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Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

DATE: 30 March 2006

SUBJECT: Update to Chapter 11, Appendix A of the Proposed Safety Analysis Report for the K-State TRIGA Mark II Nuclear Research Reactor

Dear Mr. Hughes:

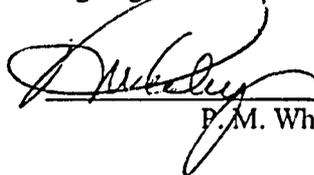
A license renewal request was submitted for the Kansas State University in 2002, and the facility is currently operating under "timely renewal" provisions. USNRC review has resulted in Requests for Additional Information (RAI), previously addressed. Further clarification is provided in the attached and revised Chapter 11, Appendix A, of the Safety Analysis Report.

The nuclear and physical characteristics of argon and nitrogen as used in calculations were moved from A.1.1, A.2.1, A.2.2, A.2.3, and A.2.5 to A.1, Introduction. Radiological Standards was revised to more clearly identify standards. The table "General Parameters" was revised (1) 1,250 kW in the title, (2) added physical dimensions related to core, (3) changed ventilation rate from "changes per hour" to flow rate. In the same section as the table, paragraph "reactor core parameters" was revised to indicate 4 rods and better define core components related to water volume in the core. The order of calculations was changed in A.2.2, and calculations in A.2.3 were expanded and corrected.

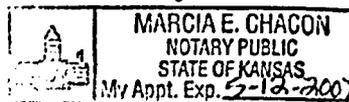
If you have any questions or comments concerning this matter, you may contact me at 785-532-6657 or whaley@ksu.edu.

I verify under penalty of perjury that the foregoing is true and correct,

Executed on 30 March 2006,


P. M. Whaley

Docket No. 50-188
Enclosures: as indicated



marciae.chacon
3/30/2006

AB20

Chapter 11 Appendix A

Radiological Impact of ^{41}Ar and ^{16}N During Normal Operations

A.1 Introduction

Normal operation of the KSU reactor results in two potential source terms for radioactive gaseous effluent at significant levels, ^{41}Ar and ^{16}N . There are variations in experimental configuration and possible scenarios where the production of ^{41}Ar may be different than the routine operations; these scenarios do not produce not long term, routine radioactive effluent but are assessed to determine if the amount of radioactive effluent is so high as to impact the annual exposure that might result from routine operations.

The nuclide ^{41}Ar is produced by thermal neutron absorption by natural ^{40}Ar 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 microscopic cross section for thermal neutron absorption in ^{40}Ar is 0.66 barns, so the macroscopic cross section for thermal neutron absorption in ^{40}Ar in air (0.0129 weight fraction) is $\mu = 1.54 \times 10^{-7} \text{ cm}^{-1}$.

The saturated concentrations of ^{40}Ar 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 of argon in water at the inlet temperature is $6.1 \times 10^{-5} \times 6.9/760 = 5.5 \times 10^{-7} \text{ g cm}^{-3}$ ($C_{40} = 8.3 \times 10^{15} \text{ atoms cm}^{-3}$). Therefore, macroscopic cross section for thermal neutron absorption in ^{40}Ar in water is $5.48 \times 10^{-9} \text{ cm}^{-1}$.

With a half-life of 109.61 minutes, the decay constant for ^{41}Ar is $\lambda_r = 0.379 \text{ h}^{-1}$, $6.32 \times 10^{-3} \text{ m}^{-1}$, or $1.05 \times 10^{-4} \text{ s}^{-1}$.

The nuclide ^{16}N is produced by fast neutron interactions with oxygen. Although a portion of the ^{16}N produced in the core is eventually released from the top of the reactor tank into the reactor bay, the half-life of ^{16}N is 7.14 seconds; radiological consequences from effluent release of ^{16}N are insignificant. The only source of ^{16}N in the reactor that needs consideration results from interactions of neutrons with oxygen in the cooling water as it passes through the reactor core. Interaction with oxygen in atmosphere is relatively insignificant and is neglected in this analysis.

According to the McClellan AFB SAR, the effective cross section for the $^{16}\text{O}(n,p)^{16}\text{N}$ 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 ^{16}N nuclide has a half-life of 7.13 seconds, corresponding to a decay constant $\lambda_{16} = 0.0972 \text{ s}^{-1} = 350 \text{ h}^{-1}$ and emits, predominantly, 6.13-MeV gamma rays.

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A.1.1 Purpose

This appendix shows methods and calculations predicting production, concentrations, and dose rates from $^{41}\text{Ar}/^{16}\text{N}$ associated with normal operation of the KSU TRIGA Mk. II nuclear reactor.

Should a cladding failure occur during normal operations, a fraction of the fission products would be released to the reactor tank with the noble gases and halogens evolving from the pool to the atmosphere via building ventilation. This operational occurrence, taking place in air, is addressed in Chapter 13 as the maximum hypothetical accident for the TRIGA reactor.

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

A.1.2 Radiological Standards

Environmental Protection Agency publication, Federal Guidance Report No. 11 (*Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*) is based on the International Council on Radiation Protection Publication 30 and the National Council on Radiation Protection Report 22. The report provides derived guides for controlling internal exposure to radionuclides in the work place, including the Annual Limit on Intake and the Derived Air Concentration (DAC), including dose conversion factors. The Derived Air Concentration is used in 10CFR20 and defined:

Derived air concentration (DAC) means the concentration of a given radionuclide in air which, if breathed by the reference man for a working year of 2,000 hours under conditions of light work (inhalation rate 1.2 cubic meters of air per hour), results in an intake of one ALI. DAC values are given in Table 1, Column 3, of appendix B to §§ 20.1001-20.2401. *Reference man* means a hypothetical aggregation of human physical and physiological characteristics arrived at by international consensus. These characteristics may be used by researchers and public health workers to standardize results of experiments and to relate biological insult to a common base.

In addition, 10CFR20 Appendix B specifies limits for assessment and control of dose to the public, equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year (i.e., 8760 h), would produce a total effective dose equivalent of 0.05 rem (±0 millirem or 0.5 millisieverts).

Table A.1, Limits and Terms

Source/Quantity	Value
10CFR20.1101(d)	10 mrem TEDE from stack emissions
10CFR20 Appendix B, Derived Air Concentration ^{41}Ar	$3 \times 10^{-6} \mu\text{Ci cm}^{-3}$
10CFR20 Appendix B, Effluent Limit ^{41}Ar	$1 \times 10^{-8} \mu\text{Ci cm}^{-3}$
EPA Federal Guidance Report 11 Conversion factor, activity to dose for ^{41}Ar submersion	$2.17 \times 10^{-10} \text{ Sv h}^{-1}$
	$0.803 \text{ mrem h}^{-1} \text{ pCi ml}^{-1}$

A.1.3 KSU TRIGA Design Bases

General System Parameters

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

$$\text{Flowrate}(kg\ s^{-1}) = 0.0115[P(kW)]^{0.3602}$$

Table A.2, General System Parameters for Normal Operations at 1,250 kWt Full Power.

Parameter	Symbol	Value
Reactor steady power	<i>P</i>	1,250,000 W
Core coolant mass flow rate (per element) ^a	<i>m</i>	0.150 kg s ⁻¹
Core coolant density	<i>ρ</i>	1.0 g cm ⁻³
Thermal neutron flux at E ring (core ave)	<i>φ_{th}</i>	2.05 × 10 ¹³ n cm ⁻² s ⁻¹
Fast neutron flux at full power at E ring (core ave)	<i>φ_f</i>	3.00 × 10 ¹³ n cm ⁻² s ⁻¹
Thermal neutron flux in RSR	<i>φ_{RSR}</i>	9.00 × 10 ¹² n cm ⁻² s ⁻¹
Total neutron flux per watt at fast (piercing) beam port		4250 n cm ⁻² s ⁻¹ (0.5 MeV avg)
Total neutron flux per watt at tangential beam port		1400 n cm ⁻² s ⁻¹ (0.1 MeV avg)
Fuel element heated length	<i>L</i>	0.381 m
Flow cross sectional area per fuel element ^a	<i>A</i>	6.2 cm ²
Mass flow rate per fuel element ^a	<i>m</i>	108 g s ⁻¹
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	<i>V_c</i>	1.92 × 10 ⁷ cm ³
Reflector inner diameter (HSR)	<i>d_{rf}</i>	45.72 cm. (18 in.)
Core volume	<i>V_{core}</i>	62,550 cm ³ (3817 in ³)
Fuel element heated volume	<i>V_E</i>	417.2 cm ³
Coolant volume in core ^d	<i>V_w</i>	27,090 cm ³
Air volume in reactor bay (144,000 ft ³)	<i>V_{bay}</i>	4.078 × 10 ⁹ cm ³
Reactor bay ventilation flow rate ^e	<i>w_v</i>	4.17 × 10 ⁵ cm ³ s ⁻¹ (884 cfm)
Air volume in rotary specimen rack	<i>V_{RSR}</i>	3.75 × 10 ⁴ cm ³

^aSee §4.6 of this report.

^bSee §5.8 of Operations Manual.

^cSee §13.2.2.2 of this report

^d *V_{core}* -85* *V_E* (91 positions, 83 elements, source, rabbit and 6 water filled positions)

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

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 (18 in.) 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, 4 standard control rods and 1 transient control rod, 1 water filled location, 1 central

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thimble (water), 1 source & 1 pneumatic transfer site (water displaced). 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. (3.6 cm) ID and 1.47-in (3.7 cm) OD, clad with type 304 stainless steel¹. Fuel density is 5996 kg m⁻³. Fuel composition is 8.5% uranium, 91.4% ZrH_{1.65}. The uranium is 20% ²³⁵U and 80% ²³⁸U. Steel density is 7900 kg m⁻³. 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 m⁻³ density). The control material may be approximated as pure graphite, with density 1700 kg m⁻³.

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 A.2, 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		
Fe	0.0469	<i>Grid Plate</i>	
U	0.0741	Al	1.0000

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.

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

A.2 Radiological Assessment of ⁴¹Ar Sources

A.2.1 Production of ⁴¹Ar from Beams

Operation with a fully open beam port is not a routine operational condition. Beam port operations normally have shielding, collimation and beam stops that prevent a full beam from penetrating the column defined by the beam port into air volume between the reactor and the reactor bay wall. Operating experience with neutron radiography performed at 10 kW involves a neutron flux of $2 \times 10^7 \text{ cm}^{-2} \text{ s}^{-1}$ or less. We assume here that this is the flux density along the beam port, which has a cross sectional area of 324 cm^2 (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 maximum distance of travel of a neutron is from the reactor tank wall to the exterior wall of the reactor bay, namely, about $L_b = 1020 \text{ cm}$. Thus, the activity concentration of airborne ⁴¹Ar after sustained operation with an open beam port at 10 kW is given by:

$$C = \frac{S \cdot \lambda \cdot (1 - e^{-\mu \cdot L_b})}{V_{bay} \cdot (\lambda_r + \lambda_{bay})} = 1.05 \times 10^{-4} \text{ Bq cm}^{-3} \quad (1)$$

or $8.32 \times 10^{-5} \text{ } \mu\text{Ci mL}^{-1}$ in conventional units. Operations at maximum power are not performed for radiography, and radiography is not performed long enough to achieve equilibrium ⁴¹Ar. Therefore, scaling the calculation for sustained operations at 1,250 kW provides an extremely conservative bound on ⁴¹Ar production. Scaling the 10 kW ⁴¹Ar production value to 1,250 kW results in $4.28 \times 10^{-7} \text{ } \mu\text{Ci mL}^{-1}$ which is slightly above submersion DAC for occupational exposure; however, conditions for the source term are related to a very unusual set of conditions (open beam port with no shielding) that are not continuous in two respects. Shielding for radiography external to the beam port limits the beam to less than $\frac{1}{2}$ of the analyzed volume. Radiography configuration is implemented only for radiography operations, a small fraction of all operations. Typically radiography occurs less than 1 day per month. Radiography operations are inherently discontinuous as the purpose of individual operations are met when the image is obtained. Typically a day of radiography operations involves less than 4 hours of operation at full power. These conservatisms assure limits are met with no further consideration.

A.2.2 Production of ⁴¹Ar in Rotary Specimen Rack

The air volume in the rotary specimen rack does not freely exchange with the air in the reactor bay; there is no motive force for circulation and the rotary specimen rack opening is routinely covered during operation. If the rotary specimen rack were to flood, water would force the air volume in the RSR into the reactor bay. After sustained operation at full power, the equilibrium ⁴¹Ar activity (Bq) generated in the RSR volume and diluted in the reactor bay atmosphere (A_{Bay}^{Ar41}) is calculated (using terms defined in the Introduction and Table A.2) by:

$$A_{Bay}^{Ar41} = \mu \cdot \Phi_{RSR} \frac{V_{RSR}}{V_{Bay}} \quad (2)$$

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This is the initial concentration, while limits are based on annual exposure. With radioactive decay and ventilation, the concentration would decline in time with an effective decay constant determined by $\lambda_{eff} = \lambda_{decay} + \lambda_v$.

Using λ_{eff} in equation (2) equation, time dependent behavior is:

$$A_{Bay}^{Ar41}(t) = A_{Bay,t=0}^{Ar41} \cdot e^{-\lambda_{eff} \cdot t} = \mu \cdot \Phi_{RSR} \frac{V_{RSR}}{V_{Bay}} \cdot e^{-\lambda_{eff} \cdot t} \quad (3)$$

If a worker were exposed to the full course of the decay over one year, average concentration ($\mu\text{Ci mL}^{-1}$) in the reactor bay would be calculated by the integration of equation (4) over 1 year:

$$\bar{A}_{Bay}^{Ar41} = \frac{\int_0^{1y} A_{Bay,t=0}^{Ar41} \cdot e^{-\lambda_{eff} \cdot t} \cdot dt}{y} = \frac{A_{Bay,t=0}^{Ar41}}{\lambda_{eff} \cdot y} = \frac{\mu \cdot \Phi_{RSR} \frac{V_{RSR}}{V_{Bay}}}{\lambda_{eff} \cdot y} \quad (4)$$

Considering only radioactive decay (λ_v of 0), equation (4) indicates an average concentration of $1.03 \times 10^{-7} \mu\text{Ci mL}^{-1}$ over the year. Considering radioactive decay in combination with stack flow (removing air), equation (4) indicates an average of $5.27 \times 10^{-8} \mu\text{Ci mL}^{-1}$ over the year. Since a worker will experience less than one Derived Air Concentration for ^{41}Ar ($3 \times 10^{-6} \mu\text{Ci/mL}$) for one year in either case, the calculated exposure of a worker in the reactor bay for 2000 hours in a year is within limits.

A.2.3 Production of ^{41}Ar from Coolant Water

The reactor tank water surface is open to the reactor bay, and ^{41}Ar activity in the reactor tank water results from irradiation of the naturally occurring ^{40}Ar as a fraction of air dissolved in the water. The activated argon is transported from the core to the pool, with a removal time constant (λ_{eff}^{core}) that is a combination of the radioactive decay constant (λ) and the fractional removal rate from the total convection flow through the core (\dot{w}_{core} , the product of the flow rate per element – a function of power – and 83 - the number of elements - divided by the volume of water in the core (V_{core}):

$$\lambda_{eff}^{core} = \lambda + \frac{\dot{w}_{core}}{V_{core}} \quad (5)$$

The activated argon is then transported to the reactor bay atmosphere, with a removal time constant (λ_{eff}^{bay}) that is a combination of the radioactive decay constant (λ) and the fractional removal rate from the reactor bay (effluent flow rate \dot{w}_{bay} divided by reactor bay air volume V_{bay}):

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$$\lambda_{eff}^{bay} = \lambda + \frac{\dot{W}_{bay}}{V_{bay}} \quad (6)$$

The rate of change in the ⁴¹Ar concentration (number of atoms cm⁻³) in the core is a function of activation of ⁴⁰Ar and removal of ⁴¹Ar through a combination of radioactive decay (λ) and flow):

$$\frac{dN_{core}^{41}}{dt} = \Sigma^{40} \cdot \Phi_{th}^{Core} - \lambda_{eff}^{core} \cdot N^{41} \quad (7)$$

At equilibrium ($dN/dt = 0$), the concentration of ⁴¹Ar (atoms cm⁻³) in the core is calculated using equation (7) as:

$$N_{core}^{41} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core}} \quad (8)$$

And the total core equilibrium inventory of ⁴¹Ar is:

$$I_{core}^{eq} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core}} \cdot V_{core} \quad (9)$$

Equilibration occurs rapidly because of the large λ_{eff} and time dependent behavior $[1 - e^{-\lambda_{eff}^{core} t}]$. The rate of ⁴¹Ar inventory leaving the core (\dot{I}_{core}^{eq}) at equilibrium is the product of ⁴¹Ar volume averaged activity in the core (I_{core}^{eq}/V_{core}) and volumetric flow rate (\dot{W}_{core}):

$$\dot{I}_{core}^{eq} = \frac{I_{core}^{eq}}{V_{core}} \cdot \dot{W}_{core} = I_{core}^{eq} \cdot \frac{\dot{W}_{core}}{V_{core}} \quad (10)$$

Substituting the equality of equation (8) into equation (10):

$$\dot{I}_{core}^{eq} = \left(\frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core}} \cdot V_{core} \right) \cdot \frac{\dot{W}_{core}}{V_{core}} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core}} \cdot \dot{W}_{core} \quad (11)$$

Not all of the ⁴¹Ar activity that leaves the core reaches the surface; some ⁴¹Ar decays because of the time required to transport the nuclide from the core 16 feet below the surface of the pool. Time delay in transport is calculated as the volume flow rate exiting the core divided by the total volume (vol) through which the material is transported; where vol is 3.76 x 10⁶ cm³, the volume of water contained in the column 6.5 ft (99.06 cm) in diameter 16 ft (4.88 x 10² cm) between the core exit and the pool surface. Transport time is a few seconds and decay can be neglected. Assuming 100% transfer from the pool to the bay atmosphere, the rate of inventory removal from the core through the pool is the rate of addition of ⁴¹Ar to the reactor bay atmosphere:

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$$\dot{I}_{core}^{eq} = \dot{I}_{bay}^{eq} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core}} \cdot \dot{W}_{core} \quad (12)$$

The change in total radionuclide inventory (I) in the reactor bay is calculated in a similar manner as the core, using the rate of contributions to inventory released into the reactor bay atmosphere (essentially all of the inventory removed from the core) and the rate of removal from radioactive decay and transport via reactor bay exhaust:

$$\frac{dI_{bay}}{dt} = \dot{I}_{bay} - \lambda_{eff}^{bay} \cdot I_{bay} \quad (13)$$

Using equation (10) and equation (11):

$$\frac{dI_{bay}}{dt} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core}} \cdot \dot{W}_{core} - \lambda_{eff}^{bay} \cdot I_{bay} \quad (14)$$

For equilibrium conditions (i.e., $dI_{bay}/dt = 0$),

$$I_{bay}^{eq} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core} \cdot \lambda_{eff}^{bay}} \cdot \dot{W}_{core} \quad (15)$$

The concentration of activity in the reactor bay is determined by the concentration in the bay (I/V_{bay}) and the decay rate (λ for ^{41}Ar); based on equation (15), the specific activity in the reactor bay atmosphere (A_{bay}^{eq}) is:

$$A_{bay}^{eq} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core} \cdot \lambda_{eff}^{bay}} \cdot \dot{W}_{core} \left(\frac{\lambda}{V_{bay}} \right) \quad (16)$$

Note that if ventilation is secured, $\lambda_{bay} = 0$, and equation (15) becomes:

$$A_{bay}^{eq} = \frac{\Sigma^{40} \cdot \Phi_{th}^{Core}}{\lambda_{eff}^{core}} \cdot \frac{\dot{W}_{core}}{V_{bay}} \quad (17)$$

Using equation (16), the specific activity of ^{41}Ar in reactor bay air is 0.378 Bq ml^{-1} ($1.02 \times 10^{-5} \mu\text{Ci ml}^{-1}$). With a Derived Air Concentration for ^{41}Ar of $3 \times 10^{-6} \mu\text{Ci/ml}$, if the reactor is operated continuously at full power for a year then it will be necessary to assure that an individual does not occupy the reactor bay during operations for more than 600 working hours; not limiting in a practical sense as (1) the assumed operating schedule is unrealistically conservative and (2) the reactor bay is not routinely occupied continuously during high power operations.

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A more realistic approach possible within facility constraints considers a series of irradiations followed by decay until the start of the next irradiation. In this approach, the buildup to equilibrium ⁴¹Ar is a significant consideration. Buildup to saturation activity is asymptotic:

$$A(t) = A_{\infty} \cdot (1 - e^{-\lambda \cdot t})$$

In the core, the dominance of the large flow term (compared to core volume) for the effective time constant drives core activity to saturation very quickly, while the exchange rate of reactor bay air has the same order of magnitude as the radiological time constant for ⁴¹Ar. Consequently, only the effective time constant for the reactor bay needs to be considered in determining time dependent behavior:

$$A_1(t) = A_{bay}^{eq} \cdot (1 - e^{-\lambda_{eff}^{bay} \cdot t})$$

Reactor bay activity over the decay interval between termination of operations (T₁) and initiation of the next operation is:

$$A_2(t) = A_{bay}^{eq} \cdot (1 - e^{-\lambda_{eff}^{bay} \cdot T_1}) \cdot e^{-\lambda_{eff}^{bay} \cdot t}$$

The average activity over the irradiation and decay intervals is the integral of the two intervals over the total time:

$$\begin{aligned} \frac{\bar{A}}{A_{bay}^{eq}} &= \frac{1}{T_2} \left\{ \int_0^{T_1} (1 - e^{-\lambda_{eff}^{bay} \cdot t}) dt + (1 - e^{-\lambda_{eff}^{bay} \cdot T_1}) \cdot \int_{T_1}^{T_2} e^{-\lambda_{eff}^{bay} \cdot t} dt \right\} \\ \frac{\bar{A}}{A_{bay}^{eq}} &= \frac{1}{T_2 \cdot \lambda_{eff}^{bay}} \left\{ \lambda_{eff}^{bay} \cdot T_1 + e^{-\lambda_{eff}^{bay} \cdot T_1} - 1 - \left(e^{-\lambda_{eff}^{bay} \cdot T_1} - 1 \right) \cdot \left(e^{-\lambda_{eff}^{bay} \cdot T_1} - e^{-\lambda_{eff}^{bay} \cdot T_2} \right) \right\} \end{aligned}$$

Since the radiological time constant is the same order of magnitude as the effective time constant for ⁴¹Ar removal from the reactor bay, the results are not very sensitive to changes in reactor bay flow rate. For periodic irradiation and decay on a 24-hour cycle, T₂ = 24 h. Figure A.1 shows how the fraction of saturation activity is affected by a single 24 hour irradiation cycle; for one year of daily irradiations on the order of 8 hours, the DAC is met. If the irradiation cycle is 5 days per week instead of seven (71% of available time), the DAC will be met with irradiations on the order of 12 hours each day.

Since fuel inventory, staffing, and operating practices limit (1) continuous full power operations and (2) continuous occupancy of the reactor bay, access to the reactor bay does not require additional controls to ensure individuals remain within exposure limits based on DAC values. Administrative requirements on reporting and experiment review are adequate to identify approaches to limiting conditions of exposure.

24 Hour Interval Irradiation & Decay Activity vs Irradiation Time

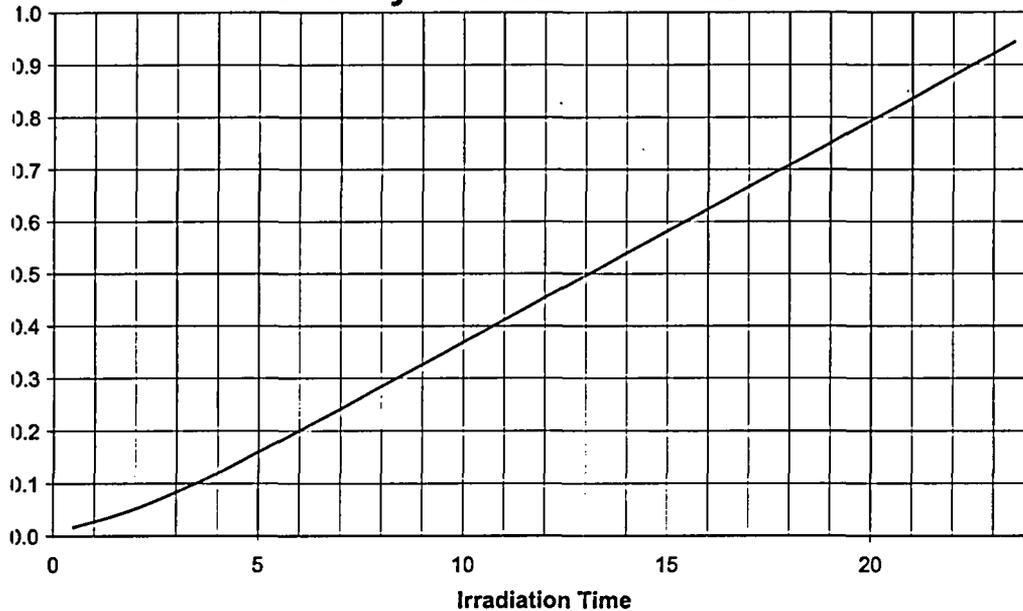


Figure A.1, 24 Hour Interval Irradiation & Decay Activity vs Irradiation Time

Note that setting $\dot{w}\lambda_{\text{bay}}$ to zero shows the effect on concentration with ventilation secured. Although the exhaust fan operates to maintain a negative pressure in the reactor bay when the reactor is not secured, an upper bound for effluent contamination of 0.746 Bq ml^{-1} ($2.01 \times 10^{-5} \text{ } \mu\text{Ci ml}^{-1}$) is calculated using equation (17), assuming no ventilation (i.e., $\lambda_{\text{bay}}=0$).

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

Although there are three modes of ^{41}Ar production, only the release of radioactive argon dissolved in water occurs routinely. The ^{41}Ar produced in the reactor bay during normal operations is released to the atmosphere via an exhaust fan at approximately $h = 11$ meters above grade. Where Q is the rate of radionuclide release, \bar{u} is the mean wind speed (m s^{-1}), $e = 2.718$, and C_y and C_z are diffusion parameters in the crosswind and vertical directions respectively, the maximum downwind concentration (pCi cm^{-3}), at grade, may be computed using the Sutton formula (Slade 1968):

$$C_{\text{max}} = \frac{2 \cdot Q}{e \cdot \pi \cdot \bar{u} \cdot h^2} \cdot \frac{C_z}{C_y} \quad (18a)$$

In terms of the concentration of the radionuclide in the reactor bay:

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$$C_{\max} = \frac{2 \cdot A_{bay}^{eq} \cdot \dot{w}}{e \cdot \pi \cdot u \cdot h^2} \cdot \frac{C_z}{C_y} \quad (13a)$$

The ratio of the maximum concentration in specific meteorological conditions to the concentration at the point of release is therefore:

$$\frac{C_{\max}}{A_{bay}^{eq}} = \left[\frac{2 \cdot \dot{w}}{e \cdot \pi \cdot h^2} \right] \cdot \frac{C_z}{C_y \cdot u} \quad (19)$$

The maximum concentration downwind occurs at distance d (m) given by:

$$d = (h / C_z)^{\frac{2}{2-n}}, \quad (20)$$

in which the parameter n is associated with the wind stability condition. The McClellan AFB SAR provides parameters associated with Pasquill stability classifications (n , C_y) and data from Chapter 2 were used to infer mean wind speeds (\bar{u}) by stability class (Class A, C, E, and G). The McClellan (site-specific) data has been modified to provide data for Class B, D, and F; graphical representation of standard deviations of material in a plume from *Meteorology and Atomic Energy* (1968, D. H. Slade), UASAEC and Regulatory Guide 1.111 show predictable, uniform behavior across each stability class in the region of interest. Therefore the McClellan n , C_z and C_y data was used to derive the data for the remaining classifications by fitting to equations (assigning $x=1$ to Class A, $x=2$ to Class B, etc.).

$$n = 0.00125 \cdot x^3 - 0.0075 \cdot x^2 + 0.03875 \cdot x + 0.1675 \quad (21)$$

$$C_y = -0.01 \cdot x^3 + 0.1287 \cdot x^2 - 0.465 \cdot x + 0.6562 \quad (22)$$

$$C_z = -0.001 \cdot x^3 + 0.02 \cdot x^2 - 0.1465x + 0.4375 \quad (23)$$

The product of (1) the maximum concentration for a classification and (2) the frequency of occurrence of the classification is the average concentration at the receptor location (where the maximum concentration occurs), with other locations bounded by this calculation.

Table A.3, Atmospheric Dispersion and Dose Calculations

Stability Class	\bar{u}	n	C_y	C_z	D	C / A_{bay}^{eq}	freq	$C / A_{bay}^{eq} \cdot f$
	m s ⁻¹		m ^{n/2}	m ^{n/2}	M			
Class A	1.600	0.200	0.31	0.31	53	5.0E-04	0.61%	3.1E-06
Class B	3.205	0.230	0.16	0.22	84	3.5E-04	4.50%	1.6E-05
Class C	4.000	0.250	0.15	0.15	134	2.0E-04	10.63%	2.5E-05
Class D	4.074	0.282	0.22	0.11	224	9.9E-05	53.21%	5.3E-05
Class E	3.500	0.330	0.30	0.075	393	5.8E-05	10.60%	6.1E-06
Class F	2.382	0.401	0.34	0.051	830	5.1E-05	11.82%	6.0E-06
Class G	0.770	0.500	0.28	0.035	2137	1.3E-04	8.64%	1.1E-05

The maximum product of C/A_{bay}^{eq} and the frequency of occurrence is 5.3×10^{-5} ; the annual dose from exposure at that location is 7.5 mrem ($5.3 \times 10^{-5} * 20.1 \text{ pCi} * 0.803 \text{ mrem h}^{-1} \text{ pCi}^{-1} * 8760 \text{ h y}^{-1}$), less than the permitted annual dose from gaseous effluents of 10 mrem .

The assumption of continuous full power operation to establish equilibrium ^{41}Ar concentration is extremely conservative, since (as previously noted) staffing and fuel inventory will not support continuous full power operations. In addition, administrative requirements on reporting and experiment review are adequate to ensure average annual discharge meets effluent limits.

A.2.5 Radiological Assessment of ^{16}N 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 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

$$C_N = \frac{\phi_f * C_O * \sigma_{n,p}}{\lambda_{16}} * (1 - e^{-\lambda t}) \quad (20)$$

where time in the core is represented by t . Fast-neutron flux varies linearly with reactor power. Time in core is a function of convection flow rate, a function of reactor power (see Chapter 4). As power increases, the rate of production increase from increased neutron flux is mitigated by a reduced time in the core from the increase in core cooling flow rate.

As the warmed coolant leaves the core, it passes through 1.5-in diameter ($A_{gp} = 11.4 \text{ cm}^2$) channels in the upper grid plate, but the triflute upper end fixture of the fuel element restricts the flow. This leaves a flow area for each element of:

$$A_0 \cong A_{gp} * \left[1 - \left(\frac{3}{\pi} \right) * \sin 30^\circ * \cos 30^\circ \right] = 6.69 \text{ cm}^2 \quad (21)$$

Operation at power requires primary cooling; primary cooling enters the pool through a flow diverter approximately 2 feet (61 cm) above the core exit, 14 feet (427 cm) below the pool surface. Core exit is at 16 feet (488 cm) below the pool surface. The flow diverter induces mixing and avoids the direct rise from the core to the pool surface (which could otherwise occur through a chimney effect from core heating). A rough estimate of hydraulic diameter of the core exit (based on total flow area) is about 13 cm; calculations show the contributions to total dose rates at the pool surface are negligible at 160-200 cm below the surface of the pool, 22-25 times the hydraulic diameter of the exit into the pool. Exit flows are a small fraction of mixing flow, and under these conditions it is considered adequate to use a nuclide concentration reduced by the ratio of the total core exit surface area (approximately 555 cm^2 for 83 elements) and the pool (with a total surface area of approximately 30900 cm^2); mixing reduces the concentration of ^{16}N from the core exit by 0.018. Therefore, concentration of the radionuclide used in calculation is reduced from core exit by dilution.

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Because of the short ½ life, the concentration of ¹⁶N is also reduced by decay during transit. Since it is difficult to characterize flow velocity field from core exit to total mixing, flow rate from the core to the surface is conservatively assumed as core exit flow rate for dose rate calculations.

Dose rate calculations were modeled as a set of disk sources, each disk containing the appropriate volume source term multiplied times the difference between the disk locations. The appropriate volume source strength for each disk source calculation was modified by exponential decay of ¹⁶N, with the time element calculated from core exit surface area, flow rate, and distance from the core exit. Dose rate calculations were based on the two major emissions, 6.13 MeV (69%) and 7.11 MeV (5%). Total dose rate at each disk (where x is the distance from the disk to the pool surface) was therefore calculated as:

$$\dot{D} = \sum_E \left[\frac{k(E) * E * S_v \Delta d}{2} * \sum_i A_i * (E_1(\mu_i x) - (E_1(\mu_i x * \sec \theta))) \right] \quad (22)$$

Where:

- $k(E): \frac{R * h^{-1}}{MeV * cm^{-1} * s^{-1}}$
- $S_v = S_0(flow) * \exp\left(-\lambda * \frac{x * A_{channel}}{\dot{m}_{H_2O} * \rho_{H_2O}^{-1}}\right)$ (see Chapter 4 for coolant flow rate)
- A_i Taylor buildup factor
- μ_i linear attenuation coefficient, modified by Taylor buildup factor a_i
- $\theta = \arctan\left(\frac{x}{R_{pool}}\right)$

Parametric variation on the distance between the disk sources showed little improvement in convergence for separations smaller than 2 cm, and essentially no improvement below 1 cm; therefore ½ cm was used for final calculations. Locations of interest for dose calculations include 30 cm (1 ft) above the pool surface (i.e., pool surface monitor), waist high (approximately 130 cm/51 in. above the pool, 100 cm/39 in. above the bridge), and at the ceiling over the pool.(549 cm/18 ft above the pool).

Only a small proportion of the ¹⁶N atoms present near the tank surface are actually 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.

Table A.4, Dose Rate (mR h⁻¹) Above Pool

KW	30 cm	130 cm	549 cm
50	0.5	0.2	0.0
100	3.5	1.1	1.3
200	15.8	4.7	0.5
300	35.7	10.5	1.2
400	60.0	17.5	1.9
500	87.2	25.3	2.8
750	166.3	47.7	5.2
1000	255.3	72.8	7.9
1250	347.9	98.8	10.6

A.3 Bibliography

EPA Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," U.S. Environmental Protection Agency, Report EPA-520/1-88-020, 1988.

Kansas State University TRIGA Mark II Reactor Hazards Summary Report, by R.W. Clack, J.R. Fagan, W.R. Kimel, and S.Z. Mikhail, License R-88, Docket 50-188, 1961.

"*Analysis of Certain Hazards Associated with Operation of the Kansas State University TRIGA Mark II Reactor at 250 kW Steady State and with Pulsed Operation to \$2.00*," by R.W. Clack, et al., and the Safety Evaluation by the U.S. Atomic Energy Commission Division of Reactor Licensing, License R-88, Docket 50-188, 1968.

Operations Manual, KSU TRIGA Mark II Nuclear Reactor Facility, License R-88, Docket 50-188.

Facility Safety Analysis Report, Rev. 2, McClellan Nuclear Radiation Center Reactor, April 1998.

Mittl, R.L. and M.H. Theys, "N-16 Concentrations in EBWR," *Nucleonics*, March 1961, p. 81. Slade, D.H. (ed.), "Meteorology and Atomic Energy," Report TID-24190, U.S. Atomic Energy Commission, 1968).