

50-188

12 February 2001



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Subject: Supplemental Information, Request for Recovery of Facility Operating License
Time, License R-88

Dear Mr. Mendonca:

Previously, a request was made to the USNRC to amend the KSU nuclear reactor operating license to recover operating time unavailable during initial construction. The attached information supplements previous analysis by addressing solid and gaseous rad waste generated at the KSU nuclear research reactor under Operating License R-88. This information has been reviewed and approved by the KSU Reactor Safeguards Committee (RSC).

The attached information shows a minimal amount of radioactively contaminated waste is generated as a result of licensed activities, and that amendment of the operating license as requested will not change the production modes or rates. Approval of the request will not negatively impact the health and safety of facility workers, the public or the environment.

Correspondence regarding this matter should be directed to me at (785) 532-6657.

Sincerely,

A handwritten signature in black ink, appearing to read "Paul M. Whaley". The signature is fluid and cursive, with a large loop at the end.

Paul M. Whaley
KSU Nuclear Reactor Facility

encl: Supplement to Environmental Report Supporting Request for Recovery of Facility
Operating License Time, License R-88

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Subject: Supplement to Environmental Report Supporting Request for Recovery of Facility Operating License Time, License R-88

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SUPPLEMENT TO ENVIRONMENTAL REPORT SUPPORTING REQUEST FOR RECOVERY OF FACILITY OPERATING LICENSE TIME, R-88

The following information supplements the previously transmitted environmental report with respect to releases of solid radioactive waste and gaseous effluent.

SOLID RADIOACTIVE WASTE

Current rate of production of radioactive waste at the KSU nuclear research reactor is small. Continuation of the current operating license for an additional 14 months will maintain production of solid wastes and radioactive effluents at current rates, with no increases in waste production.

Waste disposal records show routine items produced and disposed of as waste include (1) ventilation filters, (2) bulk shield tank water filters, and (3) radiological protection waste (gloves, shoe covers, absorbent paper, etc.) and (4) spent resin. There was a one-time small volume of waste generated, related to disposal of sealed sources and repair/replacement of contaminated equipment. Table 1 contains records of waste disposal directly related to the reactor and routine operating activities. Table 2 is a compilation of records of all solid radioactive waste transferred to the State of Kansas by the KSU reactor staff for disposal.

The total volume of routine waste associated with reactor operation since 4/19/88 (a period of approximately 13 years) is 116 ft³. Therefore, solid waste is produced at an average rate 9 ft³ per year. Independent of the request for recovery of facility operating license time, primary resin replacement will be required when the resin is expended, bulk shield tank water filters will continue to be replaced at end of filter life, and reactor bay ventilation filters will continue to be replaced on a regular schedule. A periodic step increase in volume occurs when spent primary resin is replaced. Approximately 3.5 ft³ of ventilation filters are disposed of annually. These contributors to solid waste production will continue until irradiated fuel is removed, the facility is decommissioned and decontaminated.

Each year, approximately 1/3 of one 55-gallon barrel of waste is produced, associated with routine health physics precautions. Radiological surveillances, sample handling, and routine decontamination efforts require use of consumable contamination control materials. Waste contribution from gloves, filter paper, absorbent paper, etc. is partially a function of operating history. However, routine decontamination efforts and radiological surveillances are required independent of the request for recovery of facility operating license time. Some level of waste generation in this category will continue until the facility is decommissioned and decontaminated.

Therefore, approval of the recovery of facility operating license time will not affect production of radioactive waste from the KSU nuclear research reactor. The current minimal and safe level of waste production will continue if the recovery of operating license time is approved; essentially the same level of waste production will continue if the proposed license amendment is denied.

Date	Ventilation Filters	Bulk Shield Tank Water Filters	Radiological Protection Waste	Spent Resin
02/05/98	[1] 29 ft ³		7.4 ft ³	
03/25/92			7.2 ft ³	36.8 ft ³
08/07/90	[1] 1.44 ft ³			
12/16/89	[1] 4.34 ft ³	[1] 1.3 ft ³		
06/02/89	[1] 4.34 ft ³		[1] 7.2 ft ³	
04/19/88	5.8 ft ³	1.3 ft ³	9.9 ft ³	
SUBTOTAL	44.92 ft ³	2.6 ft ³	31.7 ft ³	36.8 ft ³
RATE	3.5 ft ³ /year	0.2 ft ³ /year	2.4 ft ³ /year	2.8 ft ³ /year

NOTE: [1] Volume estimated based on data for similar entries in waste disposal log; bags are assumed to be 2500 in³

DATE	ISOTOPES/CURIES	PHYSICAL DESCRIPTION
02/05/98	55 μ Ci ⁶⁰ Co	55 gal drum dry waste (gloves, shoe covers, etc.)
	< 0.1 μ Ci	29 ventilation filters
	<0.1 μ Ci	4 ventilation filters with 500-3000 cpm local readings
11/01/95	5 Ci ²³⁹ Pu	5 Pu-Be sources
05/27/94	77.56 mCi tritium	Electron source
	1.0 mCi ¹⁴ C	Sealed source
	63.822 mCi ⁶⁰ Co	Sealed source
	79.584 mCi ⁹⁰ Sr	Sealed source
	22.151 mCi ¹³⁷ Cs	Sealed source
	0.283 μ Ci ²⁴¹ Am	Sealed source
03/25/92	22.3 μ Ci ⁶⁰ Co	1 - 50 gal drum, used ion exchange resin
	7.4 μ Ci ⁶⁰ Co	1 - 50 gal drum, used ion exchange resin
	5.0 μ Ci ⁶⁰ Co	1 - 50 gal drum, used ion exchange resin
	5.0 μ Ci ⁶⁰ Co	2 - 50 gal drums, used ion exchange resin
	6.2 μ Ci ⁶⁰ Co	1 bag (2500 in ³) dry waste (gloves kim wipes, tape, etc)
	1.2 μ Ci ⁶⁰ Co	1 bag (2500 in ³) dry waste
	2.2 μ Ci ⁶⁰ Co	1 bag (2500 in ³) dry waste
	1.1 μ Ci ⁶⁰ Co	1 bag (2500 in ³) dry waste
	10.6 μ Ci ⁶⁰ Co	1 bag (2500 in ³) dry waste
08/07/90	\leq 1 μ Ci	1 bag ventilation filters (20 in. X 20 in. X 4 in.)
	\leq 1 μ Ci	6 BST (water) filters
	\leq 1 μ Ci	1 empty quart glass container used for liquid rad waste
12/16/89	\leq 1 μ Ci	1 bag ventilation filters (20 in. X 20 in. X 4 in.)
	\leq 1 μ Ci	1 bag ventilation filters (35 in. X 10 in. X 4 in.)

	$\leq 1 \mu\text{Ci}$	1 bag ventilation filters (20 in. X 20 in. X 2 in.)
	$\leq 1 \mu\text{Ci}$	6 BST (water) filters
	$\leq 1 \mu\text{Ci}$	1 bag absorbent material, glass and tape from sump pump replacement (17 in. diameter, 8 in. high)
	$\leq 1 \mu\text{Ci}$	1 box, dismantled sump pump
06/02/89	$\leq 1 \mu\text{Ci}$	5 bags baleable waste
	$\leq 1 \mu\text{Ci}$	3 bags ventilation filters
	Est. 15 $\leq 1 \mu\text{Ci}$	1 bag baleable waste
04/19/88	≤ 100 cpm above background, direct frisk	1 box ventilation filters (6.9 ft ³)
	≤ 0.1 mR/hr $\beta\gamma$ above background	1 box bulk shield tank filters (1.3 ft ³)
	Max 2.0 mr/hr $\beta\gamma$ internal, each – activation products	2 bags baleable waste (5.1 ft ³ each)
	Max 0.2 mr/hr $\beta\gamma$ internal, each – activation products	2 bags baleable waste (3.6 ft ³ each)
	Max 5.0 mr/hr $\beta\gamma$ internal, each – activation products	1 bag baleable waste (1.2 ft ³)
	≤ 0.1 mR/hr $\beta\gamma$ above background	2 bags ventilation filters and bulk shield tank filters (2.9 ft ³ each)

GASEOUS EFFLUENT

The 1968 safety analysis (supporting steady state power limit upgrade from 100 kW to 250 kW) as well as calculations performed to support the draft safety analysis report demonstrate by analysis that the gaseous effluent releases from the KSU TRIGA Mark II nuclear research reactor are minimal and meet regulatory requirements. Approval of the proposed recovery of facility operating license time will maintain the current acceptable and safe rate of gaseous effluent production, and does not pose a significant risk to the facility workers, the public or the environment. Since the proposed change merely extends the amount of time the current authorization bases applies, no increases in production of gaseous effluent will occur as a result of approval of this amendment.

The safety analysis supporting the 1968 license upgrade for the KSU reactor to 250 kW identified three potential contributors to gaseous effluent releases: Argon 41, Nitrogen 16 and Tritium. Because of the short half-life and the time required for migration from production to release, Nitrogen 16 is not important to off-site releases. Measured tritium specific activity levels are noted in the draft Safety Analysis Report at less than 10^{-3} $\mu\text{Ci/g}$. Using the grossly conservative assumption that reactor bay atmosphere is saturated at 30°C, the water concentration in the air would be less than 3×10^{-5} g/mL and the activity concentration in the atmosphere would be less than 3×10^{-8} $\mu\text{Ci/mL}$. This is well below the 10CFR20 Appendix-B DAC of 2×10^{-5} $\mu\text{Ci/mL}$ and the atmospheric effluent limit of 10^{-7} $\mu\text{Ci/mL}$, with no credit for dilution before exposure from the plume originating on the roof of the facility. Therefore, the focus of analysis for effluent releases is production and migration of Argon 41.

The 1968 analysis identifies the largest source of Argon 41 to be the void spaces in the rotary specimen rack. The rotary specimen rack is greater than 16 feet below the surface of the reactor pool. Two dry-tube penetrations connect the void spaces in the

rotary specimen rack to the reactor bay atmosphere; one for the operating shaft and one for specimen insertion and removal. The sample insertion port is normally blocked with an aluminum cap. There is no forced circulation and no motive force to cause the Argon 41 in the rotary specimen rack to escape into the reactor bay. Therefore, the 1968 analysis focused on operation of the rabbit system using normal air to insert and remove rabbit capsules. The rabbit system has been modified since 1968 and now uses helium gas as a motive force; therefore, the original analysis provides very conservative results. The potential for release of Argon 41 is lower than identified in the 1968 analysis.

A more rigorous treatment has been proposed in the draft Safety Analysis Report, based on 500 kW operations, as attached (Radiological Impact of ^{41}Ar and ^{16}N During Normal Operations). Significant conservative assumptions in these calculations include:

- 500 kW power level (current power level is 250 kW)
- Continuous operation (operating history as previously transmitted in support of the request for recovery of operating time is not continuous)
- Steady state Argon 41 production (Argon 41 has a half life slightly less than 2-hours; typical irradiations at KSU are less than 6-8 hours)
- 100% release of Argon 41 to atmosphere

Calculations show a full year of exposure to the maximum theoretical concentration of Argon 41 possible from operation of the KSU reactor would result in a dose outside the facility of 2.8 mrem, well below applicable limits.

CONCLUSION

Recovery of 14 months of facility operating license time is safe, will not change radiological consequences from current operating conditions, and will not significantly impact the health and safety of workers, the public or the environment.

A. RADIOLOGICAL IMPACT OF ⁴¹Ar AND ¹⁶N DURING NORMAL OPERATIONS

A.1 Introduction

A.1.1 Purpose

The purpose of this appendix is to show the methods and calculations used to predict the production, concentrations, and dose rates from ⁴¹Ar and ¹⁶N associated with normal operation of the KSU TRIGA Mk. II nuclear reactor.

The nuclide ⁴¹Ar is produced by thermal neutron absorption by natural ⁴⁰Ar in the atmosphere and in air dissolved in the reactor cooling water. The activation product appears in the reactor room (bay) and is subsequently released to the atmosphere through the reactor bay ventilation exhaust stream.

The nuclide ¹⁶N is produced by fast neutron interactions with oxygen. The only source of ¹⁶N in the reactor that needs consideration results from interactions of neutrons with oxygen in the cooling water as it passes through the reactor core. Any interactions with oxygen in the atmosphere is relatively insignificant and is neglected in this analysis.

A portion of the ¹⁶N produced in the core is eventually released from the top of the reactor tank into the reactor bay. The half-life of ¹⁶N is only 7.14 seconds, so its radiological consequences outside the reactor bay are insignificant.

Although not expected, the cladding of a fuel element could fail during normal operations as a result of corrosion or manufacturing defect. Should a failure occur, a fraction of the fission products, essentially the noble gases and halogens, would be released to the reactor tank and, in part, ultimately become airborne and released to the atmosphere via building ventilation. This operational occurrence, taking place in air, is addressed in Chapter 13 as the design basis accident for the TRIGA reactor.

Neutron interactions with structural and control materials, including cladding, as well as materials irradiated for experimental purposes, result in the formation of activation products. These products are in the nature of fixed sources and are mainly a source of occupational radiation exposure. Administrative controls preclude the significant formation of airborne activation products, other than the aforementioned ⁴¹Ar.

A.1.2 Radiological Standards

The concentration to dose rate (effective dose equivalent) conversion factor for submersion in an infinite atmosphere of ⁴¹Ar is as follows: 2.3×10^{-10} Sv/h per Bq/m³ (EPA 1993). For 2000 hours annual occupational exposure and 50 mSv maximum permissible annual exposure, this translates to a derived air concentration of 1.15×10^5 Bq/m³ or 3.1×10^{-6} μ Ci/cm³ (3×10^{-6} μ Ci/cm³ as specified in 10CFR20). For 8766 hours annual public exposure and 1 mSv maximum permissible annual exposure, this translates to a derived air concentration of 526 Bq/m³ or 1.4×10^{-8} μ Ci/cm³ (1×10^{-8} μ Ci/cm³ as specified in 10CFR20).

A.1.3 KSU TRIGA Design Bases

A.1.3.1 General System Parameters

The calculations for ^{41}Ar and ^{16}N releases during normal operations are based on the following system parameters.

Table A.1. General system parameters for normal operations at 500 kWt full power.

Parameter	Symbol	Value
Reactor steady power	P	500,000 W
Core coolant mass flow rate ^a	w	0.108 kg/s
Core coolant density	ρ	1.0 g/cm ³
Core avg. thermal neutron flux at full power (E ring) ^b	ϕ_{th}	8.2×10^{12} n/cm ² s
Core avg. fast neutron flux at full power (E ring) ^b	ϕ_f	1.2×10^{13} n/cm ² s
Thermal neutron flux in RSR at full power	ϕ_{RSR}	3.6×10^{12} n/cm ² s
Total neutron flux per watt at fast (piercing) beam port		1700 n/cm ² s (0.5 MeV avg)
Total neutron flux per watt at tangential beam port		560 n/cm ² s (0.1 MeV avg)
Fuel element heated length	L	0.381 m
Flow cross sectional area per fuel element ^a	A	6.2 cm ²
Mass flow rate per fuel element ^a	\dot{m}	108 g/s
Reactor tank diameter		1.98 m
Reactor tank depth		6.25 m
Reactor tank water depth above core		4.88 m (16 ft)
Coolant volume in reactor tank	V_c	1.92×10^7 cm ³
Air volume in reactor bay (144,000 ft ³)	V_{bay}	4.078×10^9 cm ³
Air volume in rotary specimen rack	V_{RSR}	3.75×10^4 cm ³
Ventilation rate for reactor bay (air changes hourly) ^d	λ_v	0.368 h ⁻¹

^aSee §4.6 of this report.

^bSee §5.8 of Operations Manual.

^cSee §13.2.2.2 of this report

^dSee letter B.C. Ryan (KSU) to Theodore Michaels (NRC) 15 Jan 99.

A.1.3.2 Reactor Core Parameters

Modeling of the reactor core for radiation transport calculations is based on the following approximations. For purposes of radiation shielding calculations the TRIGA reactor core may be approximated as a right circular cylinder 0.4572 m diameter (OD of F ring). The fuel region is 0.381 m (15 in.) high. On each end axially is a graphite zone 0.0874 m (3.44 in.) high and an aluminum grid plate 0.0191 m (0.75 in.) thick. In 91 fuel locations, there are 83 fuel elements, 3 standard control rods and 1 transient control rod, 1 void location, 1 central thimble (void), 1 source (assume void), and 1 pneumatic transfer site (assume void). The fuel region may be treated as a homogeneous zone, as may be the axial graphite zones and the grid plates.

Fuel elements are 1.43-in. ID and 1.47-in OD, clad with type 304 stainless steel¹. Fuel density is 5996 kg/m³. Fuel composition is 8.5% uranium, 91.4% ZrH_{1.65}. The uranium is 20% ²³⁵U and 80% ²³⁸U. Steel density is 7900 kg/m³. Standard control rods are 0.875-in. OD, the transient rod 1.25-in. OD. Both types of rods are clad with 30-mil thick aluminum (2700 kg/m³ density). The control material may be approximated as pure graphite, with density 1700 kg/m³.

¹ Composition, by weight, 2% Mn, 18% Cr, 8% Ni, balance Fe.

In radiation transport calculations, the core is modeled conservatively as a central homogenous fuel zone (air density neglected) bounded on either end by a homogeneous axial reflector zone, and by a 0.75-in. thick aluminum grid plate, treated as a homogeneous solid. Densities of the homogenous zones are as follow:

Fuel	3602 kg/m ³
Reflector	1147 kg/m ³
Grid Plate	2700 kg/m ³

Composition of the three zones, by weight fraction, are given in the following table.

Table A2. Compositions of homogenized core zones.

Element	Mass Fraction	Element	Mass Fraction
<i>Fuel Zone</i>		<i>Axial Reflector Zone</i>	
C	0.0617	C	0.7920
Al	0.0010	Al	0.0033
H	0.0139	Mn	0.0041
Zr	0.7841	Cr	0.0368
Mn	0.0013	Ni	0.0164
Cr	0.0117	Fe	0.1474
Ni	0.0052	<i>Grid Plate</i>	
Fe	0.0469	Al	1.0000
U	0.0741		

A.1.3.3 Reactor Bay Parameters

For purposes of radiation dose calculations within the reactor bay, the dimensions are approximated as follows:

The reactor bay is approximated as a right circular cylinder 36 ft (10.973 m) high and 36.68 ft (11.18 m) radius. The reactor vessel structure is approximated as a right circular cylinder, co-axial with the bay, 22 ft (6.706 m) high and 11 ft (3.3528 m) radius. The free volume is 144,000 ft³ (4078 m³). The site boundary, at its nearest approach to the reactor bay, is about 2 m beyond the bay boundary, that is, at a radius of 13.13 m from the center of the reactor.

A.2 Radiological Assessment of ⁴¹Ar Sources

A.2.1 Production of ⁴¹Ar from Beams

Operating experience has shown that neutron radiography is performed at a thermal power of 10 kW or less, i.e., at a neutron flux of 2×10^7 cm⁻² s⁻¹ or less. We assume here that this is the flux

Appendix A

density along the beam port, which has a cross sectional area of 324 cm² (8-in diameter). In other words, in a radiography operation $S = 6.48 \times 10^9$ neutrons per second enter the atmosphere essentially in a parallel beam. The microscopic cross section for thermal neutron absorption in ⁴⁰Ar is 0.66 barns, so the macroscopic cross section for thermal neutron absorption in ⁴⁰Ar in air (0.0129 weight fraction) is $\mu = 1.54 \times 10^{-7}$ cm⁻¹. The maximum distance of travel of a neutron is from the reactor tank wall to the exterior wall of the reactor bay, namely, about $L_b = 1020$ cm. The decay constant for ⁴¹Ar is $\lambda_r = 0.380$ h⁻¹ and the ventilation rate is $\lambda_v = 0.368$ h⁻¹. The volume of the reactor bay is $V_{bay} = 4.08 \times 10^9$ cm³. Thus, the activity concentration of airborne ⁴¹Ar after sustained operation with an open beam port, at 10 kW power, is given by

$$C \text{ (Bq / m}^3\text{)} = \frac{S\lambda_r(1 - e^{-\mu L_b})}{V_{bay}(\lambda_r + \lambda_v)} = 1.27 \times 10^{-4} \text{ Bq/cm}^3, \quad (1)$$

or 3.42×10^{-9} $\mu\text{Ci/mL}$ in conventional units. This is well below 10CFR20 limits, even for public exposure.

A.2.2 Production of ⁴¹Ar in Rotary Specimen Rack

The air volume of the rotary specimen rack (RSR) is approximately² $V_{RSR} = 3.75 \times 10^4$ cm³ (HSR p. 28) and the thermal neutron flux density in the RSR is $\phi_{th} = 3.6 \times 10^{12}$ cm⁻²s⁻¹ at 500 kW thermal power. After sustained operation at full power, the equilibrium ⁴¹Ar activity (Bq) in the RSR volume is given by

$$A_{RSR} = \phi_{th} \Sigma_{41} V_{RSR} = 2.08 \times 10^{10} \text{ Bq}, \quad (2)$$

or, 0.561 Ci in conventional units. If this activity were flushed in to the reactor-bay atmosphere as a result of a water leak into the RSR, the initial activity concentration would be $A_0/V_{bay} = 1.38 \times 10^{-4}$ $\mu\text{Ci/mL}$. This would instantaneously be well above even the occupational DAC for ⁴¹Ar. However, with radioactive decay and ventilation, the concentration would decline in time according to

$$A(t) = A_0 e^{-(\lambda_r + \lambda_v)t}. \quad (3)$$

If a worker were exposed to the full course of the decay, the cumulative concentration ($\mu\text{Ci h/mL}$) in the reactor bay would be

$$\frac{1}{V_{bay}} \int_0^{\infty} dt A(t) = \frac{A_0}{V_{bay}(\lambda_r + \lambda_v)} = 1.8 \times 10^{-4} \quad (4)$$

This is well below the 6×10^{-3} $\mu\text{Ci h/mL}$ annual limit of 2000 DAC hours specified in 10CFR20.

A.2.3 Production of ⁴¹Ar from Condant Water

The ⁴¹Ar activity in the reactor tank water results from irradiation of the air dissolved in the water. The following calculations evaluate the rate at which ⁴¹Ar escapes from the water into the

Approximated as a section of a cylindrical annulus, with 28-in. OD, 24-in. ID, and 14-in. height.

reactor bay. The following variables plus those in Table A.1 are used in the calculations of ^{41}Ar concentrations in the core region, in the reactor tank outside the core, and in the reactor bay air.

- V_{core} = volume of core region (83 elements) = $83 \times A \times L = 2.0 \times 10^4 \text{ cm}^3$
- C_{40} = ^{40}Ar atomic density (cm^{-3}) in coolant
- σ = microscopic cross section (cm^2)
- ρ = density (1.0 g cm^{-3})
- ν = volumetric flow rate through core ($\text{cm}^3 \text{ s}^{-1}$)
- τ = residence time for coolant in core at full power (s)
- T = out-of-core cycle time for coolant.

From Chapter 4, the mass flow rate through the core for a single element at full power is $\dot{m} = 108 \text{ g/s}$, thus, for a core of 83 elements, $\nu = 83 \times 108 = 9060 \text{ cm}^3/\text{s}$, and the residence time for coolant passing through the core is $\tau = 83AL/\nu = 2.19 \text{ s}$. The velocity within the core is $108/A = 17.4 \text{ cm/s}$. Similarly, with 83 fuel elements, the average out-of-core cycle time is about $V_d/\nu = 2150 \text{ s}$.

The saturated concentrations of argon in water at the coolant inlet temperature of 27°C is approximately $6.1 \times 10^{-5} \text{ g per cm}^3$ of water (Dorsey 1940). If it is assumed that air is saturated with water vapor above the water tank (27 mm Hg vapor pressure at 27°C) and that the mole fraction of argon in dry air is 0.0094, the partial pressure of argon in air above the tank is $0.0094(760 - 27) = 6.9 \text{ mm Hg}$. By Henry's law, the concentration in water at the inlet temperature is $6.1 \times 10^{-5} \times 6.9/760 = 5.5 \times 10^{-7} \text{ g/cm}^3$ ($C_{40} = 8.3 \times 10^{15} \text{ atoms/cm}^3$).

The number of atoms per second of ^{41}Ar produced in the core is $C_{40} \times V_{core} \times \sigma\phi_{th} = 9.0 \times 10^8$. If it were assumed that all atoms escape to the containment volume, the steady-state activity concentrations in the reactor bay atmosphere would be

$$C \text{ (Bq / cm}^3\text{)} = \frac{9.0 \times 10^8 \lambda_r}{V_{bay}(\lambda_r + \lambda_v)} \quad (5)$$

With $\lambda_r = 1.06 \times 10^{-4} \text{ s}^{-1}$ and $\lambda_v = 1.02 \times 10^{-4} \text{ s}^{-1}$, the ^{41}Ar concentrations in the reactor bay would be $2.2 \times 10^{-1} \text{ Bq/cm}^3$ ($6.0 \times 10^{-6} \text{ } \mu\text{Ci/mL}$) without ventilation and 0.112 Bq/cm^3 ($3.1 \times 10^{-6} \text{ } \mu\text{Ci/mL}$) with ventilation. Even with the extremely conservative approximation of 100% release of ^{41}Ar to the atmosphere, the estimated steady state concentration, under ventilation, is less than the DAC.

A.2.4 Maximum Impact of ^{41}Ar Outside the Operations Boundary

The ^{41}Ar produced in the reactor bay during normal operations is released to the atmosphere via an exhaust fan at approximately height $h = 11$ meters above grade. The flow rate is $4.17 \times 10^5 \text{ cm}^3/\text{s}$ (884 cfm). At the steady state concentration computed in the previous section, the release rate would be $Q = 1.29 \text{ } \mu\text{Ci/s}$. The maximum downwind concentration (pCi/cm^3), at grade, may be computed using the Sutton formula (Slade 1968):

$$C_{\max} = \frac{2Q}{e\pi\bar{u}h^2} \frac{C_z}{C_y} \quad (6)$$

Appendix A

in which \bar{u} is the mean wind speed (m/s), $e = 2.718$, and C_y and C_z are diffusion parameters in the crosswind and vertical directions respectively. The maximum concentration downwind occurs at distance d (m) given by

$$d = (h / C_z)^{\frac{2}{2-n}}, \quad (7)$$

in which the parameter n is associated with the wind stability condition. In this calculation, we adopt the values of n and C_z use in the McClellan AFB SAR. Mean wind speeds, by stability class are inferred from the data in Chapter 2. Calculations are shown in Table 2.3.

Table A.3. Atmospheric dispersion calculations.

Pasquill stability class	u (m/s)	n	C_y (m ^{n/2})	C_z (m ^{n/2})	d (m)	C_{\max} (pCi/cm ³)
Extremely unstable (A)	1.6	0.2	0.31	0.31	53	0.001561
Slightly unstable (C)	4.0	0.25	0.15	0.15	135	0.000624
Slightly stable (E)	3.5	0.33	$4C_z$	0.075	393	0.000178
Extremely stable (G)	0.77	0.5	$8C_z$	0.035	2140	0.000405

Note that over the full range of conditions examined in Table 2.3, the peak downwind concentration is substantially below the DAC of 0.01 pCi/cm³ established in 10CFR20. Indeed, using the EPA concentration to dose conversion factor (8.66×10^{-4} rem/h per pCi/cm³) from Section A.1.2, a full year exposure at the maximum concentration of 0.0016 pCi/cm³ would result in a dose of only 2.8 mrem, well with applicable limits.

A2.5 Radiological Assessment of ¹⁶N Sources

Nitrogen-16 is generated by the reaction of fast neutrons with oxygen and the only significant source results from reactions with oxygen in the liquid coolant of the reactor. The nuclide has a half-life of 7.13 s (decay constant $\lambda_{16} = 0.0972 \text{ s}^{-1} = 350 \text{ h}^{-1}$) and emits, predominantly, 6.13-MeV gamma rays. According to the McClellan AFB SAR, the effective cross section for the ¹⁶O(n,p)¹⁶N reaction, averaged over the fast-neutron energy spectrum in the TRIGA or over the fission-neutron spectrum is $\sigma_{np} = 2.1 \times 10^{-29} \text{ cm}^2$.

The atomic density C_N (cm⁻³) of the nuclide as it leaves the reactor core is given in terms of the oxygen density in water, $C_O = 3.34 \times 10^{22}$, as

$$C_n = \frac{\Phi_f C_0 \sigma_{np}}{\lambda_{16}} (1 - e^{-\lambda_{16} t}) = 1.66 \times 10^7, \quad (8)$$

or 1.61×10^6 Bq/cm³ (43.6 μCi/mL).

As the warmed coolant leaves the core, it passes through 1.5-in diameter ($A_{gp} = 11.4$ cm²) channels in the upper grid plate, but the flow is restricted by the triflute upper end fixture of the fuel element. This leaves a flow area of $A_0 \approx A_{gp} [1 - (3/\pi) \sin 30^\circ \cos 30^\circ] = 6.69$ cm² for each element, or 555 cm² for the total of 83 elements.

If it is assumed very conservatively that flow to the surface of the tank (488 cm) takes place at the velocity v (8960 cm³/s) divided by the 555 cm² flow area, namely, 16.1 cm/s, then the transit time for an element of fluid to reach the tank surface is $488 \text{ cm} \div 16.1 \text{ cm/s} = t_{rise} = 30.3$ s. The ¹⁶N concentration at the tank surface is accordingly reduced by the factor $\exp(-\lambda_{16} t_{rise})$, namely, to 2.29 μCi/mL. It is assumed further that, near the surface of the reactor tank, the ¹⁶N is dispersed across the entire 30,900 cm² cross sectional area of the pool, resulting in a concentration of $2.29 \times 555/30900 = 0.0411$ μCi/mL.

To estimate the potential radiation dose rate above the reactor tank, the source is modeled as a vertical cylinder of water, 198 cm in diameter and 488 cm deep, with a uniform concentration of ¹⁶N equal to 0.0411 μCi/mL. A straightforward radiation transport calculation reveals an exposure rate of 40 mR/h at one meter above the surface of the water tank.

Only a small proportion of the ¹⁶N atoms present near the tank surface are transferred to the air of the reactor bay. Upon its formation, the ¹⁶N recoil atom has various degrees of ionization. According to Mittl and Theys (1961) practically all ¹⁶N combines with oxygen and hydrogen atoms in high purity water, and most combines in an anion form, which has a tendency to remain in the water. In this consideration, and in consideration of the very short half life of the nuclide, the occupational consequences of any airborne ¹⁶N are deemed negligible in comparison to consequences from the shine from the reactor tank. Similarly, off-site radiological consequences from airborne ¹⁶N are deemed negligible in comparison to those of ⁴¹Ar.

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