 CFR 190 limits. Note that the variance is normally expected to remain in effect until the end of the current calendar year since 40 CFR 190 limits only apply to the calculated annual dose to members of the public.

If application for variance cannot be made prior to exceeding any 40 CFR 190 limit, it may be prudent to notify the NRC by phone as soon as possible of the need for a variance, providing the expected date in which the application will be submitted. Note that the NRC may provide verbal approval for variance in situations where time is a factor.

ACTIONS (continued)

D.1

If the Required Actions and associated Completion Times of Condition C cannot be met or if the sampling and/or analysis requirements denoted in Surveillances S 2.4.1.2 are not met, then additional measures may be necessary to ensure continued safe operation or to reduce overall station risk. Therefore, a condition report must be initiated immediately to assess the impact on continued effluent release operations given the requirements that are not being met.

E.1

Surveillance S 2.4.1.5 establishes required capability of various sample analyses. A given analysis must be capable of detecting respective radioactivity at a reasonably low threshold in order to ensure radioactive gaseous releases to the public are carefully and accurately monitored. If the stated thresholds cannot be met, a condition report must be initiated and corrective action issued to ensure the condition is included and described in the annual Radioactive Effluent Release Report.

SURVEILLANCES

Continuous gaseous release paths are monitored by instrumentation denoted in Limitation L 2.2.1. Limitation L 2.2.1 provides Required Actions and Completion Times for circumstances when required instrumentation is out of service. Therefore, the Surveillances associated with this Limitation (L 2.4.1) envelop only required grab, charcoal, and particulate samples necessary to verify 10 CFR 20 limits will be met.

The Surveillance Limitations implement the requirements in 10 CFR 50, Appendix I, Section III.A, that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in this manual for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The equations in this manual provided for determining the air doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

The release rate limitations for iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man in the areas at or beyond the site boundary. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

SURVEILLANCES (continued)

S 2.4.1.1 and S 2.4.1.2

All radioactive gaseous effluent releases are required to be monitored. Because a Waste Gas Holdup Tank or Reactor (Containment) Building Purge release is of a known (or estimated) quantity and of finite duration, sampling of these effluents must be performed prior to release. In addition, the sample must be analyzed for principal gamma emitters and tritium prior to the release in order to provide high confidence that radioactive release limits will not be exceeded.

S 2.4.1.3

To meet the intent of the continuous monitoring requirement for noble gases, the noble gas activity from each SPING operating on an activity flow path must be recorded at least once every 24 hours. The current, highest, and average activity recorded from a particular SPING over the required grab sample period designated in other Surveillances associated with this Limitation are used to scale the noble gas and tritium activity obtained from the associated grab sample. The final resulting activity is used, in part, to support completion of S 2.4.1.4 and S 2.4.1.5 below.

S 2.4.1.4

Limitation L 2.4.1 establishes limits on radioactive gases discharged from the plant and the dose rates and accumulative dose that may be received by a MEMBER OF THE PUBLIC as a result of such releases. In order to determine that these limits are met and being maintained, the results of analyses required by Surveillances S 2.4.1.1 and S 2.4.1.2, as adjusted by readings taken in accordance with S 2.4.1.3 as appropriate must be compared to the Limitation requirements on a specified Frequency. Therefore, analysis results obtained within a given 31-day period must be considered, in some cases along with previous analysis results of all gaseous releases over a specified period of time (calendar quarter or calendar year), to ensure limits are not exceeded.

The ratio of the sample flow rate to the sampled stream flow rate must be known for the time period covered by each dose or dose rate calculation made in accordance with this Limitation.

S 2.4.1.5

In accordance with 40 CFR 190, a variance must be received from the regulatory authority (NRC) is offsite dose to a member of the public will, or has exceeded, limits established in 40 CFR 190. Because Surveillance S 2.4.1.3 tracks the accumulated dose to members of the public over specified time periods (calendar quarter or calendar year), the dose may be projected and a determination made with regard to whether it is likely 40 CFR 190 limits will be exceeded. The 31-day Frequency is acceptable because associated ODCM limits for these releases are significantly less than those described in 40 CFR 190 and, therefore, it is unlikely any 40 CFR 190 limit would be exceeded in any 31-day period.

SURVEILLANCES (continued)

S 2.4.1.6

The Lower Limit of Detection (LLD) is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal. This Surveillance contains a list of isotopes and required LLD for each. Sample analysis sensitivity must be such that radioactivies can be detected and measured at the LLD value. The Surveillance also contains the LLD for the Noble Gas Monitors associated with Limitation 2.2.1.

For an explanation of the LLD calculation, refer to the S 2.3.1.5 Bases.

For certain radionuclides with low gamma yield or low energies, or for certain radionuclides 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 > 10% of the MPC value specified in 10 CFR 20, Appendix B, Table II, Column 1.

The principal gamma emitters for which the LLD limitation will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are 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 result in LLD's higher than required, the reasons shall be documented in the Radioactive Effluent Release Report.

B 2.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

B 2.5.1 Environmental Sampling

BASES

BACKGROUND

The ODCM includes, in tables and figures, specific parameters of distance and direction from the centerline of one reactor, and additional description where pertinent, for each sample location required by the Radiological Environmental Monitoring Program. NUREG-0133, "Preparation of Radiological Technical Specifications for Nuclear Power Plants," October 1978, and Radiological Assessment Branch Technical Position (BTP), Revision 1, November 1979, provide guidance with regard to environmental sampling.

With regard to the aforementioned BTP, one airborne sample location should be from the vicinity of a community having the highest calculated annual average ground-level D/Q. Community as defined by Webster's dictionary is people with common interests living in a particular area; broadly: the area itself. The local municipalities of London, Russellville, and Dardanelle are all part of the River Valley community located near Dardanelle Lake and the Arkansas River. The grouping of houses that reside within WSW (highest D/Q) sector are located in London which is part of the River Valley community. Air Station #2 per the above mentioned NRC BTP meets the requirements of being located within the highest D/Q and also in the vicinity of a community. Reference CR-ANO-C-2016-2732.

The approximate locations of selected sample sites are shown on ODCM Figures 4-1, 4-1A, and 4-1B for illustrative purposes. ODCM Table 4-1 lists the approximate distances and directions of the sample stations from the plant.

"D/Q" refers to a radiological deposition rate considering prevalent winds around the site and is used to determine natural settling of effluents from the atmosphere.

LIMITATION

This Limitation specifies the sample locations and distances, sample analysis type and frequency, and parameters to be sampled as part of the Radiological Environmental Monitoring Program.

The Limitation is modified by a Note that permits other instrumentation to be used in place of, or in addition to, integrating dosimeters for measuring and recording dose rate continuously. For the purposes of this Limitation, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

APPLICABILITY

The Limitations are required to be met at all times.

ACTIONS

Because more than one Limitation or Surveillance requirement may not be met at a given time, the ACTIONS are modified by a Note that permits separate Condition entry for each Limitation and/or Surveillance requirement that is not met.

A.1 and A.2

Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunctions, every effort shall be made to complete corrective action before the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report.

This ACTION lists several items that would result in the intent of the Radiological Environmental Monitoring Program not being met. In addition, this ACTION provides guidance for conditions where radionuclides other than those listed in Table 2.5-2 could result in a noteworthy dose to a MEMBER OF THE PUBLIC. Immediate action is required to restore conditions needed to meet the intent of the Radiological Environmental Monitoring Program. All deviations from the Limitations and Surveillances required to meet the intent of the Radiological Environmental Monitoring Program must be reported in the Annual Radiological Environmental Operating Report. However, the condition need not be reported in the Annual Radiological Environmental Operating Report if reported otherwise (i.e., in accordance with reporting requirements of 10 CFR 20, 10 CFR 50.72, 10 CFR 50.73, or 40 CFR 190).

With the level of radioactivity as the result of plant effluents in an environmental sampling medium at one or more required locations exceeding the limits of Table 2.5-2 when averaged over any calendar quarter, the condition must be reported in accordance with Required Action A.2. The report should include an evaluation of any release conditions, environmental factors or other aspects which caused the limits to be exceeded, and define the actions taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC will remain less than the calendar year limits of Limitations L 2.3.1 and L 2.4.1. When more than one of the radionuclides in Table 2.5-2 is detected in the sampling medium, the information shall be included in the report if:

$$\frac{\text{Concentration 1}}{\text{Reporting Level 1}} + \frac{\text{Concentration 2}}{\text{Reporting Level 2}} + \frac{\textit{etc.}}{\textit{etc.}} \geq 1.0$$

B.1

In addition to the requirements of Required Actions A.1 and A.2, a new location must be identified and added to the Radiological Environmental Monitoring Program within 30 days when required samples cannot be obtained from designated locations. Note that broad leaf samples are only required when milk samples are unavailable, pursuant to S 2.5.1.8.

ACTIONS (continued)

B.1 (continued)

The specific locations from which samples were unavailable may then be deleted from the monitoring program. The cause(s) of the unavailability of samples the new location(s) for obtaining replacement samples shall be identified in next Annual Radiological Environmental Operating Report. The report shall also include a revised Table 4-1 reflecting the new location(s).

SURVEILLANCES

S 2.5.1.1 through S 2.5.1.8

These Surveillances ensure samples are collected and analyzed at specified frequencies of the parameters, and from the locations, designated in Limitation L 2.5.1. The approximate locations of selected sample sites are shown on ODCM Figures 4-1, 4-1A, and 4-1B for illustrative purposes. ODCM Table 4-1 lists the approximate distances and directions of the sample stations from the plant.

Note that the gross beta analysis of required particulate samplers should not be performed within the first 24 hours following particulate filter change. This is to allow for radon and thoron daughter decay. If it is discovered that the particulate gross beta activity is more than 10 times the yearly mean of control samples for any medium, consideration should be given to performing a gamma isotopic analysis of the individual particulate samples. Also note that particulate samples may need to be collected more frequently than the specified 14-day Frequency due to dust or other accumulation of matter.

Gamma isotopic analysis includes the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

S 2.5.1.9

The Lower Limit of Detection (LLD) is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a “real” signal. Table 2.5-1 contains a list of isotopes and required LLD for each. Sample analysis sensitivity must be such that radioactivities can be detected and measured at the LLD value.

For an explanation of the LLD calculation, refer to the S 2.3.1.5 Bases.

S 2.5.1.10

With the level of radioactivity as the result of plant effluents in an environmental sampling medium at one or more required locations exceeding the limits of Table 2.5-2 when averaged over any calendar quarter, the condition must be reported in accordance with Required Action A.2.

B 2.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

B 2.5.2 Land Use Census

BASES

BACKGROUND

The surveys required by this Limitation ensure that changes in environmental conditions as they relate to radioactive effluent releases from the site are identified and accounted for in the overall dose commitment to the public.

LIMITATION

This Limitation ensures changes in the use of unrestricted areas are identified and that modifications are subsequently included in the Radiological Environmental Monitoring Program. The census satisfies 10 CFR 50, Appendix I, Section IV.B.3.

Restricting the census to gardens of > 500 ft^2 provides assurance that significant exposure pathway 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 RG 1.109 for consumption by a child. This minimum garden size was determined assuming 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/m^2.

The Limitation is modified by a Note that permits broad leaf vegetation sampling to be performed at the site boundary in the directional sector having the highest D/Q in lieu of performing a garden census. "D/Q" refers to a radiological deposition rate considering prevalent winds around the site and is used to determine natural settling of effluents from the atmosphere.

APPLICABILITY

The Limitations are required to be met at all times.

ACTIONS

Because more than one new sample location may be identified during a given census, the ACTIONS are modified by a Note permit separate Condition entry for each new location identified.

ACTIONS (continued)

A.1, A.2.1, and A.2.2

When new locations are discovered that indicate higher radioactivity levels than current locations being sample pursuant to Limitation L 2.5.1 or if radioactivity levels at a new location are projected to exceed 40 CFR 190 limits (with regard to I-131, H-3, and particulate sources), a condition report must be immediately initiated. Initiating a condition report will ensure reporting criteria is evaluated for the given condition. Regardless of any other report, the new location must be included in the next Annual Radiological Environmental Operating Report.

In addition to the requirements of Required Action A.1, the new location must be added to the Radiological Environmental Monitoring Program within 30 days. Following October 31 of the year in which the census is taken, the old sample location in this same pathway may be deleted from the Radiological Environmental Monitoring Program. This is expected to be performed within 90 days following the October 31 limit.

SURVEILLANCES

S 2.5.2.1 through S 2.5.2.2

The land use census must be performed every 24 months and between the dates of June 1 and October 1 of the given year. The results of the census must be reported in the next Annual Radiological Environmental Operating Report.

The Surveillance requirements are modified by a Note that prevents the use of S 2.0.2. Therefore, the 25% Frequency extension associated with S 2.0.2 cannot be applied to the Surveillances associated with this Limitation. This is because the Frequencies are associated with strict performance and reporting dates which cannot be exceeded.

B 2.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

B 2.5.3 Interlaboratory Comparison Program

BASES

BACKGROUND

This Limitation refers to the off-site radiochemistry laboratory. The Limitation provides independent checks on the accuracy of the measurements of radioactive material in environmental samples.

LIMITATION

The requirement for participation in an 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.

APPLICABILITY

The Limitations are required to be met at all times.

ACTION

A.1

Failure to meet the requirements of the Interlaboratory Comparison Program requires initiating a condition report to ensure the circumstances are included in the next Annual Radiological Environmental Operating Report.

SURVEILLANCE

S 2.5.3.1

The results of the Interlaboratory Comparison Program analyses must be reported in the next Annual Radiological Environmental Operating Report.

ENCLOSURE, ATTACHMENT 2

0CAN042202

PROCESS CONTROL PROGRAM

PROCESS CONTROL PROGRAM

Procedure Contains NMM ECH eB REFLIB Forms: YES ☐ NO ☒
Procedure Revision Type: New ☐ **NON-Editorial** ☒ **Editorial** ☐ **TC** ☐ **Cancellation** ☐

HQN Effective Date	Procedure Owner: Title: Site:	Donnie Marvel Manager, RP ANO	Governance Owner: Title: Site:	Reid Tagliamonte Manager, Fleet RP HQN
8/27/15				

Site	Site Procedure Champion	Title
ANO	Donnie Marvel	Manager, RP
BRP	N/A	N/A
CNS	Chris Sunderman	Manager, RP
GGNS	Roy Miller	Manager, RP
IPEC	Frank Mitchell	Manager, RP
JAF	Robert Heath	Manager, RP
PLP	David Nestle	Manager, RP
PNPS	Alan Zelie	Manager, RP
RBS	Shannon Peterkin	Manager, RP
W3	Daniel Frey	Manager, RP
HQN	Reid Tagliamonte	Manager, Fleet RP

For site implementation dates see ECH eB REFLIB using site tree view (Navigation panel).

Site and NMM Procedures Canceled or Superseded By This Revision

None

Process Applicability Exclusion: All Sites: ☐
Specific Sites: ANO ☐ BRP ☐ CNS ☐ GGNS ☒ IPEC ☐ JAF ☐ PLP ☐ PNPS ☐ RBS ☐ W3 ☐

Change Statement

The primary purpose of this revision is to incorporate GGNS Temp Change in response to CR-GGN-2015-1277. Specifically:

- Step 5.1[1](b) added the words "owned by Entergy"
- Added new step 5.9[2] (same as step 5.1[1](b))

Other changes:

- Removed VY from coversheet and deleted step 5.8[4](e) as fleet procedures no longer apply to VY.
- Reformatted table in section 8 for compliance with EN-AD-101-01, updated the table and deleted VY entries from the table. Updated cross references to section 8 within the body of the procedure.
- Deleted reference to VY commitments from step 5.8[3]

Associated PRHQN #: 2015-00273 | **Procedure Writer:** Ron Schwartz

Contains Proprietary Information: YES ☐ NO ☒

TABLE OF CONTENTS

1.0 PURPOSE

The Process Control Program (PCP) requires formulas, sampling, analyses, test and determinations to be made to ensure that the processing and packing of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste. The scope of a PCP is to assure that radioactive waste will be handled, shipped, and disposed of in a safe manner in accordance with approved site or vendor procedures, whichever is applicable. [GGNS UFSAR, Chapter 16B.1 / TRM – 7.6.3.8 paragraph 1]

1.1 The purpose of this document is to provide a description of the solid radioactive waste Process Control Program (PCP) at all the Entergy fleet sites. The PCP describes the methods used for processing, classification and packaging low-level wet radioactive waste into a form acceptable for interim on-site storage, shipping and disposal, in accordance with 10 CFR Part 61 and current disposal site criteria.

1.2 To ensure the safe operation of the solid radwaste system, the solid radwaste system will be used in accordance with this Process Control Program to process radioactive wastes to meet interim on-site storage, shipping and burial ground requirements.

1.3 This document addresses the process control program in the context of disposal criteria, on-site processing and vendor processing requirements.

1.4 The Process Control Program implements the requirements of 10CFR50.36a and General Design Criteria 60 of Appendix A to 10CFR Part 50. The process parameters included in the Process Control Program may include but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

1.5 This document does NOT address the requirements for 10CFR Part 61.56 (waste characteristics) for material sent to intermediate processors, because the final treatment and packaging is performed at the vendor facilities.

2.0 REFERENCES

[1] EN-QV-104, “Entergy Quality Assurance Program Manual Control”

[2] Title 49, Code of Federal Regulations

[3] Title 10, Code of Federal Regulations, Part 20

2.0 continued

[4] Title 10, Code of Federal Regulations, Part 61

[5] Title 10, Code of Federal Regulations, Part 71, Appendix H [QAPM, Section A.1.c]

[6] Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification, 11 May 1983

[7] Disposal Site Criteria and License

[8] Waste Processor Acceptance Criteria

[9] EN-LI-100, "Process Applicability Determination"

[10] NRC Information and Enforcement Bulletins

- NRC Information Notice 79-19: Packaging of Low-Level Radioactive Waste for Transport and Burial.
- NRC Information Notice 80-24: Low-Level Radioactive Waste Burial Criteria.
- NRC Information Notice 80-32: Clarification of Certain Requirements for Exclusive-Use Shipments of Radioactive Materials.
- NRC Information Notice 80-32, Rev. 1: Clarification of Certain Requirements for Exclusive-Use Shipments of Radioactive Materials.
- NRC Information Notice 83-05: Obtaining Approval for Disposing of Very-Low-Level Radioactive Waste - 10CFR Section 20.302.
- NRC Information Notice 83-10: Clarification of Several Aspects Relating to Use of NRC-Certified Transport Packages.
- NRC Information Notice 83-33: Non-Representative Sampling of Contaminated Oil.
- NRC Information Notice 84-50: Clarification of Scope of Quality Assurance Programs for Transport Packages Pursuant to 10CFR 50 Appendix B.
- NRC Information Notice 84-72: Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation.
- NRC Information Notice 85-92: Surveys of Wastes Before Disposal from Nuclear Reactor Facilities.
- NRC Information Notice 86-20: Low-Level Radioactive Waste Scaling Factors, 10CFR 61.
- NRC Information Notice 86-90: Requests to Dispose of Very Low-Level Radioactive Waste Pursuant 10CFR 20.302

2.0[10], continued

- NRC Information Notice 87-03: Segregation of Hazardous and Low-Level Radioactive Wastes
- NRC Information Notice 87-07: Quality Control of On-Site Dewatering/ Solidification Operations by Outside Contractors

[11] NRC Information and Enforcement Bulletins (continued)

- NRC Information Notice 89-27: Limitations on the Use of Waste Forms and High Integrity Containers for the Disposal of Low-Level Radioactive Waste
- NRC Information Notice 92-62: Emergency Response Information Requirements for Radioactive Material Shipments
- NRC Information Notice 92-72: Employee Training and Shipper Registration Requirements for Transporting Radioactive Materials
- NRC 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”.

[12] Nureg-0800 Standard Review Plan Section 11.4 Revision 2, Solid Waste Management Systems.

[13] NRC Waste Form Technical Position, Revision 1 Jan 24 1991.

[14] NRC SECY 94-198 Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste.

[15] EPRI TR-106925 Rev-1, Interim On-Site Storage of Low Level Waste: Guidelines for Extended Storage - October1996

[16] NRC Branch Technical Position On Concentration Averaging And Encapsulation Jan 17 1995

[17] Commitment Documents (U-2 and U-3)

- IPN-99-079, “Supplement to Proposed Changes to Technical Specifications Incorporating Recommendations of Generic Letter 89-01 and the Revised 10 CFR Part 20 and 10 CFR Part 50.36a.
- Appendix B Technical Specifications, Section 4.5 [IP, RECS ODCM Part 1]

3.0 DEFINITIONS

[1] **Batch** – A quantity of waste to be processed having essentially consistent physical and chemical characteristics as determined through past experience or system operation knowledge by the Radwaste Shipping Specialist. A batch could be a waste tank, several waste tanks grouped together or a designated time period such as between outages as with the DAW waste stream. An isolated quantity of feed waste to be processed having essentially constant physical and chemical characteristics. (The addition or removal of water will not be considered to create a new batch).

[2] **Certificate of Compliance** - Document issued by the USNRC regulating use of a NRC licensed cask or issued by (SCDHEC) South Carolina Department of Health and Environmental Conservation regulating a High Integrity Container.

[3] **Chelating Agents** - EDTA, DTPA, hydroxy-carboxylic acids, citric acid, carbolic acid and glucinic acid.

[4] **Compaction** - The process of volume reducing solid waste by applying external pressure.

[5] **Confirmatory Analysis** - The practice of verifying that gross radioactivity measurements using MCA are reasonably consistent with independent laboratory sample data.

[6] **Dewatered Waste** - Wet waste that has been processed by means other than solidification, encapsulation, or absorption to meet the free standing liquid requirements of 10CFR Part 61.56 (a)(3) and (b)(2).

[7] **De-watering** - The removal of water or liquid from a waste form, usually by gravity or pumping.

[8] **Dilution Factor** - The RADMAN computer code factor to account for the non-radioactive binder added to the waste stream in the final product when waste is solidified.

[9] **Dry Waste** - Radioactive waste which exist primarily in a non-liquid form and includes such items as dry materials, metals, resins, filter media and sludges.

[10] **Encapsulation** - Encapsulation is a means of providing stability for certain types of waste by surrounding the waste by an appropriate encapsulation media.

[11] **Gamma-Spectral-Analysis** - Also known as IG, MCA, Ge/Li and gamma spectroscopy.

[12] **Gross Radioactivity Measurements** - More commonly known as dose to curie conversion for packaged waste characterization and classification.

3.0 continued

[13] **Homogeneous** - Of the same kind or nature; essentially alike. Most Volumetric waste streams are considered homogeneous for purposes of waste classification.

[14] **Incineration** – The process of burning a combustible material to reduce its volume and yield an ash residue.

[15] **Liquid Waste** - Radioactive waste that exist primarily in a liquid form and is contained in other than installed plant systems, to include such items as oil, EHC fluid, and other liquids. This waste is normally processed off-site.

[16] **Low-Level Radioactive Waste (LLW)** - Those wastes containing source, special nuclear, or by-product material that are acceptable for disposal in a land disposal facility. For the purposes of this definition, low-level radioactive waste has the same meaning as in the Low-Level Waste Policy Act, that is, radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined in section 11e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste).

[17] **Measurement of Specific Radionuclides** - More commonly known as direct sample or container sample using MCA data for packaged waste characterization and classification.

[18] **Operable** - A system, subsystem, train, component or device SHALL be OPERABLE or have OPERABILITY when it is capable of performing its specified functions(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

[19] **Prequalification Program** - The testing program implemented to demonstrate that the proposed method of wet waste processing will result in a waste form acceptable to the land disposal facility and the NRC.

[20] **Processing** - Changing, modifying, and/or packaging radioactive waste into a form that is acceptable to a disposal facility.

[21] **Quality Assurance/Quality Control** - As used in this document, "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material structure, component, or system to predetermined requirements.

3.0, continued

[22] **Reportable Quantity Radionuclides (RQ)** - Any radionuclide listed in column (1) of Table 2 of 49CFR Part 172.101 which is present in quantities as listed in column (3) of Table 2 of 49CFR Part 172.101.

[23] **Sampling Plan** - A program to ensure that representative samples from the feed waste and the final waste form are obtained and tested for conformance with parameters stated in the PCP and waste form acceptance criteria.

[24] **Scaling Factor** - A dimensionless number which relates the concentration of an easy to measure radionuclide (gamma emitter) to one which is difficult to measure (beta and/or alpha emitters).

[25] **Significant Quantity** - For purposes of waste classification all the following radionuclide values SHALL be considered significant and must be reported on the disposal manifest.

- Any value (real or LLD) for radionuclides listed in Appendix G to 10CFR20 (H-3, C-14, I-129, Tc-99).
- Greater than or equal to 1 percent of the concentration limits as listed in 10CFR Part 61.55 Table 1.
- Greater than or equal to 1 percent of the Class A concentration limits listed in 10CFR Part 61.55 Table 2.
- Greater than or equal to 1 percent of the total activity.
- Greater than or equal to 1 percent of the Reportable Quantity limits listed on 49CFR Part 172.101 Table 2.

[26] **Solidification** - The conversion of wet waste into a free-standing monolith by the addition of an agent so that the waste meets the stability and free-standing liquid requirements of the disposal site.

[27] **Special Radionuclides** - The RADMAN computer code term for radionuclides listed in Appendix G to 10CFR20 (i.e., H-3, C-14, I-129 & Tc-99)

[28] **Stability** – Structural stability per 10CFR61.2, Waste Form Technical Position, and Waste Form Technical Position Revision 1. This can be provided by the waste form, or by placing the waste in a disposal container or structure that provides stability after disposal. Stability requires that the waste form maintain its structural integrity under the expected disposal conditions.

3.0, continued

[29] **Training** - A systematic program that ensures a person has knowledge of hazardous materials and hazardous materials regulations.

[30] **Type A Package** - Is the packaging together with its radioactive contents limited to A1 or A2 as appropriate that meets the requirements of 49CFR Part 173.410 and Part 173.412, and is designed to retain the integrity of containment and shielding under normal conditions of transport as demonstrated by the tests set forth in 49CFR Part 173.465 or Part 173.466 as appropriate.

[31] **Type B Package** - Is the packaging together with its radioactive contents that is designed to retain the integrity of containment and shielding when subjected to the normal conditions of transport and hypothetical accident test conditions set forth in 10CFR Part 71.

[32] **Volume Reduction** – any process that reduces the volume of waste. This includes but is not limited to, compaction and incineration.

[33] **Waste Container** - A vessel of any shape, size, and composition used to contain the waste media.

[34] **Waste Form** - Waste in a waste container acceptable for disposal at a licensed disposal facility.

[35] **Waste Stream** - A Plant specific and constant source of waste with a distinct radionuclide content and distribution.

[36] **Waste Type** – A single packaging configuration and waste form tied to a specific waste stream.

4.0 RESPONSIBILITIES

[1] The **Vice President Operations Support (VPOS)** is responsible for the implementation of this procedure.

[2] Each site **Senior Nuclear Executive (SNE)** is responsible for ensuring that necessary site staff implements this procedure.

[3] The **Low Level RadWaste (LLRW) Focus Group** is responsible for evaluating and recommending changes and revisions to this procedure.

4.0, continued

[4] Each site **RP Department – Radwaste Supervisor / Specialist** (title may vary at the site's respectively) has the overall responsibility for implementing the PCP and is responsible for processing and transportation is tasked with the day-to-day responsibilities for the following:

- Implementing the requirements of this document.
- Ensuring that radioactive waste is characterized and classified in accordance with 10CFR Part 61.55 and Part 61.56.
- Ensuring that radioactive waste is characterized and classified in accordance with volume reduction facility and disposal site licenses and other requirements.
- Designating other approved procedures (if required) to be implemented in the packaging of any specific batch of waste.
- Providing a designated regulatory point of contact between the Plant and the NRC, volume reduction facility or disposal site.
- Maintaining records of on-site and off-site waste stream sample analysis and Plant evaluations.
- Suspending shipments of defectively processed or defectively packaged radioactive wastes from the site when the provisions of this process control program are not satisfied.

5.0 DETAILS

An isotopic analysis SHALL be performed on every batch for each waste stream so that the waste can be classified in accordance with 10CFR61. The isotopic and curie content of each shipping container SHALL be determined in accordance with 49CFR packaging requirements. The total activity in the container may be determined by either isotopic analysis or by dose-rate-to-curie conversion.

5.1. Precautions and Limitations

[1] Precautions

(a) Radioactive materials SHALL be handled in accordance with applicable radiation protection procedures.

(b) All radioactive waste owned by Entergy processed on-site **OR** off-site by vendors must be processed or packaged to meet the minimum requirements listed in 10CFR Part 61.56 (a) (1) through (8).

5.1[1], continued

(c) If the provisions of the Process Control Program are not satisfied, suspend shipment of the defectively processed or defectively packaged waste from the site. Shipment may be accomplished when the waste is processed / packaged in accordance with the Process Control Program.

(d) The generation of combustible gases is dependent on the waste form, radioactive concentration and accumulated dose in the waste. Changes to organic inputs (e.g. oil) to waste stream may change biogas generation rates.

[2] Limitations

(a) Only qualified personnel will characterize OR package radioactive waste OR radioactive materials for transportation or disposal.

(b) All site personnel that have any involvement with radioactive waste management computer software SHALL be familiar with its functions, operation and maintenance.

5.2. Waste Management Practices

[1] Waste processing methods include the following:

(a) Present and planned practice is NOT to solidify or encapsulate any waste streams.

(b) Waste being shipped directly for burial in a HIC (High Integrity Container) is dewatered to less than 1 percent by volume prior to shipment.

(c) Waste being shipped directly for burial in a container other than a HIC is dewatered to less than 0.5 percent by volume prior to shipment.

(d) IF solidification is required in the future, THEN at least one representative test specimen from at least every 10th batch of each type of radioactive waste will be checked to verify solidification.

(1) IF any specimen fails to verify solidification, THEN the solidification of the batch under test SHALL be suspended until such time as additional test specimens can be obtained, alternative solidification parameters can be determined, and a subsequent test verifies solidification. If alternative parameters are determined, the subsequent tests shall be verified using the alternative parameters determined.

5.2[1](d), continued

(2) IF the initial test specimen from a batch of waste fails to verify solidification, THEN provide for the collection and testing of representative test specimens from each consecutive batch of the same type of waste until at least 3 consecutive initial test specimens demonstrates solidification. The process SHALL be modified as required to assure solidification of subsequent batches of waste.

[2] Operation and maintenance of dewatering systems and equipment include the following:

(a) Present and planned practice is to utilize plant personnel supplemented by vendor personnel or contracted vendor personnel, to operate AND maintain dewatering systems and equipment (as needed to meet disposal site requirements).

(b) All disposal liners are manufactured by and purchased from QA-approved vendors.

[3] ALARA considerations are addressed in all phases of the processes involving handling, packaging AND transfer of any type OR form of radioactive waste (dewatered or dry). Resin, charcoal media, spent filter cartridges AND sludges are typically processed within shields. Sluiceable demineralizers are shielded when in service. Radiation exposure and other health physics requirements are controlled by the issuance of a Radiation Work Permit (RWP) for each task.

5.3. Waste Stream Sampling Methods and Frequency

[1] The following general requirements apply to Plant waste stream sampling:

(a) Treat each waste stream separately for classification purposes.

(b) Ensure samples are representative of or can be correlated to the final waste form.

(c) Determine the density for each new waste stream initially or as needed (not applicable for DAW and filters).

(d) Perform an in-house analysis for gamma-emitting radionuclides for each sample sent to an independent laboratory.

(e) Periodically perform in-house analysis for gamma emitting radionuclides for comparison to the current data base values for gamma emitters. (The current database is usually based on the most recent independent laboratory results.)

(f) Resolve any discrepancies between in-house results AND the independent laboratory results for the same or replicate sample as soon as possible.

(g) Maintain records of on-site and off-site waste stream sample analysis and evaluations.

5.3, continued

[2] When required, waste stream samples should be analyzed, re-evaluated and if necessary, shipped to a vendor laboratory for additional analysis. The same is true when there is a reason to believe that an equipment or process change has significantly altered the previously determined scaling factors by a factor of 10.

Specific examples include but are not limited to:

- Changes in oxidation reduction methods such as zinc, injection, hydrogen water chemistry,
- Changes in purification methods including media specialization, media distribution, ion/cation ratios,
- Changes in fuel performance criteria including fuel leaks
- Other changes in reactor coolant chemistry.
- Sustained, unexplained, changes in the routinely monitored Beta/Alpha ratios, as determined by Radiation Protection,
- When there is an extended reactor shutdown (> 90 days).
- When there are changes to liquid waste processing, such as bypassing filters, utilizing filters or a change in ion exchange media.
- When there are changes to the waste stream that could change the biogas generation rate.

[3] The following requirements apply to infrequent or abnormal waste types:

(a) Infrequent **OR** abnormal waste types that may be generated must be evaluated on a case-by-case basis.

(b) The RP Department Supervisor / Specialist responsible for processing AND shipping will determine if the waste can be correlated to an existing waste stream.

(c) IF the radioactive waste cannot be correlated to an existing waste stream, THEN the RP Department Supervisor / Specialist responsible for processing and shipping SHALL determine specific off-site sampling and analysis requirements necessary to properly classify the material.

[4] Specific sampling methods and data evaluation criteria are detailed in EN-RW-104 for specific waste streams.

5.4. Waste Classification

[1] General requirements for scaling factors include the following:

(a) The Plant has established an inferential measurement program whereby concentrations of radionuclides which cannot be readily measured are estimated through ratio-ing with radionuclides which can be readily measured.

(b) Scaling factor relationships are developed on a waste stream-specific basis. These relationships are periodically revised to reflect current independent lab data from direct measurement of samples. The scaling factor relationships currently used by the sites are as follows:

- Hard to detect ACTIVATION product radionuclides and C-14 are estimated by using scaling factors with measured Co-60 activities.
- Hard to detect FISSION product radionuclides and H-3, Tc-99 and I-129 are estimated by using scaling factors with measured Cs-137 activities.
- Hard to detect TRANSURANIC radionuclides are estimated by using scaling factors with measured Ce-144 activities. Where Ce-144 cannot be readily measured, transuranics are estimated by using scaling factors with measured Cs-137 activities. Second order scaling of transuranics is acceptable when Cs-137 and Ce-144 are not readily measurable.

[2] General requirements for the determination of total activity and radionuclide concentrations include the following:

(a) The activity for the waste streams is estimated by using either Gross Radioactivity Measurement OR Direct Measurement of Radionuclides. Current specific practices are as follows:

- DAW - Gross radioactivity measurement in conjunction with the RADMAN computer codes, other approved computer codes or hand calculation.
- Filters - Gross radioactivity measurement in conjunction with the FILTRK computer code, other approved computer codes or hand calculation.
- All Other Waste Streams - Direct measurement of radionuclides in conjunction with the RADMAN computer codes, other approved computer codes or hand calculation.

(b) Determination of the NRC waste classification is performed by comparing the measured or calculated concentrations of significant radionuclides in the final waste form to those listed in 10CFR Part 61.55.

5.5. Quality Control

[1] The RADMAN computer code provides a mechanism to assist the Plant in conducting a quality control program in accordance with the waste classification requirements listed in 10CFR Part 61.55. All waste stream sample data changes are written to a computer data file for future review and reference.

[2] Audits and Management Review includes the following:

(a) Appendix G to 10CFR20 requires conduct of a QC program which must include management review of audits.

(b) Management audits of the Plant Sampling and Classification Program SHALL be periodically performed to verify the adequacy of maintenance sampling and analysis.

(c) Audits and assessments are performed and documented by any of the following:

- Radiation Protection Department
- Quality Assurance Department
- Qualified Vendors

(d) Certain elements of the Entergy Quality Assurance Program Manual are applied to the Process Control Program. [QAPM, Section A.1.c]

5.6. Dewatering Operations

[1] Processing requirements during dewatering operations include the following:

(a) All dewatering operations are performed per approved Plant or vendor operating procedures and instructions.

(b) Dewatering limitations and capabilities are verified by vendor Topical Reports or Operating and Testing Procedures.

[2] Dewatered resin activity limitations include the following:

(a) Dewatered resins will not be shipped off-site that have activities which will produce greater than 1.0E+8 rads total accumulated dose over 300 years. This is usually verified by comparing the container specific activity at the time of shipment to the following concentration limits for radionuclides with a half-life greater than five years:

- 10 Ci (0.37 TBq) per cubic foot.
- 350 uCi (12.95 MBq) per cubic centimeter

5.7. Waste Packaging

Waste in final form will be packaged in accordance with Title 10 and Title 49 of the Code of federal regulations and in accordance with current burial site criteria as is detailed in EN-RW-102.

5.8. Administrative Controls

[1] Information on solid radioactive waste shipped off-site is reported annually to the Nuclear Regulatory Commission in the Annual Radioactive Effluent Release Report as specified by the Offsite Dose Calculation Manual (ODCM) or Technical Specification. [ANO1 Technical Specifications - 5.6.3] [ANO2 Technical Specifications – 6.6.3] [WF3 Technical Specifications – 6.9.18] [GGNS ODCM – 5.6.3.c] [JAF Technical Specifications – 5.6.3] [JAF ODCM 6.2.1] [PLP ODCM, Appendix A – IV. A].

[2] All changes to the PCP SHALL be documented. All records of reviews performed SHALL be retained as required by the Quality Assurance Program. The documentation of the changes SHALL [GGNS UFSAR, Chapter 16B.1 / TRM – 7.6.3.8 paragraph 2]:

(a) Contain sufficient information to support the change with appropriate analyses or evaluations justifying the change.

(b) Include a determination that the change will maintain the overall conformance of the solidified waste product (if applicable) to existing requirements of Federal, State or other applicable regulations.

[3] All changes in the Process Control Program and supporting documentation are included in each site's next Annual Radiological Effluent Release Report to the Nuclear Regulatory Commission. [ANO ODCM - L3.2.1.C] [RBS Technical Requirements 5.5.14.1]

[4] The changes to EN-RW-105 SHALL become effective upon review and acceptance by the site's General Plant Manager (equivalent title at Palisades is Plant Superintendent) except as listed below: [PLP Technical Specifications 5.5.15]

(a) For Grand Gulf Nuclear Station, the changes to RW-105 SHALL be accomplished as specified in Grand Gulf Nuclear Station Technical Requirements Manual (TRM) Section 7.6.3.8. The changes SHALL become effective upon review and acceptance by the On-site Safety Review Committee (OSRC) **AND** the approval of the GGNS Plant General Manager. [GGNS UFSAR, Chapter 16B.1 / TRM – 7.6.3.8 paragraph 2]

5.8[4], continued

(b) For River Bend Nuclear Station, the procedure approval along with changes to RW-105 SHALL be accomplished per the River Bend Nuclear Station Technical Requirements, Section 5.5.14.1. The changes SHALL become effective upon review and acceptance by approval from the River Bend Nuclear Station Plant Manager or Radiation Protection Manager. [RBS Technical Requirements – 5.5.14.1, 5.5.14.2 & 5.8.2]

(c) For Waterford 3, the procedure approval along with changes to RW-105 SHALL be accomplished per Waterford 3 Technical Specifications 6.13.2. The changes SHALL become effective upon review and acceptance by the Waterford 3 General Plant Manager. [WF3 Technical Specifications – 6.13.2.b]

(d) For James A. FitzPatrick Nuclear Station, the procedure approval along with changes to EN-RW-105 SHALL be accomplished per the James A. FitzPatrick Station Technical Specifications, Section 5.6.3. The changes SHALL become effective upon review and acceptance through approval from the James A. FitzPatrick Nuclear Station On-Site Safety Review Committee. [JAF FSAR 11.3.5, 13.10.1.1]

(e) For IPEC, Changes to the Process Control Program SHALL become effective after final review and acceptance by the On-Site Safety Review Committee (OSRC).

5.9. Vendor Requirements

[1] Vendors performing radwaste services under 10CFR61 and 10CFR71 requirements will be on the Entergy Qualified Supplier's List (QSL). [QAPM, Section A.1.c]

[2] All radioactive waste owned by Entergy processed off-site by vendors must be processed or packaged to meet the minimum requirements listed in 10CFR Part 61.56 (a) (1) through (8) and any applicable burial site criteria.

[3] Vendors performing radwaste services on-site are to comply with the following:

(a) Dewatering and solidification services SHALL have a NRC-approved Topical Report or other form of certification documenting NRC approval of the processes and associated equipment/containers. [GGNS FSAR 11.4.4.S2, 11.4.2.3AS7]

5.9[2], continued

(b) All vendor procedures utilized for performing on-site radwaste processing services (to assure compliance with 10 CFR Parts 20, 61 and 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste) will be reviewed per the requirements of EN-LI-100, technically by the applicable site's Radiation Protection organization and only be accepted per the approvals specified in Section 5.8 [4].

(c) All changes to vendor procedures for ongoing on-site radwaste services will be reviewed technically by the site's Radiation Protection organization and screened per the requirements of EN-LI-100. Significant procedural changes will require the approvals specified in Section 5.8 [4]. During screening, the level of significance for procedural changes on equipment and process parameters may warrant the full 10CFR50.59 documentation and approval process.

(d) Plant management SHALL review vendor(s) topical reports and test procedures per applicable requirements in Section 5.8.

NOTE

The PCP does not have to include the vendor's Topical Report if it has NRC approval, or has been previously submitted to the NRC.

(e) Plant management review will assure that the vendor's operations and requirements are compatible with the responsibilities and operation of the Plant.

(f) Training requirements and records listed in Section 5.10 also apply to contracted vendors.

5.10. Miscellaneous

[1] Special tools and equipment

(a) Frequency of Use and Descriptions

Required tools and equipment will vary depending on the specific process and waste container that is used. The various tools and equipment which may be required are detailed in specific procedures developed to govern activities described in this document.

5.10, continued

[2] Pre-requisites

(a) Maintenance of Regulatory Material

Ensure that a current set of DOT, NRC, EPA and applicable State regulations, vendor processing facility and disposal site regulations and requirements are maintained at the site and are readily available for reference. The use of web based regulations is acceptable.

(b) Representative Radionuclide Sample Data

Ensure that representative radionuclide sample data is on file for each active waste stream. Unless operation conditions or changes in processing methods require increased sample frequency, data is considered to be current if it meets the requirements of EN-RW-104.

(c) Initial and Cyclic Training

- A training program SHALL be developed, implemented and maintained for all personnel involved in processing, packaging, handling and transportation of radioactive waste to ensure radwaste operations are performed within the requirements of NRC Information Bulletin 79-19 and 49CFR Part 172.700 through Part 172.704.
- Training requirements and documentation also apply to contracted on-site vendors.

NOTE

Cyclic training is defined as within three years for DOT, and two years for IATA

(d) Specific employee training is required for each person who performs the following job functions [172.702(b)].

- Classifies hazardous materials.
- Packages hazardous materials.
- Fills, loads and/or closes packages.
- Marks and labels packages containing hazardous materials.
- Prepares shipping papers for hazardous materials.
- Offers or accepts hazardous materials for transportation.

5.10[2](d), continued

- Handles hazardous materials.
- Marks or placards transport vehicles.
- Operates transport vehicles.
- Works in a transportation facility and performs functions in proximity to hazardous materials which are to be transported.
- Inspects or tests packages.

(e) Cyclic training is defined as within three years for DOT & within two years for IATA.

Copies of training records are required for as long as a person is employed and 90 days thereafter. The records should include, as a minimum, the following:

- Trainee's name and signature
- Training dates
- Training material or source reference
- Trainer's information

6.0 INTERFACES

[1] EN-LI-100, “Process Applicability Determination”

[2] EN-RW-104, “Scaling Factors”

[3] EN-QV-104, “Entergy Quality Assurance Program Manual Control”

7.0 RECORDS

[1] Documentation of pertinent data required to classify waste and verify solidification will be maintained on each batch of processed waste as required by approved procedures.

[2] Documentation will also be maintained to ensure that containers, shipping casks, and methods of packaging wastes meet applicable Federal regulations and disposal site criteria. The records of reviews performed and documents associated with these reviews will be maintained as QA records.

8.0 SITE SPECIFIC COMMITMENTS

Site	Document	Commitment Number or Reference	NMM Procedure Section/Step
ANO	ANO ODCM	L3.2.1.C	5.8 [3]
ANO	ANO1 Technical Specifications	5.6.3	5.8 [1]
ANO	ANO2 Technical Specifications	6.6.3	5.8 [1]
RBS	RBS Technical Requirements	5.5.14	*
RBS	RBS Technical Requirements	5.5.14.1	5.8 [3] 5.8 [4] (b)
RBS	RBS Technical Requirements	5.5.14.2	5.8 [4] (b)
RBS	RBS Technical Requirements	5.8.2	5.8 [4] (b)
WF3	WF3 Technical Specifications	1.22	*
WF3	WF3 Technical Specifications	6.9.18	5.8 [1]
WF3	WF3 Technical Specifications	6.13.2.b	5.8 [4] (c)
JAF	JAF ODCM	6.2.1	5.8 [1]
JAF	JAF Technical Specifications	5.6.3	5.8 [1], 5.8 [4](d)
JAF	JAF FSAR	11.3.5, 13.10.1	5.8 [4](d)
WF3	11759 – NRC IN 79-19	All	*
GGNS	GGNS UFSAR, Chapter 16B.1 / TRM	7.6.3.8 paragraph 1	1.0
GGNS	GGNS ODCM	5.6.3.c	5.8 [1]
GGNS	GGNS FSAR	11.4.5.S2	5.9 [3](a)
GGNS	GGNS FSAR	11.4.2.3AS7	5.9 [3](a)
IPEC	IPN-99-079	All	*
IPEC	Appendix B Technical Specifications	Section 4.5, RECS ODCM Part 1	*
PLP	PLP Technical Specifications	5.5.15	5.8 [4]
PLP	PLP ODCM	Appendix A – IV. A	5.8 [1]
PNPS	NRC Letter 1.98.091	All	*
PNPS	NRC Letter 1.88.078	All	*
All	QAPM	Section A.1.c	*

* Covered by directive as a whole or by various paragraphs of the directive.

9.0 ATTACHMENTS

None