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Final Report

M. E. Battat
R. J. Henninger
J. L. Macdonald
D. J. Dudziak

University of California

LOS ALAMOS SCIENTIFIC LABORATORY

Post Office Box 1663 Los Alamos, New Mexico 87545

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FINAL REPORT

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ABSTRACT

Results of shielding calculations for the Intense Neutron Source (INS) facility are presented. The INS facility is designed to house two sources, each of which will produce D-T neutrons with intensities in the range from 1 to 3×10^{15} n/s on a continuous basis. Topics covered include the design of the biological shield, use of two-dimensional discrete-ordinates results to specify the source terms for a Monte Carlo skyshine calculation, air activation, and dose rates in the source cell (after shutdown) due to activation of the biological shield.

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I. INTRODUCTION

The Intense Neutron Source¹ (INS) facility is designed to provide a continuously operating source of D-T neutrons with intensities in the range from 1 to 3×10^{15} n/s. These intense sources, two of which are planned, will be achieved by intersecting an accelerated tritium ion beam with a supersonic jet of deuterium gas. The inside dimensions of each source cell are 7.32 m square by 5.49 m high. Although some of the results to be quoted here have been reported previously,^{2,3} the purpose of this document is to set forth in detail the calculational procedures and techniques employed, specify the elemental compositions of the materials used, and report on recent work on this problem. It is hoped that the results and/or methods to be discussed will be useful in future shielding problems involving D-T fusion neutrons.

II. BIOLOGICAL-SHIELD CALCULATIONS

The nominal thickness required for the biological shield was computed assuming a 1-cm³ source of 3×10^{15} D-T n/s located in a spherical enclosure with radius equal to the perpendicular source-to-wall distance (3.66 m) in the actual source cell. This air-filled enclosure was surrounded by spherical concrete shields of selected thicknesses. Discrete-ordinates multigroup calculations were then

made to obtain the neutron and gamma-ray fluxes, and hence dose rates, at points external to the shield.

In addition to providing an intense source of D-T neutrons for materials research and development, the INS will be designed to yield a neutron spectrum similar to that anticipated for the first-wall spectra of proposed fusion reactors. This is achieved by surrounding the D-T source with a "spectrum-tailoring blanket," consisting of concentric spheres of lithium, highly enriched uranium, and beryllium.⁴ Source multiplications in the range from 5 to 10 are typical for proposed blanket designs. In order to insure the adequacy of the shielding under all conditions, calculations were performed for both the "blanket in" and "blanket out" configurations. As will be discussed in a later section of this report, the biological-shield design calls for a 0.3-m-thick inner liner of borated gypsum. The purpose of such a liner is to minimize the induced activities in this portion of the shield and thus permit earlier access to the source cell after shutdown. Compared to a all-concrete shield, the effect of this borated gypsum liner is a reduction of about 30% in the dose rates external to the shield.

A. Composition of Concrete, Gypsum, and Air

The mix proportions specified for the ordinary concrete in the biological shield were those used

recently for a large construction project at the Los Alamos Scientific Laboratory (LASL). These proportions (kg/m^3) were as follows:

cement - 335,
water - 163,
sand - 801, and
gravel - 1 032.

This gives a fresh concrete density of $2\,331\text{ kg/m}^3$. In calculating the attenuation of 1 to 15-MeV neutrons in concrete shields, the water content, and hence the H content, is an important parameter. Lacking more detailed information on water retention in cured thick-concrete shields, we assumed the lowest value which seemed reasonable, namely, that half of the mix water is retained. Based on this assumption, the density of cured concrete is $2\,250\text{ kg/m}^3$ with a water content of 82 kg/m^3 . Values quoted for the total water content in concrete range from 110 to 140 kg/m^3 for normal curing conditions;⁵ the measured value for 0.101 6-m-thick slabs at room temperature is reported⁶ as 130 kg/m^3 . Thus, the water content we assumed is low and will cause us to err on the safe side. For a 3.35-m-thick shield, the calculated effect of assuming all of the mix water to be retained, and correcting for the resulting change in density, is a reduction factor between 2 and 3 in the dose rate external to the shield.

Elemental densities for ordinary concrete, using the above mix proportions, were inferred from analyses of cement and aggregate available to contractors in the Los Alamos area. The elemental and atom densities for the concrete used in our calculations, designated as "LASL concrete," are shown in Table I. For future reference, we give the same information for the Tower Shielding Facility (TSF) concrete used in similar calculations by Roussin, Allsmiller, and Barish.⁷ Also shown in Table I are the constituents of pure gypsum ($\text{CaSO}_4\text{H}_2\text{O}$) with a density of $1\,900\text{ kg/m}^3$. Note that the pure gypsum was used in the neutronics calculations for the bulk shielding; however, for the activation studies, the gypsum impurities were included. Air densities were computed assuming an air composition of 79 vol% nitrogen and 21 vol% oxygen. Listed below are the constituent densities appropriate to the altitude

(2.2 km) at Los Alamos:

Nitrogen - $0.742\text{ kg/m}^3 - 3.191 \times 10^{-5}\text{ (barn-cm)}^{-1}$,
Oxygen - $0.221\,7\text{ kg/m}^3 - 8.346 \times 10^{-6}\text{ (barn-cm)}^{-1}$.

B. Calculational Procedure and Cross-Section Data

All of the bulk-shielding calculations were performed with the ONETRAN-DA discrete-ordinates code⁸ in the S_8P_3 approximation and for a spherical geometry. A coupled set of multigroup cross sections, 30 neutron and 12 gamma-ray groups, was used in the calculations. The energy boundaries and flux-to-dose-rate factors for this 42-group set are shown in Tables II and III. For H, N, O, Al, Si, Fe, and B, the cross sections were generated from ENDF/B-IV data by the Nuclear Data group at LASL. Cross sections for C, Na, Mg, K, Ca, ^{235}U , and ^{238}U were extracted from a LASL Library used for weapons effect calculations.⁹ Aluminum cross sections were substituted for sulfur because it was not present in the libraries used. The flux-to-dose-rate factors listed in Tables II and III were derived from data given in a recent American National Standard.¹⁰ Flat flux weighting was assumed within each group in deriving these factors. Comparison calculations were also made for two selected configurations using the cross sections above (denoted as "LASL" cross sections) and a preliminary few-group library issued for testing purposes by Subcommittee ANS-6 of the American Nuclear Society (ANS) Standards Committee. This cross-section set was issued in March 1977 by working group ANS-6.1.2 of ANS-6 and distributed by the Radiation Shielding Information Center (RSIC) under the designation PDS-45/ANS612. This library is a 45 neutron/16 gamma-ray set (P_0 to P_3) for which the nuclides were self-shielded assuming a homogeneous concrete medium. As before, the flux-to-dose-rate factors were based on the data in Ref. 10.

C. Summary of Results

The geometry specified for these calculations was an air-filled spherical enclosure with a radius of 3.66 m, at the center of which was located a 0.7-cm-radius source of $3 \times 10^{15}\text{ D-T n/s}$. Dose rates were computed for 3.05- and 3.35-m-thick shields of ordinary concrete surrounding the enclosure. For reasons which will be discussed later, the air region outside the shield extended to a radius of 1 000

TABLE I
CONCRETE AND GYPSUM COMPOSITIONS

Element	LASL Concrete		TSF Concrete		Pure Gypsum	
	Density (kg/m ³)	Atom Density (barn·cm) ⁻¹	Density (kg/m ³)	Atom Density (barn·cm) ⁻¹	Density (kg/m ³)	Atom Density (barn·cm) ⁻¹
H	10.2	6.094 E-03 ^a	14.2	8.50 E-03	45.	2.658 E-02
C	-	-	408.8	2.050 E-02	-	-
O	1 153.	4.342 E-02	943.1	3.550 E-02	1 059.	3.988 E-02
Na	26.	6.811 E-04	-	-	-	-
Mg	8.7	2.156 E-04	75.1	1.86 E-03	-	-
Al	80.0	1.786 E-03	26.9	6.0 E-04	-	-
Si	811.	1.739 E-02	79.3	1.70 E-03	-	-
S	-	-	-	-	354.	6.646E-03
K	32.0	4.929 E-04	-	-	-	-
Ca	98.0	1.472 E-03	752.0	1.130 E-02	442.	6.646E-03
Fe	31.0	3.343 E-04	17.6	1.9 E-04	-	-
	2 249.9		2 317.0		1 900.	

^aRead as 6.094×10^{-3}

meters. Because 3.35-m-thick ordinary concrete walls will be required for the source cells in order to meet the specified 100 mrem/occupational-year near the outer surface of the shield, extensive calculations were performed for this shield thickness. An occupational year is defined as 40 h/wk for 50 wk/yr or 2 000 h/yr.

For a 3.05-m-thick concrete shield and a source of 3×10^{15} D-T n/s, the computed dose rates at a radius of 7.7 m (1 m away from the outer shield surface) were:

Total = 561,
Neutron = 57, and
Gamma = 504 mrem/occupational year.

Calculations for the 3.35-m-thick shield were performed for both the "blanket in" and "blanket out" cases; problems were also run for inner liners of gypsum with 0.25 and 1.0 wt% boron. The blanket configuration used for the calculations is given in Table IV; the calculated source multiplication for this blanket was 6.3. Table V summarizes the results obtained for the various blanket configurations. A plot of dose rate vs radius for the 3.35-m-

thick concrete shield is shown in Fig. 1. It is seen that the external dose rates with the blanket

Fig. 1. Dose rate vs radius for a 3×10^{15} D-T n/s source in a 3.66-m-radius spherical enclosure and 3.35-m-thick ordinary concrete shield.

TABLE II
NEUTRON FLUX-TO-DOSE RATE CONVERSION FACTORS

E-Upper (MeV)	E-Lower (MeV)	(rem/h) (n/cm ² ·s)
1.700 +01	1.500 +01	2.15 -04
1.500 +01	1.350 +01	2.08 -04
1.350 +01	1.200 +01	1.89 -04
1.200 +01	1.000 +01	1.62 -04
1.000 +01	7.790 +00	1.47 -04
7.790 +00	6.070 +00	1.48 -04
6.070 +00	3.680 +00	1.51 -04
3.680 +00	2.865 +00	1.36 -04
2.865 +00	2.232 +00	1.27 -04
2.232 +00	1.738 +00	1.27 -04
1.738 +00	1.353 +00	1.28 -04
1.353 +00	8.230 -01	1.29 -04
8.230 -01	5.000 -01	1.06 -04
5.000 -01	3.030 -01	7.59 -05
3.030 -01	1.840 -01	4.84 -05
1.840 -01	6.760 -02	2.67 -05
6.760 -02	2.480 -02	1.18 -05
2.480 -02	9.120 -03	5.38 -06
9.120 -03	3.350 -03	3.57 -06
3.350 -03	1.235 -03	3.65 -06
1.235 -03	4.540 -04	3.79 -06
4.540 -04	1.670 -04	3.97 -06
1.670 -04	6.140 -05	4.16 -06
6.140 -05	2.260 -05	4.35 -06
2.260 -05	8.320 -06	4.49 -06
8.320 -06	3.060 -06	4.57 -06
3.060 -06	1.130 -06	4.54 -06
1.130 -06	4.140 -07	4.39 -06
4.140 -07	1.520 -07	4.10 -06
thermal		3.67 -06

in are a factor of 3 less than for the "blanket out" case. Also, the replacement of the inner 0.305 m of concrete with pure or boroated gypsum reduces the dose rates outside the shield by 15 to 30 percent. However, the addition of 0.25 wt% or 1.0 wt% boron to the gypsum reduces the thermal flux in the liner by a factor of about 10 and 100, respectively.

D. Effects of Density and Water Content on Concrete Shielding

The calculations described above were made with 2 250 kg/m³ ordinary concrete and composition as

TABLE III
GAMMA-RAY FLUX-TO-DOSE-RATE CONVERSION FACTORS

E-Upper (MeV)	E-Lower (MeV)	(rem/h) (gamma/cm ² ·s)
1.000 +01	9.000 +00	9.15 -06
9.000 +00	8.000 +00	8.41 -06
8.000 +00	7.000 +00	7.67 -06
7.000 +00	6.000 +00	6.93 -06
6.000 +00	5.000 +00	6.20 -06
5.000 +00	4.000 +00	5.41 -06
4.000 +00	3.000 +00	4.62 -06
3.000 +00	2.000 +00	3.71 -06
2.000 +00	1.000 +00	2.62 -06
1.000 +00	5.000 -01	1.60 -06
5.000 -01	1.000 -01	7.41 -07
1.000 -01	1.000 -02	5.64 -07

given in Table I. We first obtained the effect of increasing this density to 2 320 kg/m³, an increase of 3.1 percent. This change in density was made by increasing the elemental densities by the latter amount. For a 3.35-m-thick concrete shield, this 3.1% increase resulted in a factor of 2 decrease in the external dose rates. The next change made was to take the 2 250 kg/m³ concrete and add that half of the mix water that was assumed lost in the original specifications. As shown in Sec. II.A, this amounts to the addition of 81.5 kg/m³ of water;

TABLE IV
BLANKET CONFIGURATION (SPHERICAL GEOMETRY) SPECIFIED FOR INS SHIELDING CALCULATIONS

Zone	Material	Outer Radius (cm)
1	void - source	0.7
2	void	1.25
3	Fe	1.4
4	void	2.0
5	Fe - sample	3.0
6	Li	4.0
7	50 vol% Cu + 50 vol% H ₂ O	4.1
8	U (93 at.% 235)	5.3
9	50 vol% Cu + 50 vol% H ₂ O	5.4
10	Be	35.4
11	B ₄ C	35.9

TABLE V
DOSE RATES IN mrem/OCCUPATIONAL yr AT A RADIUS OF 8.04 m (1 m FROM OUTER
SURFACE OF SHIELD) FOR A SOURCE OF 3×10^{15} D-T n/s

Shield	Total	With Blanket		Total	Without Blanket	
		Neutron	Gamma		Neutron	Gamma
3.35 m concrete	27.1	0.3	26.8	83	5	78
0.305 m gypsum + 3.05 m concrete	22.8	0.2	22.6	63	5	58
0.305 m gypsum (0.25 wt% B) + 3.05 m concrete	20.7	0.2	20.5	58	6	52
0.305 m gypsum (1 wt% B) + 3.05 m concrete	20.5	0.2	20.3	57	5	52

with this addition the concrete density becomes $2\ 330\ \text{kg/m}^3$. The result of this calculation gave dose rates outside the shield that were a factor of 5.3 lower than the $2\ 250\ \text{kg/m}^3$ concrete. To a good approximation, we can attribute a factor of 2 of the 5.3 to a density change, thus giving a factor of about 2.5 attributable to the added water content. Finally, we made similar calculations using the TSF concrete listed in Table I. The net result was a reduction factor of 7.3 in the external dose rates as compared to the LASL concrete. Again, correcting for the change in density, the TSF concrete appears to provide an attenuation a factor of 3.7 higher than the LASL concrete. Undoubtedly, some of this is due to the higher hydrogen content in the TSF concrete, with the rest assigned to the difference in concrete compositions. Calculations by Chilton¹¹ indicate that siliceous types of concrete (LASL) are not as effective as carbonaceous types (TSF) for the same water content and areal density; our results are in qualitative agreement with these observations.

E. Cross-Section Comparison Calculations

Two test problems were specified in order to compare calculated results obtained with the "LASL" cross section set and the PDS-45/ANS612 library. As before, the ONETRAN-DA code in the S_8P_3 approximation was used. The convergence criterion was 10^{-4} and the geometry was spherical. The concrete (LASL) composition is shown in Table I; the air composition is given in Sec. II.A. Dose rates were computed for a 10^{15} D-T n/s source. Note the change from the 3×10^{15} n/s source used in the design calculations. For reference purposes, we show in Table VI the energy group structure for this 45 x 16 coupled set

and the flux-to-dose-rate factors we used. In the event that others might wish to use these test problems for cross-section checking purposes, details of the geometry are given in Table VII as are the computed dose rates. As can be seen from Table VII, the two libraries yield dose rates differing at most by $\pm 20\%$, which is very good agreement for this type of calculation.

III. SKYSHINE CALCULATIONS

Because neutrons penetrating the roof of the INS source cell contribute to the dose rate at ground level only by air scattering, the roof can be thinner than the cell walls. The adequacy of a given roof thickness was determined using coupled discrete-ordinates and Monte Carlo calculations. Upper bounds to the skyshine dose rates were obtained from results of calculations for a spherical configuration with air or vacuum outside the shield (Sec. III.A). The difference of the two computed dose rates (air minus vacuum) is a measure of the maximum skyshine contribution. It is for this reason that an air region extending to 1 000 m was specified for the calculations in Sec. II above. The skyshine dose rate thus inferred is an overestimate because the source is surrounded by a shield of constant thickness in the spherical problems. In the actual case, with a flat roof, the average shield thickness is greater because of slant penetration effects. In earlier calculations, a first-order correction was applied to the upper-bound values to account for the latter effect.³ However, in light of the later two-dimensional discrete-ordinates calculations this correction was found to be unsatisfactory and its use is not recommended.

TABLE VI
FLUX-TO-DOSE-RATE CONVERSION FACTORS USED BY LASL FOR THE (45,16) COUPLED
CROSS-SECTION LIBRARY FOR ANS-6.1.2 STUDIES

Neutron Groups	$E_{\max} = 17.33 \text{ MeV}$	Gamma-ray Groups	$E_{\max} = 14 \text{ MeV}$
	(rem/h) ($n/cm^2 \text{ s}$)		(rem/h) ($\text{Gamma}/cm^2 \text{ s}$)
<u>E-Lower (MeV)</u>		<u>E-Lower (MeV)</u>	
1.492 E+01	2.16 E-04	1.000 E+01	1.10 E-05
1.419 E+01	2.10 E-04	8.000 E+00	8.78 E-06
1.350 E+01	2.06 E-04	7.000 E+00	7.67 E-06
1.000 E+01	1.74 E-04	6.000 E+00	6.93 E-06
7.408 E+00	1.47 E-04	5.000 E+00	6.20 E-06
6.065 E+00	1.48 E-04	4.000 E+00	5.41 E-06
4.966 E+00	1.54 E-04	3.000 E+00	4.62 E-06
4.066 E+00	1.51 E-04	2.500 E+00	3.96 E-06
3.679 E+00	1.44 E-04	2.000 E+00	3.47 E-06
2.725 E+00	1.35 E-04	1.500 E+00	2.93 E-06
2.365 E+00	1.26 E-04	1.000 E+00	2.32 E-06
2.307 E+00	1.25 E-04	7.000 E-01	1.76 E-06
2.231 E+00	1.26 E-04	4.500 E-01	1.31 E-06
1.653 E+00	1.27 E-04	3.000 E-01	9.28 E-07
1.353 E+00	1.29 E-04	1.000 E-01	5.07 E-07
8.629 E-01	1.29 E-04	1.000 E-02	5.64 E-07
8.208 E-01	1.21 E-04		
7.427 E-01	1.16 E-04		
6.081 E-01	1.08 E-04		
4.979 E-01	9.74 E-05		
3.688 E-01	8.14 E-05		
2.985 E-01	6.43 E-05		
2.972 E-01	5.81 E-05		
1.832 E-01	4.78 E-05		
1.111 E-01	3.08 E-05		
6.738 E-02	1.99 E-05		
4.087 E-02	1.34 E-05		
2.479 E-02	9.06 E-06		
2.358 E-02	7.14 E-06		
1.503 E-02	5.98 E-06		
9.119 E-03	4.15 E-06		
5.531 E-03	3.56 E-06		
3.355 E-03	3.58 E-06		
2.035 E-03	3.63 E-06		
1.234 E-03	3.69 E-06		
7.485 E-04	3.76 E-06		
4.540 E-04	3.84 E-06		
2.754 E-04	3.93 E-06		
1.670 E-04	4.03 E-06		
1.013 E-04	4.12 E-06		

TABLE VI (cont)

Neutron Groups	$E_{\max} = 17.33 \text{ MeV}$
	(rem/h)
E-Lower (MeV)	(n/cm ² ·s)
6.144 E-05	4.22 E-06
3.727 E-05	4.31 E-06
1.068 E-05	4.44 E-06
4.140 E-07	4.55 E-06
thermal	3.67 E-06

The Monte Carlo skyshine analysis involved two distinct calculational steps. First, we computed angular fluxes at the roof top, for a cylindrical geometry, using discrete-ordinates methods (Sec. III.B). These fluxes were translated into source probability distributions for the second step which was a Monte Carlo air-transport calculation to determine dose rates at selected points near ground level (Sec. III.C).

A. Upper-Limit Estimates for Skyshine Contribution

The spherical-geometry calculations with air or vacuum outside the shield were performed using the procedures described in Sec. II.B. As noted

above, this simplified approach should overestimate the dose rates. In addition, because of attenuation in the air, there will be a "critical" radius beyond which the dose rate in air will be less than that in vacuum and the difference between these two quantities will be negative. Although these negative values are physically meaningful for the problem specified, interpreting the results for detector locations near to and beyond the "critical" radius as the skyshine contribution is invalid. The difficulty stems from the fact that the direct and scattered components cannot be divorced from each other in the one-dimensional case. However, as

TABLE VII

CROSS-SECTION COMPARISON CALCULATIONS - TEST PROBLEM SPECIFICATIONS AND RESULTS - 10^{15} D-T n/s SOURCE

- (a) Test problems 1 and 2. Zone boundaries, materials, and mesh points per zone. Figures in parentheses are for problem 2.

Radius (cm) -			
0		Air - source	2
1.		Air	10
10.		Air	6
91.44		Air	10
365.76		Air	10
670.56 (548.64)	Concrete		80 (48)
700. (580.)	Air		8

- (b) Calculational results.

Test problem 1. Dose rate in rem/h at a radius of 6.83 m

	Total	Neutron	Gamma-Ray
LASL library	1.641-04	1.920-05	1.449-04
ANS612 library	1.476-04	2.295-05	1.246-04

Test problem 2. Dose rate in rem/h at a radius of 5.62 m

	Total	Neutron	Gamma-Ray
LASL library	5.811-01	2.996-01	2.815-01
ANS612 library	5.774-01	3.409-01	2.365-01

will be shown later, this method does give useful preliminary estimates for detector locations not too close to the outer surface of the shield and up to a few hundred meters from the source. For illustrative purposes, we have plotted our computed results, in Fig. 2, for a 2.13-m-thick ordinary concrete shield and a source of 10^{15} D-T n/s. Although not indicated on Fig. 2, the computed dose rate at the top surface of the roof is 50 mrem/h. At radii of 50 m and 150 m (site boundary), the dose rates are 240 and 29 mrem/occupational year, respectively. It is emphasized that these values are upper limits and are for a 10^{15} n/s source. Because of the increase in design source strength to 3×10^{15} n/s, and to limit dose rates within the site boundary to no more than 25-35 mrem/occupational year, a roof thickness greater than 2.13 m will be required. This matter will be discussed further in Sec. III.C below.

Fig. 2. Dose rates due to skyshine as inferred from discrete-ordinates calculations for 2.13-m concrete shield in spherical geometry. Outer radius of shield = 5.79 m. Source = 10^{15} D-T n/s.

B. Source Input for Monte Carlo Calculations

Discrete-ordinates calculations were made to obtain the neutron and gamma-ray angular fluxes at the surface of the roof. This information was obtained using the TWOTRAN-FC code¹² for r-z geometry and with an S_8P_3 approximation. Cross sections used were those described in Sec. II.B. The cell roof was approximated by a disk of 5.9-m radius (see Fig. 3) and 2.29 m thick. The roof consisted of 0.305 m of gypsum (1900 kg/m^3 , 1 wt% boron) followed by a top layer of 1.98 m of ordinary concrete. A point source of 3×10^{15} D-T n/s was located on the cylinder axis and 3.66 m from the bottom surface of the roof. Eighty equally spaced mesh points were specified in the radial direction. The mesh arrangement in the z direction was as follows: between source and bottom of the roof - 10, gypsum - 4, and concrete - 28. The running time, on the LASL CDC-7600 computer, for this TWOTRAN-FC problem was 140 min.

The angular fluxes obtained above were then converted to probability distributions which were used as the source input for the air-transport Monte Carlo calculations. This conversion was accomplished through the use of a special code (LINK) written for this purpose. The output of the LINK code provides

Fig. 3. Plan view of INS source cell showing disk source at top of roof.

the spatial, energy, and directional distributions of the particles which emerge from the top of the roof. The leakage spectra of these emerging particles, inferred from the TWOTRAN-FC results, is shown in Table VIII. It is noted that 92% of the neutrons which leak out are in the thermal group, and 58% of the photons are in the 0.1 to 0.5 MeV range. The leakage from a 2.13-m-thick concrete roof was also

TABLE VIII
LEAKAGE SPECTRA OF NEUTRONS AND GAMMA RAYS
EMERGING FROM TOP OF 2.29-m-THICK
(GYPSUM + CONCRETE) ROOF. ISOTROPIC SOURCE OF
 3×10^{15} D-T n/s.

Neutrons		Gamma-Rays	
Group	n/s	Group	Gamma/s
1	0.	31	2.14 E+07
2	3.84 E+05	32	4.08 E+07
3	6.76 E+05	33	3.57 E+08
4	6.23 E+05	34	3.97 E+08
5	8.85 E+05	35	3.58 E+08
6	2.37 E+06	36	7.01 E+08
7	2.00 E+06	37	8.72 E+08
8	2.49 E+06	38	1.15 E+09
9	1.22 E+07	39	1.55 E+09
10	5.90 E+06	40	1.59 E+09
11	5.03 E+06	41	9.95 E+09
12	6.42 E+06	42	4.89 E+07
13	1.09 E+07	Total	1.70 E+10
14	7.18 E+06		
15	5.42 E+06		
16	1.28 E+07		
17	1.15 E+07		
18	1.13 E+07		
19	1.13 E+07		
20	1.24 E+07		
21	1.39 E+07		
22	1.53 E+07		
23	1.69 E+07		
24	1.85 E+07		
25	2.03 E+07		
26	2.22 E+07		
27	2.40 E+07		
28	2.60 E+07		
29	2.78 E+07		
30	3.36 E+09		
Total	3.67 E+09		

computed by Celnik and Peng¹³ from one-dimensional discrete-ordinates calculations (ANISN code¹⁴) to which a correction factor was applied to account for slant penetration effects in the actual slab roof. The configuration assumed was a 3.66-m-radius source cell surrounded by 2.13 m of concrete. Leakages obtained were 4.59×10^8 n/s and 4.11×10^9 gamma rays/s. Applying an estimated factor of 3 additional attenuation for 0.16 m of concrete, the latter figures become 1.5×10^8 n/s and 1.4×10^9 gamma rays/s. Compared to the TWOTRAN-FC results, the one-dimensional corrected values are about a factor of 10 lower. Monte Carlo calculations using both these leakage spectra will be compared in Sec. III.C.

C. Monte Carlo Calculations

The Monte Carlo calculations performed at LASL employed the MCNP code¹⁵ and the calculational model shown in Fig. 4. The 14-m roof diameter was equal to the lateral outer dimensions of the source cell (3.35-m-thick walls) as shown in Fig. 3. Thus, shadowing effects for detector points close to the outer walls are accounted for in the calculations. The source input for the LASL calculations were obtained using the TWOTRAN-FC and LINK codes as described in Sec. III.B for a roof thickness of 2.29 m. Thin concrete regions were added to the cell interior and at ground level to provide for reflection of particles headed in those directions. Ring detector (cylindrically symmetric point detector) tallies¹⁵ were obtained at the six locations shown

Fig. 4. Geometry for MCNP Monte Carlo calculations. Air region extends to 2 000 m in r and z directions.

in Table IX; the contributions to the total dose rate by primary neutrons, primary photons, and secondary photons are also listed. The relative errors quoted in the table represent $\sigma(\bar{x})/\bar{x}$, namely, the relative error corresponding to one standard deviation of the mean. This may be interpreted to mean that there is a 68.3% chance that the error is no larger than the value listed. It is seen from Table IX that the neutron contribution to the total dose rate varies from 74% at the closest point to 27% at the farthest point. The primary photons account for about half of the total dose rate for detectors between 150 and 1 340 m. The secondary photons are responsible for 10 to 15% of the total dose rate, except at 1 340 m where the contribution is 32%. If the dose rates (rem/h) in Table IX are converted to annual values (see Table X), we observe that the requirement of 25-32 mrem/occupational yr (direct + skyshine) within the site boundary cannot be met with the 2.29-m-thick roof. After these calculations were made, the roof design was modified to include a "cap" of 0.3 m of concrete in order to provide cover for the instrumentation to be located on the roof and shielding for the dosimeter ports. This added concrete should decrease the skyshine dose rates by a factor of about 10. Even so, the total dose rates near the outer wall of the cell will be above the design dose-rate value. To provide shielding sufficient to meet the 25-35 mrem/occupational yr figure is not feasible. If, after the facility is in operation, a radiation

survey indicates higher than specified dose rates outside the cell, a reasonable solution would be to exercise administrative control over the area in question.

D. Other Computational Methods to Estimate Skyshine Dose Rates

Skyshine dose rates computed using other methods were compared to the MCNP results discussed in Sec. III.C. These are summarized in Table X. Dose rates are given in mrem/occupational yr within the LASL boundary and in mrem/yr for points farther out. Results shown for the 2.13-m-thick roof were obtained during the early design stage when the source strength was specified as 10^{15} n/s; hence, no Monte Carlo calculations were made at LASL for this case. The "ONETRAN-DA Maximum" figures were computed using the methodology described in Sec. III.A.

The calculational steps involved in arriving at the values listed in the column labeled "ANISN-MORSE" are discussed in Ref. 13, but a brief outline will be given here. First, a one-dimensional discrete-ordinates calculation (in spherical geometry and S_8P_3) was made using the ANISN code¹⁴ and the CASK 40-group coupled cross section set.¹⁶ The angular distribution of neutrons and gamma rays at the shield surface was taken to be representative for particles emerging at all points on the roof of the source cell. The flux, $N(0)$, at a point on the roof directly above the source cell was set equal to that computed for the spherical system. At a radius r , with respect to the latter reference point, the flux $N(r)$ was taken to be

TABLE IX
SKYSHINE DOSE RATES (rem/h) FOR A SOURCE OF 3×10^{15} D-T n/s AND A 2.29-m-THICK ROOF.
CALCULATIONS MADE WITH THE MCNP MONTE CARLO CODE - RELATIVE ERRORS SHOWN IN PARENTHESES.

Distance from Source (m)	Location	Neutron	Photon	Secondary Photon	Total
3 ^a	1 m from shield	3.34 E-04 (0.04)	5.38 E-05 (0.03)	6.49 E-05 (0.08)	4.53 E-04 (0.03)
15.4	parking lot	2.53 E-04 (0.05)	8.49 E-05 (0.08)	7.08 E-05 (0.06)	4.09 E-04 (0.04)
150	site boundary	4.26 E-06 (0.13)	5.47 E-06 (0.03)	9.87 E-07 (0.09)	1.07 E-05 (0.05)
170	nearest offsite office	1.25 E-06 (0.21)	1.08 E-06 (0.07)	1.96 E-07 (0.08)	2.53 E-06 (0.11)
700	LASL boundary	2.75 E-08 (0.19)	3.12 E-08 (0.19)	9.61 E-09 (0.09)	6.83 E-08 (0.12)
1 340	nearest residence	3.54 E-10 (0.30)	5.54 E-10 (0.26)	4.22 E-10 (0.09)	1.33 E-09 (0.14)

^a Detector located 3.7 m above ground level, corresponding to second story of office building. All other detector locations were at 1 m above ground level.

TABLE X
COMPARISON OF CALCULATED SKYSHINE DOSE RATES FOR 3×10^{15} n/s SOURCE

Distance from source (m)	2.29-m-THICK ROOF			2.13-m-THICK ROOF	
	TWOTRAN-FC/ MCNP	ONETRAN-DA Maximum	LINDENBAUM	ANISN/MORSE	ONETRAN-DA Maximum
			mrem/occupational yr		
8	910 673 ^a	2 724	1 630 ^a	-	8 010
15.4	820 508 ^a	1 275	847 ^a	130	3 782
150	21 8.4 ^c	27	87 ^a	5.9	87
270	5.1	3	-	2.7	13
			mrem/yr		
700	0.60	-	-	0.39	-
1 340	0.012	-	-	0.009	-

^a Neutron contribution only.

$$N(r) = N(0) e^{-(t-t \sec \theta)/\lambda}$$

The geometry assumed for the source characterization is shown in Fig. 5. The average relaxation length, λ , was taken as 14.3 cm. The values for t and H were 2.13 and 5.79 m, respectively, and the roof radius was 7 m. It is for this set of conditions and using these procedures that the leakages quoted,

Fig. 5. Geometry for ANISN-MORSE source characterization.

4.59×10^8 n/s and 4.11×10^9 gamma rays/s, in Sec. III.B were obtained. The second step in the analysis was to use the source characteristics inferred from the ANISN run as input to the MORSE-CG Monte Carlo code¹⁷ and compute skyshine dose rates for selected locations at ground level. A 0.2-m-thick soil region was specified at ground level and the air regions extended to about 1 500 m in the radial and vertical directions.

An estimate for the skyshine from neutrons can be obtained using an approach set forth by Lindenbaum.¹⁸ The approximation made is that the ratio of skyshine flux to the unattenuated sideward-directed flux at a distance of r meters is $(\Omega/4\pi)(r/60)$ where Ω is the solid angle formed by the enclosure walls and the neutron source. It is important to point out that the skyshine flux so computed is for an enclosure with no roof. The attenuation provided by the roof must be accounted for in any practical situation. For the problem at hand, the value of $\Omega/4\pi$ is 0.27, corresponding to the solid angle subtended by a 7-m-radius disk at a point on the disk axis and 3.66 m from the disk. Observe that the quantity $(\Omega/4\pi)(r/60)$ becomes greater than unity for r greater than 223 m, thus implying a skyshine flux greater than the direct unattenuated flux. Therefore, the approximation

breaks down for r greater than 223 m; also, the estimates become less reliable as the latter radius is approached. Assuming that the latter approximation holds if the term "dose rate" is substituted for "flux," we computed the neutron skyshine dose rates for the first three detector points in Table IX. The attenuation afforded by the 2.29-m-thick roof was estimated using 0.35 m for the dose attenuation decade length in concrete, a value derived from our bulk shielding studies; this gives an attenuation of 3.4×10^6 . The neutron skyshine dose-rate estimates using Lindenbaum's method are listed in Table X. For comparison purposes, we also give the neutron contribution to the total dose rate for the MCNP calculations. It should be stated that we have not used the Lindenbaum method to its full capability; the formula we have used is extremely simplified. Much better results, valid for larger distances than we have indicated, can be obtained by using the analytical treatment discussed in Lindenbaum's review article.¹⁹ We did not pursue this matter further because of the availability of the discrete-ordinates and Monte Carlo codes. Lacking this capability, it would be most worthwhile to utilize Lindenbaum's method to its fullest extent to obtain neutron skyshine dose-rate estimates. The reader is referred to Sec. 4.5.1 of Ref. 11 for further information on estimating neutron and gamma-ray skyshine dose rates.

In comparing the results shown in Table X, we take the TWOTRAN-FC/MCNP as the reference calculation. Based on the ONETRAN-DA calculations for two roof thicknesses, we divide the ANISN/MORSE results by 3 to normalize them to the 2.29-m thick roof results. Consider first the ANISN/MORSE results. We observe that the latter dose rates are lower at all points by factors between 5 and 20. It is suggested that this is due to the lower source intensity, a factor of 10, specified for the MORSE calculations (see Sec. III.B). With this stipulation, the two Monte Carlo calculations are in good agreement. This is as it should be because the same problem was solved using similar methods. Looking next at the ONETRAN-DA figures, we note that they are higher, by factors of 3 to 1.3, for the first three detector points. At the 270-m point, the ONETRAN-DA dose rate is about a factor of 2 lower. To a good approximation, the statement (see Sec. III)

that this procedure should overestimate the dose rates is confirmed. What is surprising is that the difference is only a factor of a few instead of the order-of-magnitude effect which one might have expected. Finally, we compare the dose rates computed using the Lindenbaum approach. For the first three detector points, the neutron dose rates are a factor of 2.4, 1.7, and 10 higher. However, as stated in the previous paragraph, better estimates for the more distant points can be obtained if the detailed analytical expressions derived by Lindenbaum are used in the calculations.

In summary, it is encouraging to find that fairly reasonable preliminary estimates for skyshine effects can be made without resorting to a larger computer effort. It is still true, however, that accurate calculations of skyshine dose rates require the use of sophisticated and time-consuming computer codes. The somewhat favorable comparisons observed in the previous paragraph are valid for the class of problems we have studied. For other types of problems, the experience may not be as good. The approximate methods we chose turned out to be quite useful but, taking Lindenbaum's work as an example, a thorough understanding of the analytical treatment and correct application of the method are essential if the results are to be meaningful. This note of caution cannot be too strongly emphasized.

IV. ACTIVATION CALCULATIONS

The neutron-induced activation of materials in the source cell and the surrounding shield is an important consideration in the design of the INS facility. We have examined three particular aspects of this problem, namely, (1) the activation of small samples of selected materials placed at 0.1 m from the source, (2) the dose rates in the source cell, after shutdown, due to activation of the gypsum + concrete shield, and (3) activation of the air in the source cell. The neutron environment in the source cell is also of interest to material-science experimentalists who will be using the facility. In addition to the source neutrons, there exists in the source cell a sea of lower energy neutrons due to reflections from the surrounding shield materials. We have plotted in Fig. 6 the calculated fluxes in the various energy bands as a function of radius. The source assumed was 3×10^{15} D-T n/s and

Fig. 6. Neutron fluxes in source cell for a source of 3×10^{15} D-T n/s and cell radius of 4.62 m. Shield is 0.31 m of borated (1 wt% B) gypsum plus 1.98 m of concrete.

the cell radius was 4.62 m, this being the radius needed to provide a surface area equal to that of the actual source cell. The shield specified was 0.31 m of borated gypsum (1 wt% B) followed by 1.98 m of concrete. Borating the gypsum resulted in a factor of 100 reduction in the source-cell thermal flux. It is useful to note that if the cell radius is changed, the fluxes, except for the source group, will be inversely proportional to the surface area of the enclosure. The rise in flux at radii below 1 m (Fig. 6) is attributed to air scattering of the primary 14-MeV neutrons. The general expression for the activity A , in units of Bq/cm³ (dis/s·cm³), induced in a material with activation cross section σ_{act} , atom density N , and neutron flux ϕ is

$$A = \lambda n = \sigma_{act} N \phi (1 - e^{-\lambda t}),$$

where

- λ = decay constant = $0.693/T_{1/2}$,
- $T_{1/2}$ = half-life of radioactive product
- n = number of radioactive nuclei per cm³,
- t = irradiation time,
- N = atom density in units of (barn·cm)⁻¹ for σ_{act} in barns, and
- ϕ = flux in units of (n/cm²·s).

Activation cross sections used in this study were obtained from the data published by Alley and Lessler.²⁰ Reactions considered were those leading to gamma-emitting radioactive products with half lives greater than ten min. Atom densities were computed using the elemental and isotopic compositions for the materials of interest; examples will be given later in this report. For each reaction, information was needed on the energies of the photons emitted and their intensities (photons/dis). These data were obtained from the Table of Isotopes.²¹ Photons emitted were binned according to the energy grouping specified in Table III. The expression given above for the activation A is valid at zero shutdown time. For longer shutdown times (T), an additional factor, $e^{-\lambda T}$, must be included.

A. Activation of Materials in Source Cell

As a preliminary guide in the selection of materials, we computed the activity induced in 1-cm³ samples located at a radius of 0.1 m from the source and irradiated for 20 years. Because the 14-MeV neutron flux is dominant at a distance of 0.1 m, only the activation due to the source neutrons was considered. At this radius, the ratio of the source-group flux to the flux below 14 MeV is 10^3 ; the 14-MeV to thermal-flux ratio is 2×10^5 . Using the activities calculated for each reaction together with the corresponding spectral and intensity data for the emitted photons, one can derive a source term $S_{i,j}$ (γ's/cm³·s). The index i refers to the photon group ($i = 1, 12$) and j denotes a particular reaction. Summing over all reactions gives the integrated volumetric source G_i for each group. The expression

$$D = \sum_i \frac{G_i F_i}{4\pi(100)^2}$$

represents the dose rate at 1 m from a 1-cm³ unshielded sample, with F_i being the flux-to-dose rate factor for group i (Table III). Note that for a thick sample, self-shielding effects will be important as will the change in gamma-ray spectrum with shutdown time.

However, the unshielded dose rates computed above will serve as a useful guide in evaluating the activation of materials. We have examined the activation of the following samples: pure copper and tungsten, 6061 aluminum, 304 stainless steel, and low-carbon steel. Atom densities used for these materials are tabulated in Table XI. The unshielded dose rates for shutdown times from 1 h to 1 yr are plotted in Fig. 7. The values shown are for a 10^{15} D-T n/s source operating continuously for a period of 20 yrs. Reactions which contribute significantly to the unshielded dose rates are listed in Table XII.

B. Activation of the Biological Shield

A preliminary survey was made to obtain estimates for the post-shutdown dose rates in the source cell due to the activation of the biological shield. Because gamma rays originating in the outer portion of the shield are not expected to contribute much to the dose rates in the cell, only those photons created in the inner 1.52 m of the shield were considered in the calculations. For a given configuration, the procedure used to estimate these dose rates was as follows. First, the neutron fluxes in the shield were calculated using the ONETRAN-DA code

for spherical geometry. The fluxes in the inner 1.52 m of the shield and the data contained in the activation library were next combined to yield volumetric photon sources as a function of radius and energy group for selected shutdown times. For each shutdown time, and using the appropriate source terms, a gamma-ray transport calculation was made with the ONETRAN-DA code. The photon fluxes so obtained were then converted to dose rates using the flux-to-dose-rate factors given in Table III.

Two configurations were chosen for this study. The first was an all-concrete shield (1.52 m) and the second was 0.305 m of gypsum (1 wt% B) plus 1.219 m of ordinary concrete. For each case, calculations were performed with and without rebar. The rebar geometry used was 0.025 4-m-diam iron rods placed at right angles to each other, in a two-layer configuration, and 0.152 4 m on centers. This corresponds to a homogenized rebar plus concrete (or gypsum) region with 13 vol% iron and 87 vol% concrete (or gypsum). The rebar plus concrete (or gypsum) region was located between radii ($R_0 + 0.0508$) m and ($R_0 + 0.1016$) m, where R_0 is the inner radius of the shield. The constituent atom densities for gypsum are listed in Table XIII. These figures are based on spectrochemical and neutron activation analyses of the gypsum, Oklahoma C base, proposed for use in the INS facility.

Results of our calculations are plotted in Fig. 8 for a 3×10^{15} D-T n/s source and 20-yr operation; the dose rates shown are the maximum computed values. However, the dose rate in the cell does not vary much with radius and can be considered to be constant for all practical purposes. It is seen that for the case without rebar, the gypsum plus concrete shield is far superior to the all-concrete shield. But the addition of rebar eliminates most of the advantage of the gypsum-concrete combination as compared to the all-concrete case; the dose-rate reduction factor is about 3 at best. It must be emphasized that these calculations are exploratory in nature and that assuming the rebar to be present throughout the inner surface (spherical geometry) is unrealistic; if rebar is used, it will probably be required only for the cell roof. Whether rebar will be required for the roof depends, in part, on how the gypsum will be supported, and this will be known when the final engineering design of the biological shield

Fig. 7. Unshielded dose rates at 1 m from a 1-cm³ sample following a 20-yr operation with a 10^{15} D-T n/s source. Source to sample distance is 0.1 m.

TABLE XI
ATOM DENSITIES USED FOR ACTIVATION CALCULATIONS

	<u>Isotope</u>		<u>Atoms/barn-cm</u>		<u>Isotope</u>		<u>Atoms/barn-cm</u>
Pure Copper	CU	63	5.828 E-02		FE	54	3.352 E-03
	CU	65	2.606 E-02		FE	56	5.300 E-02
					FE	57	1.237 E-03
					FE	58	1.792 E-04
					CR	50	7.562 E-04
	W	180	8.006 E-05		CR	52	1.457 E-02
	W	182	1.620 E-02		CR	53	1.652 E-03
Pure Tungsten	W	183	8.806 E-03		CR	54	4.103 E-04
	W	184	1.891 E-02		NI	58	5.521 E-03
	W	186	1.761 E-02	304 Stainless Steel	NI	60	2.110 E-03
					NI	61	8.891 E-05
					NI	62	2.910 E-04
					NI	64	7.275 E-05
	AL	27	5.810 E-02		CO	59	2.018 E-05
	MG	24	5.285 E-04		MN	55	1.732 E-03
	MG	25	6.690 E-05		SI	28	1.562 E-03
	MG	26	7.366 E-05		SI	29	7.961 E-05
	SI	28	3.203 E-04		SI	30	5.251 E-05
	SI	29	1.633 E-05		P	31	6.144 E-05
	SI	30	1.077 E-05		S	32	4.229 E-05
	TI	46	4.074 E-06		S	33	3.339 E-07
	TI	47	3.819 E-06		S	34	1.870 E-06
	TI	48	3.753 E-05		S	36	6.678 E-09
6061 Aluminum	TI	49	2.801 E-06				
	TI	50	2.699 E-06				
	CR	50	4.761 E-06				
	CR	52	9.171 E-05				
	CR	53	1.040 E-05				
	CR	54	2.583 E-06		MN	55	1.825 E-04
	MN	55	8.880 E-05		P	31	1.259 E-05
	FE	54	1.182 E-05		S	32	1.486 E-05
	FE	56	1.869 E-04		S	33	1.173 E-07
	FE	57	4.362 E-06	Low Carbon Steel	S	34	6.568 E-07
	FE	58	6.318 E-07		S	36	2.346 E-09
	CU	63	4.420 E-05		FE	54	1.142 E-03
	CU	65	1.977 E-05		FE	56	1.806 E-02
	ZN	64	3.040 E-05		FE	57	4.215 E-04
	ZN	66	1.723 E-05		FE	58	6.106 E-05
	ZN	67	2.549 E-06				
	ZN	68	1.156 E-05				
	ZN	70	3.855 E-07				

TABLE XII
ACTIVATION OF MATERIALS BY D-T NEUTRONS - IMPORTANT REACTIONS AND
HALF-LIVES OF GAMMA-EMITTING PRODUCTS

Pure Copper:	$^{65}\text{Cu}(n,2n)^{64}\text{Cu}$	12.9 h
	$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	5.26 yr
Pure tungsten:	$^{184}\text{W}(n,p)^{184}\text{Ta}$	8.7 h
	$^{182}\text{W}(n,2n)^{181}\text{W}$	140 days
	$^{184}\text{W}(n,\alpha)^{181}\text{Hf}$	42.5 days
6061 aluminum:	$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	15.0 h
	$^{24}\text{Mg}(n,p)^{24}\text{Na}$	15.0 h
	$^{55}\text{Mn}(n,2n)^{54}\text{Mn}$	303 days
	$^{66}\text{Zn}(n,2n)^{65}\text{Zn}$	245 days
	$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	5.26 yr
304 stainless steel:	$^{56}\text{Fe}(n,p)^{56}\text{Mn}$	2.58 h
	$^{58}\text{Ni}(n,p)^{58}\text{Co}$	71 days
	$^{58}\text{Ni}(n,d)^{57}\text{Co}$	272 days
	$^{58}\text{Ni}(n,pn)^{57}\text{Co}$	272 days
	$^{55}\text{Mn}(n,2n)^{54}\text{Mn}$	303 days
	$^{60}\text{Ni}(n,p)^{60}\text{Co}$	5.26 yr
Low-carbon steel:	$^{56}\text{Fe}(n,p)^{56}\text{Mn}$	2.58 h
	$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	303 days
	$^{55}\text{Mn}(n,2n)^{54}\text{Mn}$	303 days

is complete. The fact remains, however, that dose rates (after shutdown) in the cell will be of the order of rems/h if rebar is used. Without rebar, and with a gypsum liner, the latter dose rates can be reduced by an order of magnitude.

C. Activation of Air on Source Cell

In order to provide input to the designers of the ventilation system, the gaseous radioactive products resulting from activation of the air in the source cell must be considered. Results were obtained for the production rates of ^{13}N , ^{14}C , ^{16}N , ^{39}Ar , and ^{41}Ar . The reactions which produce these isotopes and their half-lives are listed below:

$^{14}\text{N}(n,2n)^{13}\text{N}$	9.97 min
$^{14}\text{N}(n,p)^{14}\text{C}$	5 730 yr
$^{16}\text{O}(n,p)^{16}\text{N}$	7.1 s
$^{40}\text{Ar}(n,2n)^{39}\text{Ar}$	269 yr
$^{40}\text{Ar}(n,\gamma)^{41}\text{Ar}$	1.83 h

For reference purposes, 25-group cross sections used for these reactions are given in Table XIV. The group boundaries, in terms of the 30-group set (Table II), are also listed in Table XIV.

The starting point for these calculations was to obtain the neutron fluxes (ONETRAN-DA code) in the cell during operation. The geometry and source

TABLE XIII
ISOTOPIC ATOM DENSITIES ($\text{b}^{-1} \cdot \text{cm}^{-1}$) OF GYPSUM USED IN ACTIVATION CALCULATIONS

TI	46	1.911 E-08	TA	180	7.588 E-13
TI	47	1.792 E-08	TA	181	6.323 E-09
TI	48	1.761 E-07	FE	54	2.377 E-07
TI	49	1.314 E-08	FE	56	3.758 E-06
TI	50	1.266 E-08	FE	57	8.769 E-08
SR	84	1.865 E-07	FE	58	1.270 E-08
SR	86	3.297 E-06	CO	59	5.825 E-08
SR	87	2.331 E-06	SI	28	5.634 E-05
SR	88	2.751 E-05	SI	29	2.872 E-06
MG	24	1.064 E-04	SI	30	1.894 E-06
MG	25	1.346 E-05	P	31	3.694 E-06
MG	26	1.482 E-05	S	32	6.314 E-03
CU	63	7.465 E-08	S	33	4.985 E-05
CU	65	3.338 E-08	S	34	2.792 E-04
NA	23	1.603 E-05	S	36	9.970 E-07
K	39	4.341 E-05	CA	40	6.442 E-03
K	41	3.118 E-06	CA	42	4.320 E-05
AL	27	1.802 E-05	CA	43	9.304 E-06
MN	55	5.832 E-07	CA	44	1.382 E-04
CL	35	1.467 E-06	CA	46	1.994 E-07
CL	37	4.692 E-07			
W	180	8.091 E-12			
W	182	1.637 E-09			
W	183	8.900 E-10			
W	184	1.911 E-09			
W	186	1.780 E-09			
CR	50	2.872 E-09			
CR	52	5.532 E-03			
CR	53	6.272 E-09			
CR	54	1.558 E-09			
ZN	64	8.558 E-07			
ZN	66	4.865 E-07			
ZN	67	7.175 E-08			
ZN	68	3.255 E-07			
ZN	70	1.085 E-08			

specifications were the same as those used to calculate the neutron environment in the source cell (Sec. IV). From these computed fluxes, we derived a group volume-averaged flux, $\bar{\phi}$, defined as

$$\bar{\phi}_g = \sum \phi_{g,i} V_i / \sum V_i$$

where V_i is the volume of the i^{th} mesh interval and the summation extends over the volume of the cell. Finally, we compute a production rate, P , using the expression

$$P = \sum_g \bar{\phi}_g \sigma_{\text{act},g} N$$

TABLE XIV
NEUTRON CROSS SECTIONS (b) - FOR AIR ACTIVATION CALCULATIONS

Group	$^{14}\text{N}(n,p)^{14}\text{C}$	$^{14}\text{N}(n,2n)^{13}\text{N}$	$^{16}\text{O}(n,p)^{16}\text{N}$	$^{40}\text{Ar}(n,2n)^{39}\text{Ar}$	$^{40}\text{Ar}(n,\gamma)^{41}\text{Ar}$
1-2	3.61 E-02	6.69 E-03	4.33 E-02	6.44 E-01	1.51 E-03
3	4.02 E-02	3.87 E-03	3.74 E-02	4.42 E-01	1.37 E-03
4	4.06 E-02	1.59 E-04	2.44 E-02	1.91 E-02	1.07 E-03
5-6	2.27 E-02				5.22 E-04
7	4.38 E-02				1.96 E-04
8-9	4.81 E-02				3.33 E-04
10-11	4.90 E-02				4.29 E-04
12	1.45 E-02				4.73 E-04
13	5.24 E-02				9.99 E-04
14	1.51 E-02				2.32 E-03
15	1.50 E-03				2.48 E-03
16	1.50 E-03				6.75 E-03
17	1.64 E-03				1.78 E-02
18	2.39 E-03				9.26 E-03
19	3.89 E-03				3.89 E-02
20	6.45 E-03				8.64 E-03
21	1.07 E-02				6.03 E-03
22	1.74 E-02				8.05 E-03
23	2.89 E-02				1.13 E-02
24	4.74 E-02				1.82 E-02
25	7.73 E-02				2.97 E-02
26	1.29 E-01				4.94 E-02
27	2.13 E-01				8.15 E-02
28	3.50 E-01				1.34 E-01
29-30	1.82				6.20 E-01

Fig. 8. Dose rates in source cell following 20-yr operation with 3×10^{15} D-T n/s source for all-concrete and gypsum plus concrete shields.

where g is the group index and N is the atom density of the parent nuclide. The atom densities, in units of $(b \cdot cm)^{-1}$, for air used in these calculations were as follows:

^{14}N	2.994 E-05
^{16}O	8.062 E-06
^{40}Ar	1.802 E-07

The calculated production rates, expressed in units of Bq/m^3 , were:

^{13}N	8.16 E+08	(2.72 E-02 $\mu\text{Ci/cm}^3$)
^{14}C	7.86 E+09	(2.62 E-01 $\mu\text{Ci/cm}^3$)
^{16}N	1.44 E+09	(4.81 E-02 $\mu\text{Ci/cm}^3$)

³⁹ Ar	4.74 E+08	(1.58 E-02 μ Ci/cm ³)
⁴¹ Ar	7.23 E+06	(2.41 E-04 μ Ci/cm ³)

It should be kept in mind that the above figures are production rates and that the ventilation system will have to be designed to keep the effluent release to the environment below prescribed limits. This can be achieved by proper choices of ventilation rates and holdup time before release from the stack.¹

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